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

June 15, 2012

Mr. Kevin Walsh
Site Vice President, North Region
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 1 – NRC EVALUATION OF CHANGES, TESTS, OR EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS TEAM INSPECTION REPORT 05000443/2012007

Dear Mr. Walsh:

On May 7, 2012, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at the Seabrook Station, Unit 1. The enclosed inspection report documents the inspection results, which were preliminarily discussed on March 29, 2012, with Mr. Paul Freeman and other members of your staff. The final inspection results were discussed during a teleconference exit meeting with Mr. Paul Freeman and other members of your staff on May 7, 2012.

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

This report documents one NRC-identified traditional enforcement Severity Level IV violation. Additionally, a licensee-identified violation which was determined to be of very low safety significance is listed in this report. However, because of the very low safety significance and because these issues were entered into your corrective action program, the NRC is treating the violations as non-cited violations (NCV), consistent with Section 2.3.2 of the NRC's 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 U. S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001, with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at Seabrook Station.

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

Sincerely,

Lawrence T. Doerflein

Lawrence T. Doerflein, Chief
Engineering Branch 2
Division of Reactor Safety

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

Enclosure:
Inspection Report 05000443/2012007
w/Attachment: Supplemental Information

cc w/encl: Distribution via ListServ

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

Sincerely,

/RA/

Lawrence T. Doerflein, Chief
Engineering Branch 2
Division of Reactor Safety

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

Enclosure:
Inspection Report 05000443/2012007
w/Attachment: Supplemental Information

cc w/encl: Distribution via ListServ

Distribution w/encl: **(via e-mail)**
W. Dean, RA (R1ORAMAIL Resource)
D. Lew, DRA (R1ORAMAIL Resource)
J. Clifford, DRP (R1DRPMail Resource)
J. Trapp, DRP (R1DRPMAIL Resource)
C. Miller, DRS (R1DRSMail Resource)
P. Wilson, DRS (R1DRSMail Resource)
A. Burritt, DRP
L. Cline, DRP
A. Turilin, DRP
R. Montgomery, DRP
W. Raymond, DRP, SRI
J. DeBoer, Acting RI
A. Cass, DRP, Resident AA
M. McCoppin, RI, OEDO
RidsNrrPMSeabrook Resource
RidsNrrDorlLpl1-2 Resource
ROPreports Resource
D. Bearde, DRS

DOCUMENT NAME: G:\DRS\Engineering Branch 2\Mangan\Seabrookmodsreport2012007.docx
ADAMS ACCESSION NUMBER: ML12170A099

☑ SUNSI Review		☑ Non-Sensitive ☐ Sensitive		☑ Publicly Available ☐ Non-Publicly Available	
OFFICE	RI/DRS	RI/DRP	RI/DRS	RI/DRS	
NAME	KMangan	ABurritt	WCook	LDoerflein	
DATE	5/31/12	6/14/12	6/7/12	6/15/12	

* See Previous Concurrence OFFICIAL RECORD COPY

U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	50-443
License No.:	NPF-86
Report No.:	05000443/2012007
Licensee:	NextEra Energy Seabrook, LCC
Facility:	Seabrook Station, Unit 1
Location:	Seabrook, NH 03874
Inspection Period:	March 12 through March 29, 2012 (on-site inspection period) April 2 through May 7, 2012 (in-office review) (part time)
Inspectors:	K. Mangan, Senior Reactor Inspector, Division of Reactor Safety (DRS), Team Leader L. Scholl, Senior Reactor Inspector, DRS M. Orr, Reactor Inspector, DRS
Approved By:	Lawrence T. Doerflein, Chief Engineering Branch 2 Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000443/2012007; 3/12/12 – 5/7/12; Seabrook Station, Unit 1; Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications.

This report covers a two week on-site inspection period of the evaluations of changes, tests, or experiments and permanent plant modifications. Following the on-site inspection, additional in-office review was performed. The inspection was conducted by three region based engineering inspectors. One Severity Level IV (SL-IV) violation was identified and characterized as a non-cited violation. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

<u>NRC-Identified and Self-Revealing Findings</u>

- <u>SL-IV</u>. The team identified a Severity Level IV non-cited violation of 10 CFR 50.59 in that NextEra made changes to an analysis listed in the Technical Specifications (TS) without obtaining a license amendment. The team found that prior to replacing incore probe detectors used to determine neutron and gamma flux in the core NextEra added two correction factors to the S3FINC code in order to adjust the signals produced by the detectors. The changes were made under the 10 CFR 50.59 process. The team also found that a third correction factor had been applied in 2002 to address a divergence between the measured and predicted flux levels. In this case the changes were made without using the 10 CFR 50.59 process. The team's review determined that in 1992 the licensee had evaluated the methodology used to convert the detector signal to a flux map via YAEC-1855PA, Seabrook Station Unit 1 Fixed Incore Detector System Analysis. This analysis had been submitted to the NRC as part of License Amendment Request 92-14. The NRC had evaluated and approved the analysis in a Safety Evaluation associated with License Amendment 27. The analysis was then listed in Section 6.8.1.6.b.10 of the TS. The team determined that the changes impacted the analysis and assumptions used as the basis for the conclusions reached in the NRC Safety Evaluation. Following identification of the issue, NextEra entered the issue into the corrective action program, performed an operability assessment, and planned to correct the discrepancy between the license and plant configuration.

  The team determined that the failure to perform an assessment of the changes made to the plant in 2002 and that the incorrect conclusion reached in the 2010 10 CFR 50.59 evaluation constituted a performance deficiency. Because the issue impacted the ability of the NRC to perform its regulatory function, traditional enforcement was used to disposition the violation. The issue was considered more than minor because the changes involved a change to the TS, and the NRC review and approval was required prior to implementing. The team used IMC 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," to evaluate the risk significance of the issue. The team determined the issue adversely impacted the Barrier Integrity Cornerstone and had very low safety significance (Green) per Table 4a in the Phase 1 screening because it only potentially impacted the fuel barrier. (Section 1R17.2.7.b)

Licensee Identified Violations

- A Severity Level IV violation that was identified by NextEra was reviewed by the team. Corrective actions taken or planned by NextEra have been entered into NextEra's corrective action program. The violation and corrective action tracking number are documented in Section 4OA7 of this report.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R17 Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications (IP 71111.17)

.1 Evaluations of Changes, Tests, or Experiments (25 samples)

a. Inspection Scope

The team reviewed three safety evaluations to determine whether the changes to the facility or procedures, as described in the Updated Final Safety Analysis Report (UFSAR), had been reviewed and documented in accordance with 10 CFR 50.59 requirements. In addition, the team evaluated whether NextEra had been required to obtain U.S. Nuclear Regulatory Commission (NRC) approval prior to implementing the changes. The team interviewed plant staff and reviewed supporting information including calculations, analyses, design change documentation, procedures, the UFSAR, the Technical Specifications (TS), and plant drawings to assess the adequacy of the safety evaluations. The team compared the safety evaluations and supporting documents to the guidance and methods provided in Nuclear Energy Institute (NEI) 96-07, "Guidelines for 10 CFR 50.59 Evaluations," as endorsed by NRC Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," to determine the adequacy of the safety evaluations.

The team also reviewed a sample of twenty-two 10 CFR 50.59 screenings for which NextEra had concluded that a safety evaluation was not required to be performed. These reviews were performed to assess whether NextEra's threshold for performing safety evaluations was consistent with 10 CFR 50.59. The sample included design changes, calculations, and procedure changes.

The team reviewed the safety evaluations that NextEra had performed and approved during the time period covered by this inspection (i.e., since the last plant modifications inspection) not previously reviewed by NRC inspectors. The screenings and applicability determinations were selected based on the safety significance, risk significance, and complexity of the change to the facility.

In addition, the team compared NextEra's administrative procedures used to control the screening, preparation, review, and approval of safety evaluations to the guidance in NEI 96-07 to determine whether those procedures adequately implemented the requirements of 10 CFR 50.59. The reviewed safety evaluations and screenings are listed in the attachment.

b. Findings

No findings were identified.

.2 Permanent Plant Modifications (11 samples)

.2.1 Emergency Diesel Generator Engine High Temperature Protection Circuit Modification

a. Inspection Scope

The team reviewed modification EC-12723 that was implemented to improve the reliability of the emergency diesel generators (EDG). The EDGs provide power to the 4.16 kV electrical buses to operate safety equipment in the event offsite electrical power is lost during normal operation, operational transients, or design basis accidents. During an accident that results in the initiation of a safety injection (SI) signal, the EDG protective trips for generator over current, reverse power, loss of field, high lube oil temperature, and high jacket water temperature are automatically bypassed as required by plant technical specifications. However, EDG protective trips associated with high lube oil temperature and high jacket water temperature trips were not bypassed during emergency starts not associated with a safety injection actuation, e.g., manual emergency start or loss-of-offsite power condition. This design change modified the EDG control circuitry to bypass these two high temperature trips for all emergency starts of the EDGs. All engine protective trips remain active for testing or normal starts of the EDGs.

The team reviewed the modification to verify that the design basis, licensing basis, and performance capability of the EDGs had not been degraded by the control circuit modifications. The team interviewed design engineers and reviewed design drawings to determine if the circuit changes met the design and licensing requirements. Additionally, the team reviewed post-modification testing (PMT) results, and associated maintenance work orders to verify that the changes were appropriately implemented. The team also performed a walk down of the EDGs and their associated control panels to verify that annunciators added by with the modification were in accordance with the design and to assess the overall material condition of the systems following the modification work. Finally, the team reviewed affected operating procedures, alarm response procedures, and surveillance test procedures to verify they had been appropriately updated to reflect the post-modification design and operation. The 10 CFR 50.59 screening determination associated with this modification was also reviewed as described in Section 1R17.1 of this report. Documents reviewed are listed in the attachment.

b. Findings

No findings were identified.

.2.2 4.16 kV Bus-3 and Bus-4 62B Time Delay Relay Setpoint Change

a. Inspection Scope

The team reviewed modification EC-2331 that increased the 62B Agastat time delay relay setpoint from 1.2 to 2 seconds. The relays monitor voltage on electrical buses 3 and 4 which provide 4.16 kV power to non-safety equipment that includes the condensate pumps and the start-up feedwater pump, and electrical buses 5 and 6 which provide 4.16 kV power to safety related equipment. During plant operation buses 3 and 5 are powered from the 'A' unit auxiliary transformer (UAT), buses 4 and 6 from the 'B' UAT. When the UAT source is lost, buses 3 and 5 automatically transfer to the 'A' reserve auxiliary transformer (RAT) supplied from offsite power. Buses 4 and 6 will transfer to the 'B' RAT. If the associated RAT is in synchronism with buses 3 and 5 (or 4 and 6), a fast bus transfer to the RAT will occur. If the buses are not in-sync, the fast transfer is blocked. When residual voltage on the buses decays to less than 25 percent of rated voltage a delayed automatic transfer occurs. The 62B time delay relay establishes the permissive circuit that allows the fast or residual voltage transfers to occur. However, if the relay 'times out' all automatic bus transfers are blocked and the buses must be restored by manual operator actions. NextEra determined the 1.2 seconds did not provide sufficient time for the bus voltage to decay below 25 percent voltage because residual voltage on the 5 (6) bus is produced from the coast down of the condensate pump and heater drain pump motors following the loss of the UAT. This system response occurred following a plant event in 2008 when both transfers were blocked. The time delay setpoint was increased by this modification to allow additional time for the bus voltage to decrease below the 25 percent voltage setpoint of the relay to allow the transfer to occur.

The team reviewed the modification to determine if the design basis, licensing basis, or performance capability of the electrical system had been degraded by the modification. In particular, the team verified the time delay change for the non-safety bus transfer scheme would not adversely impact the transfer of the safety-related buses. The team interviewed design engineers, and reviewed design drawings, PMT results, and associated maintenance work orders to determine the impact the modification had on the transfer scheme. The team also verified that associated procedures and drawings had been updated. Finally, the team walked down the affected switchgear to assess the general material condition of the equipment. The 10 CFR 50.59 screening determination associated with this modification was also reviewed as described in Section 1R17.1 of this report. Documents reviewed are listed in the attachment.

b. Findings

No findings were identified.

.2.3 Main Steam Isolation Valve Control Module Replacement

a. Inspection Scope

The team reviewed modification EC-145113 that evaluated and approved the acquisition of replacement main steam isolation valve (MSIV) control modules. The original model of circuit cards were no longer available and this design change evaluated the acceptability of using replacement cards that had the same logic and functional capabilities as the original cards, but were made with new components and circuit board surface mount technology. The main steam isolation valve logic cabinets control the operation of the MSIVs by providing for the opening, closing, and test functions for the valves. The control modules provide a portion of the control circuitry contained in the logic cabinets. The logic cabinets interface with the engineered safeguards feature actuation system to provide MSIV closure when required for steam line isolation.

The team reviewed the modification to determine if the design basis, licensing basis, and performance capability of the MSIV control circuitry could be degraded by the modification. The replacement modules were currently in stock as spares and none have yet been installed in the plant. The team reviewed the associated technical evaluations and factory acceptance test results, and interviewed design engineers to assess whether the modification was consistent with design assumptions. The team also confirmed that the components were procured from a vendor that met the quality assurance program requirements of 10 CFR Part 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants. The 10 CFR 50.59 screening determination associated with this modification was also reviewed as described in section 1R17.1 of this report. Documents reviewed are listed in the attachment.

b. Findings

No findings were identified.

.2.4 Steam Dump System Loss-of-Load Interlock Setpoint Change

a. Inspection Scope

The team reviewed modification EC-12711 that evaluated and implemented a permanent change to the main turbine loss-of-load interlock bistable (1-FW-PB-506-C) setpoint. The steam dump system is designed to reduce the magnitude of plant transients following turbine load reductions or turbine trips. To prevent undesired steam dump valve opening following small load perturbations, the steam dump control system contains an arming circuit interlock that does not allow steam dump valve opening unless the magnitude of the turbine load reductions is equal to or greater than the loss-of-load interlock setpoint. The original loss of load setpoint was set to arm the valves for opening following a 10 percent step load decrease or a sustained ramp load decrease of 5 percent per minute. To prevent undesired arming and opening of the steam dumps, this modification evaluated and implemented a setpoint change to arm the steam dump valve opening circuit following a step load decrease of greater than 15 percent.

The team reviewed the modification to verify that the design and licensing bases of plant systems had not been degraded by the set-point change. The team determined if NextEra had adequately evaluated the impact of the setpoint change on other potentially affected plant analyses and setpoints. The team also verified the calibration procedures had been updated for the revised set-point and reviewed the results of the completed calibration performance that implemented the change in the plant. Finally, the team reviewed the TS to verify that limits in the TS had not been adversely impacted by the change. Additionally, the 10 CFR 50.59 screening determination associated with this modification was reviewed as described in Section 1R17.1 of this report. The documents reviewed are listed in the attachment.

b. Findings

No findings were identified.

.2.5 Replacement of 1-SI-V-32 Motor

a. Inspection Scope

The team reviewed modification EC-145087 that evaluated and replaced the actuator motor for the accumulator isolation valve 1-SI-V-32. The 1-SI-V-32 motor is a safety-related motor qualified to operate in harsh environments. The motor is located in the containment structure and is credited to operate the 1-SI-V-32 isolation valve for some beyond design basis events in order to isolate or un-isolate the safety injection tank. The existing motor was susceptible to internal corrosion and had to be removed from the valve in order to perform an adequate inspection. NextEra replaced the motor with the motor procured during site construction originally intended for use on this valve.

The team reviewed the modification to verify that the design and licensing bases of systems had not been degraded by the motor change. The team reviewed the associated technical evaluations to ensure that the motor had similar electrical characteristics to the installed motor. The team also reviewed post overhaul acceptance test results and interviewed design engineers to assess whether the modification maintained the original design requirements. The team also confirmed that the components were procured from a vendor that met the quality assurance program requirements of 10 CFR Part 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants, and that the replacement motor had been stored in an acceptable storage location. Additionally, the team verified that the maintenance facility that overhauled the replacement motor also had a 10 CFR Part 50, Appendix B, quality assurance program. Finally, the team reviewed the post maintenance testing to determine if the motor and valve would operate as required. The 10 CFR 50.59 screening determination associated with this modification was reviewed as described in Section 1R17.1 of this report. The documents reviewed are listed in the attachment.

b. Findings

No findings were identified.

.2.6 Service Water Strainer Basket Modification

a. Inspection Scope

The team reviewed modification EC-145164 that modified the service water (SW) strainer basket. The basket is credited to collect debris in the SW system prior to the debris impacting safety-related heat exchangers. NextEra determined during SW header inspections that debris was bypassing the strainer basket and found that the support plate that held the basket to the strainer housing was the source of the debris bypass. The modification extended the surface area of the basket structural support plate to prevent debris from bypassing the strainer.

The team reviewed the modification to verify that the design and licensing bases of plant systems had not been degraded by the basket modification. The team interviewed the system engineer and walked down the system to determine the impact the modification had on the SW system. The team also reviewed the post modification testing to ensure the strainer had been tested in accordance with the American Society of Mechanical Engineers (ASME) code requirements for Class 3 piping. Finally, the team discussed the corrective actions with the system engineer to determine what follow-up actions were being taken to verify the modification had corrected the bypass condition. Additionally, the 10 CFR 50.59 screening determination associated with this modification was reviewed as described in Section 1R17.1 of this report. The documents reviewed are listed in the attachment.

b. Findings

No findings were identified.

.2.7 Fixed Incore Detector Assembly Replacement Batch 2

a. Inspection Scope

The team reviewed modification EC-145087281 that evaluated and replaced three permanently installed incore detector assemblies. The fixed incore detectors are used to monitor the neutron flux in the core during power operations. The detector assemblies, made up of five detectors mounted vertically in the center of a fuel assembly, convert the local neutron and gamma flux levels into a proportional electrical signal. The signals from detectors in 290 core locations are sent to the fixed incore detector data acquisition system and subsequently the S3FINC code to develop the neutron flux profile for the entire core. The modification replaced three detector assemblies including the detectors, wiring, and electrical connectors/pigtails at the seal table and containment penetration boundary. Additionally, the modification required two constants be developed and incorporated into the S3FINC code. The constants are used to adjust the electrical signal created by the old and new detectors so that the adjusted signal would equate to a signal that would have been produced by the original detectors when they had been newly installed.

The team reviewed the modification to verify that the design and licensing bases of systems had not been degraded by the detector replacement. The team confirmed that the components were procured from a vendor that met the quality assurance program requirements of 10 CFR Part 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants. The team also reviewed that post modification testing of the equipment verified the proper connection of the wiring. Additionally, the team verified the new seal table connections met the requirement of the ASME code. The team reviewed the associated technical evaluations that evaluated if the changes to the S3FINC code adequately corrected the probe output signal such that the output reflected a consistent measure of neutron flux by the probe. The team also assessed if the flux measurements remained within the measurement error assumed in the Core Operating Limits Report (COLR). The 10 CFR 50.59 screening determination associated with this modification was reviewed as described in Section 1R17.1 of this report. This review included the changes made to the S3FINC code by this modification and changes made to the code in 2002 to determine if the changes constituted a change to the methodology. The documents reviewed are listed in the attachment.

b. Findings

Introduction: The team identified a Severity Level IV non-cited violation of 10 CFR 50.59 in that NextEra made changes that affected the TS without obtaining a license amendment. Specifically, NextEra made changes to the S3FINC code using their 10 CFR 50.59 process; however, the team determined the code had been evaluated and approved by the NRC in an NRC Safety Evaluation and the analysis was listed in Section 6.8.1.6.b.10 of the TS.

Description: The team reviewed a modification that replaced three Platinum incore detector assemblies (15 detectors) and the associated 10 CFR 50.59 screen. The team found that the modification screen referenced a previous 10 CFR 50.59 screen that had been performed in 2010 when two new detectors assemblies had been installed. The team found the previous screen stated, "SPD evaluation will use the S3FINC SPD signal evaluation methodology referenced in the FSAR" and that there was no impact on TSs. The team's review of the modification determined that in order to replace the fixed incore detector assemblies NextEra added two correction factors to the S3FINC code. The first correction factor was the Platinum depletion correction factor which was applied to each batch of detectors to account for detector Platinum 195 depletion. The second correction factor, called the gamma correction factor, was applied to the new detectors to account for differences in surface geometry of the new detector. Additionally, during the inspection the team found that a correction factor had also been applied to the CASMO-3 code in 2002. The code is used to compare the predicted flux to the measured flux. This correction factor, called the neutron conversion factor, was used to account for the transmutation of Platinum 195 to Platinum 196 which results from the exposure of Platinum 195 to a neutron flux. NextEra had concluded that this transmutation caused detectors to have a decreased sensitivity to gamma flux resulting in a divergence between the profiles.

The team reviewed analysis YAEC-1855PA, Seabrook Station Unit 1 Fixed Incore Detector System Analysis, which the licensee had developed "to demonstrate that the fixed incore detector system is comparable in accuracy and functionality to the standard movable incore detector system and to define the uncertainty for these power distribution measurements." The team found that only one correction to the signal from the detectors had been discussed in the analysis. This one-time adjustment, called the raw signal adjustment, was used to adjust the original probe signal outputs to a standard probe signal output in order to account for probe differences resulting from the manufacturing process. Additionally, the team found that the analysis, YAEC-1855PA, had been submitted to the NRC for review and approval as part of License Amendment Request (LAR) 92-14, Incore Detector System.

Finally, the team reviewed the NRC Safety Evaluation in License Amendment 27, which approved LAR 92-14. The team found the Safety Evaluation (SE) stated, in part; "An uncertainty analysis was performed on this data which showed uncertainties of 4.13 percent for Fdh and 5.12 percent for Fxy and Fq. These uncertainties are specific to the analytical physics methods, CASMO-3/SIMULATE-3, used at Yankee, the incore data processing code, FINC, and the Platinum fixed incore detectors currently in use at Seabrook." The SE concluded that "Based on the staff's review of YAEC-1855PA, the staff finds the methods employed to convert the Platinum detector signal to power distribution are mathematically accurate and reasonable from an engineering standpoint." The Licensee Amendment resulted in the addition of YAEC-1855PA to TS Section 6.8.1.6.b.10., and surveillance requirements in TS 3/4.2, "Power Distribution Limits," were changed to allow the use of the fixed incore probes.

The team concluded that the YAEC analysis could not be changed without a license amendment because it is listed as an analysis in the TS. Additionally, the NRC had approved LAR 92-14 in the License Amendment 27 based on an evaluation of the entire signal conversion processing from raw signal to power and the subsequent comparison of measured to predicted signal as described in the analysis. By adding three correction factors to the signal processing of the probes, NextEra had changed the YAEC analysis as approved by the NRC and, therefore, potentially invalidated the conclusions reached in the associated Safety Evaluation. Based on the team's conclusions NextEra entered the issue into the corrective action program, performed an operability assessment, and will evaluate actions to ensure the license and plant configuration are in alignment.

Analysis: The team determined that the failure to perform an assessment of the changes made to the CASMO-3 code in 2002 and the incorrect assessment performed in the 10 CFR 50.59 evaluation related to the 2010 changes to the incore detector assemblies was a performance deficiency. Specifically, in both cases a discussion of the impact the changes had on the analysis listed in the plant Technical Specifications was not performed, and the changes potentially invalidated the conclusions reached in the NRC Safety Evaluation. Because the issue impacted the ability of the NRC to perform its regulatory function, the team evaluated the issue using the traditional enforcement process. The violation was more than minor because there was a reasonable likelihood that the changes requiring the 10 CFR 50.59 evaluation would require NRC review and approval prior to implementation. The team used IMC 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," to evaluate the risk significance of the issue. This finding adversely impacted the Barrier Integrity Cornerstone objective of providing reasonable assurance that physical design barriers (fuel cladding) protect the public from radionuclide releases caused by accidents or events. Specifically, this finding challenged the design control attribute to ensure the fuel cladding is maintained within the established limits of the core operating limit report and reload analysis. The issue was determined to have very low safety significance (Green) per Table 4a in the Phase 1 screening because it only potentially impacted the fuel barrier. Traditional Enforcement violations are not screened for cross cutting aspects.

Enforcement: 10 CFR 50.59 (c)(1)states in part that the licensee may make changes in the facility as described in the FSAR (as updated) without obtaining a license amendment pursuant to 10 CFR 50.90 only if: (i) A change to the TSs incorporated in the license is not required. Contrary to the above in 2002 and in 2010 NextEra made changes that impacted the YAEC-1855PA analysis without obtaining a license amendment and a license amendment was required. Specifically, this analysis was listed in the TS and approved by the NRC and, therefore, a license amendment in accordance with 10 CFR 50.90 was required. In accordance with the NRC Enforcement Manual Section 7.3, "Enforcement of 10 CFR 50.59 and Related FSAR," and NRC Enforcement Policy Section 6.1, the violation was given a Severity Level IV characterization because the issue was evaluated to be of very low safety significance. However, because this violation was determined to be of very low safety significance and was entered into the corrective action program (AR 1761442), this violation was treated as a non-cited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy. (**NCV 05000443/2012007-001, Inadequate 10 CFR 50.59 Evaluation**)

.2.8 New Non-metallic Liner for Spent Fuel Pool

a. Inspection Scope

The team reviewed modification EC-156597 that replaced the liner installed on the stainless steel plates in the cask loading and fuel transfer canal areas of the spent fuel pool (SFP). The original liner had been installed in 2004 in order to provide a water tight barrier between the SFP water and the stainless steel plating and associated welds. The non-metallic liner design function is to prevent the leakage of tritiated water from SFP. Due to indications of degradation, NextEra determined that the liner needed to be replaced. The new non-metallic liner is made up of a primer coating and an epoxy resin applied to the stainless steel liner. In addition to the installation of the liner, the modification established an inspection program to monitor the new liner and the epoxy coupons that were created during the modification.

The team reviewed the modification to verify that the design and licensing bases of plant systems had not been degraded by the replacement of the liner. The team reviewed the installation procedure to verify the surface had been properly prepared and the material had been installed in accordance with the manufacturers' recommendations. The team also reviewed the results of the post installation inspection to determine if deficiencies had been identified, and to determine if corrective actions had been taken to address the deficiencies. The team reviewed NextEra's program that was developed to determine the adequacy of bonding capability of the non-metallic liner to the stainless steel liner prior to installing the product. Additionally, the team reviewed the program for monitoring the liner, which included the testing program for adhesion of the material to coupons placed in the SFP, developed to verify the integrity of the liner following installation. The team conducted this review to ensure that degraded coating would not transit to the reactor vessel during refueling operations. Finally, the 10 CFR 50.59 screening determination associated with this modification was reviewed as described in Section 1R17.1 of this report. The documents reviewed are listed in the attachment.

b. Findings

No findings were identified.

.2.9 Ambient Temperature Monitoring East and West Pipe Chase

a. Inspection Scope

The team reviewed modification EC 2334 that installed temperature indicators in the vicinity of the feedwater isolation valves (FIV). NextEra installed local temperature indicators in order to monitor ambient air temperature at the middle level of each pipe chase. The modification was performed after it was determined that both nitrogen pressure and the hydraulic fluid flow characteristics for the FIVs could be affected by ambient and component/piping temperatures. Specifically, NextEra recognized that low ambient temperatures had an adverse influence on the stroke time of the FIVs. The modification provides data used by operators to monitor FIV operability during the winter months.

The team reviewed the modification to verify that the design basis, licensing basis, and performance capability of the FIVs had not been degraded by the modification. The team interviewed engineering staff and reviewed technical evaluations associated with the modification to determine if the instruments would provide adequate data to support proper operation of the valves. The team reviewed drawings, procedures, and maintenance plans to ensure that they were properly updated or developed. The team also performed walkdowns of both pipe chases to verify the installation was completed as designed and to assess the material condition of the equipment. Additionally, the 10 CFR 50.59 screening determination associated with this modification was reviewed as described in Section 1R17.1 of this report. The documents reviewed are listed in the attachment.

b. Findings

No findings were identified.

.2.10 Substitution of Mobil DTE732 for Mobil 797

a. Inspection Scope

The team reviewed modification EC 2520 authorizing the substitution of Mobil DTE 732 for Mobil 797 oil in all applications at the Seabrook Station. The equipment/applications affected by the change included charging pumps, containment building spray (CBS) pumps, EDG governors, feedwater pumps and turbines, and SI pumps. The substitution was required as a result of Exxon/Mobil discontinuing production of Mobil 797.

The team reviewed the modification to verify that the design basis, licensing basis, and performance capability of the various systems or components had not been degraded by the modification. The team reviewed vendor product literature along with technical evaluations to ensure compatibility of the two products if mixing were to occur in various quantities. The team interviewed engineering staff to determine if compatibility issues had arisen since the introduction of the new product. Additionally, the team reviewed a sampling of affected system maintenance procedures to verify the revised documentation contained the appropriate information. Finally, the team walked down the accessible portions of the SI, CBS and charging systems, and the EDGs to assess their material condition. Additionally, the 10 CFR 50.59 screen associated with this modification was reviewed as described in Section 1R17.1 of this report. The documents reviewed are listed in the attachment.

b. Findings

No findings were identified.

.2.11 Service Water Piping Repair to 1-SW-1814-1-156-24

a. Inspection Scope

The team reviewed modification EC 145189 that repaired a through-wall leak on SW line 1-SW-1814-1-156-24 by full penetration welding of a PMCap on the outside of the pipe over the degraded area. The PMCap is a pre-engineered pipe cap designed to meet ASME Section III Class 3 requirements. The proximity of the repair to piping support No. 1804-SG-02 additionally necessitated a modification to that support. The function of the SW system is to transfer the heat loads from various sources in both the primary and secondary portions of the plant to the ultimate heat sink.

The team reviewed the modification to verify that the design basis, licensing basis, and performance capability of the SW system had not been degraded by the modification. The applicable ASME Boiler and Pressure Vessel Code and interpretations were reviewed to ensure NextEra's justification and design assumptions were acceptable. The team ensured pipe stress calculations were appropriately revised to account for the permanent removal of the east-west restraint function of piping support No. 1804-SG-02 and addressed the revised loading on other pipe supports. The team reviewed the ultrasonic test examination data, the PMCap vendor product and installation drawings, work order packages, and photographs of the finished installation. Additionally, the team interviewed engineers to confirm that the repair satisfactorily functioned in accordance with design assumptions. A walkdown of the accessible system piping was performed to assess material condition. The 10 CFR 50.59 screen associated with this modification was reviewed as described in Section 1R17.1 of this report. The documents reviewed are listed in the attachment.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (IP 71152)

a. Inspection Scope

The team reviewed a sample of condition reports (CR) associated with 10 CFR 50.59 and plant modification issues to determine whether NextEra was appropriately identifying, characterizing, and correcting problems associated with these areas, and whether the planned or completed corrective actions were appropriate. In addition, the team reviewed CRs written on issues identified during the inspection to verify adequate problem identification and incorporation of the problem into the corrective action system. The CRs reviewed are listed in the attachment.

b. Findings

No findings were identified.

4OA6 Meetings, including Exit

The team presented the preliminary inspection results to Mr. Paul Freeman, Site Vice President and other members of NextEra's staff at a meeting on March 29, 2012. Following additional in-office review, which included support from the Office of Nuclear Reactor Regulation (NRR) staff, the team conducted a final exit meeting on May 7, 2012, via teleconference with Mr. Freeman and other members of NextEra's staff to discuss results of the inspection. The team returned proprietary information reviewed during the inspection and verified that this report does not contain proprietary information.

4OA7 Licensee-Identified Violations

The following violation of Severity Level IV was identified by the licensee and is a violation of NRC requirements which meets the criteria of the NRC Enforcement Policy for being dispositioned as a Non-Cited Violation.

- A violation of 10 CFR 50.59 was identified in that NextEra improperly made a change to the UFSAR using the 10 CFR 50.59 process that required NRC review and approval prior to implementation. Specifically, in 2010, NextEra changed a graph in the UFSAR used to determine the reactivity in a fuel bundle based on burn-up so that the fuel could be safely placed in the spent fuel pool. This change to the UFSAR was made using the 10 CFR 50.59 process. However, because this same graph is in the Technical Specifications, the 10 CFR 50.59 process could not be used and a licensee amendment under 10 CFR 50.90 was required. Traditional enforcement applied because the change impeded the regulatory process. The issue is more than minor because the change was made without NRC review and approval prior to implementation. The issue was determined to be of very low safety significance (Green) using IMC 0609, Attachment 4, because it was a spent fuel pool issue that did not result in a loss of SFP cooling, did not involve a fuel handling error, and did not result in a loss of SFP inventory. Subsequent to making the change to the UFSAR, NextEra submitted a license amendment request (LAR 11-04) in January 2012 to obtain approval for the change. NextEra entered the issue into the corrective action program (CR 1744734) for evaluation and resolution.

ATTACHMENT

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

P. Freeman	Site Vice President
R. Noble	Engineering Director
M. Collins	Design Engineering Manager
J. Sobotka	System Design Manager
M. O'Keefe	Licensing Manager
P. Gurney	Reactor Engineer Supervisor
G. Kotkowski	Electrical Design Engineering Supervisor
D. Yates	System Engineer
R. Dean	Principal Engineer I&C
R. Perry	CBM IST Group
C. Cronin	Design Engineer
H. Mentel	Design Engineer
N. Pietrantonio	Design Engineer
V. Brown	Regulatory Assurance
T. Carter	Design Engineer
R. Dean	Design Engineer
C. Mello	Design Engineer
T. Nagle	Design Engineer
T. Schulz	Design Engineer
J. Sweeney	Design Engineer
E. Trump	Fire Protection Engineer
J. Esteves	Design Engineer
K. Randall	Reactor Engineer
A. Merrill	Reactor Engineer

ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

05000443/2012007-001	NCV	Inadequate 10 CFR 50.59 Evaluation

LIST OF DOCUMENTS REVIEWED

10 CFR 50.59 Evaluations

08-005, Turbine Control System Replacement, Rev. 0

09-001, Risk Informed ISI of Class 2 Main Steam and Feedwater Break Excursion Region, Rev. 0

10-003, Spent Fuel Pool Administrative Controls, Rev. 0

10 CFR 50.59 Screened-out Evaluations
03-212, Boral Blistering Non Conformance Report, Rev. 0
08-052, Ambient Temperature Monitoring East and West Pipe Chase, Rev. 0*
08-495, Substitution of Mobil DTE 732 for Mobil 797, Rev. 0*
09-227, Replacement of 1-SI-V-32 Motor, Rev. 0*
09-063, Fixed Incore Detector Assembly Replacement Demonstration, Rev. 2
09-323, Service Water Strainer Basket Modification, Rev. 0*
09-336, SW Piping Repair to 1-SW-1814-1-156-24, Rev. 0*
10-065, Reconciliation of Vortex Issue for the Condensate Storage Tank, Rev. 0
10-091, PORV Block Valve Operation, Rev. 0
10-099, Fixed Incore Detector Assembly Replacement Batch 2 - OR14, Rev. 2*
10-121, EDG Engine Temperature Control Upgrade, Rev. 1
10-136, New Non-Metallic Liner for Spent Fuel Pool, Rev. 0*
10-271, Replacement of SCO-5T Relay for Rx Coolant Pump Locked Rotor Protection, Rev. 0
10-032, Reactor Trip or Safety Injection Emergency Operating Procedure Change, Rev. 0
11-133, Service Water Cooling Tower Operation, Rev. 0
11-243, Service Water Pump 'D' Motor Winding Temperature High, Rev. 0
11-281, MSO Resolution: EFW Flow Control Valve Circuit Changes, Rev. 0
11-350, Revise Emergency Feedwater High Flow Isolation Setpoint, Rev. 0

(* designates a 10 CFR 50.59 screen-out evaluation sample that was also a modification sample)

Modification Packages
EC 12711 (08 MMOD 506), Steam Dump Loss of Load Arming Setpoint Change, DCN 00 & 01*
EC 12723 (08 MMOD 518), Emergency Diesel Engine High Temperature Protection Circuit Modification, DCNs 00-06*
EC 145087, Replacement of 1-SI-V-32 Motor, Rev. 1*
EC 145113, Main Steam Isolation Valve Control Module Replacement, Rev. 0*
EC 145164, Service Water Strainer Basket Modification, Rev. 1*
EC 145189, SW Piping Repair to 1-SW-1814-1-156-24, Rev. 0*
EC 145281, Fixed Incore Detector Assembly Replacement Batch 2 - OR14, Rev. 4*
EC 156597, New Non-Metallic Liner for Spent Fuel Pool, Rev. 0*
EC 2331 (08 MSE 021), 4.16 kV Bus-3 & Bus-4 62B Time Delay Relay Setpoint Change, Rev. 0*
EC 2334 (08 MSE 024), Ambient Temperature Monitoring East and West Pipe Chase, Rev. 0*
EC 2520 (08 MSE 210), Substitution of Mobil DTE 732 for Mobil 797, Rev. 0*

(* designates a modification sample that is also a 10 CFR 50.59 screen-out evaluation sample)

Calculations, Analysis, and Evaluations
0570-021-003, Lubrication Data Sheets/Thermal and Radiation Properties, Rev. 1
08-01003, Apparent Cause Evaluation: Feedwater Isolation Valve FW-V-30 Failed Stroke Time, Rev. 0
CN-CPS-07-30, Seabrook Steam Dump Arming Setpoint Analysis, Rev. 1
C-S-1-45560, SW Spools in Tank Farm/PAB Service Water Piping, Rev. 4
EC-274551, Revise EFW High-Flow Isolation Setpoint, Rev. 0
EE-07-035, Engineering Evaluation – Risk Informed Inservice Inspection of Class 2 Main Steam and Feedwater Break Excursion Region, Rev. 0
EE-08-006, Engineering Evaluation for FWIV Ambient Temperature Assessment, Rev. 0

EQ File 225-03-01M, Environmental Lubrication Evaluation, Rev. 0
NSS-220-01, Environmental Qualification of Electrical Equipment, Rev. 4
PM Technical Basis: 1-EAH-TI-8600/8601 Run to Failure Basis, dated 9/1/09
EC-07-030, Status of Fixed/Movable Incore Detector System, Rev. 1

Condition Reports

00000903	01744734*	01761442*	08-01039
00046104	01748223*	07-00944	08-01197
00091229	01748223*	03-02019	08-01204
00208369	01748224*	07-07449	08-13609
01646426	01748224*	07-11928	
01681835	01749138*	08-00894	
01744075*	01750077*	08-00978	

(* denotes NRC identified during this inspection)

Drawings
118716G52, Sht. 5, Wiring Diagram Power Unit, Rev. 12
1-MAH-B20503, Miscellaneous Air Handling Details, Rev. 15
1-NHY-31002, Electrical Distribution One Line Diagram, Rev. 42
1-NHY-31009, Shts. 1 and 2, 4160V Switchgear Buses 1-3 and 1-4 One Line Diagram, Revs. 14 and 2
1-NHY-310102, Sht. G10/b, DG-1A Monitoring & Auxiliary Control Schematic Diagram, Rev. 11
1-NHY-310102, Sht. G1073d, Diesel Generator 1A Annunciator Auxiliary Relays Schematic Diagram, Rev. 10
1-NHY-310102, Sht. G19/3d, Diesel Generator 1B Annunciator Auxiliary Relays Schematic Diagram, Rev. 11
1-NHY-310102, Sht. G20b, DG-1B Monitoring & Auxiliary Control Schematic Diagram, Rev. 12
1-NHY-310857, Sht. E93/8a, Emergency Diesel Generator 1-A Start Circuit No. 1 Schematic Diagram, Rev. 9
1-NHY-310857, Shts. E93/8c, e, and f, Emergency Diesel Generator 1-A Start Circuit No. 2 Schematic Diagram, Revs. 11, 8, and 8
1-NHY-310857, Shts. E94/8c, f, and e, Emergency Diesel Generator 1-B Monitor Circuit Schematic Diagram, Revs. 9, 8, and 8
1-NHY-503491, Sht. 2, DG-Diesel Air Start Logic Diagram, Rev. 0
1-NHY503492, DG-Diesel Shut-Down and Emergency Stop Logic Diagram, Rev. 11
1-NHY-506425, EAH Main Steam and Feedwater Pipe Enclosure Control Loop Diagram, Rev. 6
1-SW-D20795, Service Water System Nuclear Detail, Rev. 40
200917-M-0001, Shts. 1-3, Service Water Pipe 1-SW-1814-156-24 PMCap Shop Fabrication Details, Rev. 2
200917-M-0002, Sht. 1, Service Water Pipe 1-SW-1814-156-24 PM Cap Installation Details, Rev. 0
9763-F-500221, Main Steam and Feedwater East Pipe Chase Instrument Piping, Rev. 8
Dwg. 128417, SW Strainer Baskets, Detail, SW-9-10&11, Rev. 11
MSE-08024-2000, Add Temperature Indicators PID MAH-B20503, Rev. 0
Partial Dwg. 5531-SG-7, Installation Detail for Temperature Indicators EAH-TI-8600 and 8601, Rev. 0
PID-1-SW-D20795, Service Water System, Rev. 40
SK-1000, Modification to Support 1-1814-SG-02, Rev. 0

SK-EC145164, Service Water Strainer Basket Modification, Rev. 1
SK-EC145189-2000, SW System Piping Repair 1-SW-1814-1-156-24, Rev. 0
SW-1827-08, Sht. 1, Service Water 1-SW-1827-1, Rev. 13

Licensing Documents
Amendment No. 27 to Facility Operating License NPF-86, Incore Detector System – License Amendment Request 92-14, dated 12/22/93
Amendment No. 6 to Facility Operating License NPF-86, Seabrook Station, Unit No. 1, dated 8/27/91
Letter from North Atlantic Energy Services to USNRC, "License Amendment Request 92-14, Incore Detector System," dated 11/25/92
Letter from North Atlantic Energy Services to USNRC, "Response to Request for Additional Information: License Amendment Request 92-14," dated 7/2/93
Letter from North Atlantic Energy Services to USNRC, "Response to Request for Additional Information: License Amendment Request 92-14," dated 11/24/93
Seabrook Station Final Safety Analysis Report, Rev. 14A
Seabrook Technical Specifications, Rev. 9-29
YAEC-1855PA, Seabrook Station Unit 1 Fixed Incore Detector System Analysis, dated 10/92

Procedures
5059RM, 50.59 Resource Manual, Rev. 10
DG-CP-75A UA-9558, Panel DG-CP-75 UA-9558 Local Alarm Responses, Rev. 53
DG-CP-76A UA-9568, Panel DG-CP-76 UA-9568 Local Alarm Responses, Rev. 55
E-0, Reactor Trip or Safety Injection, Rev. 49
ES1807.025, Inservice Inspection Visual Examination Procedure, Rev. 5
IS1630.410, DGA-T-CTHA DGA Jacket Cooling Water Outlet Temperature Switch Calibration, Rev. 9
IS1630.412, DGA-T-OTHA DGA Lube Oil Temperature Switch Calibration, Rev. 8
IS1632.412, DGB-T-OTHA DGB Lube Oil Temperature Switch Calibration, Rev. 7
MMDI-10, Economy Engineering Hi-Jacker Model HJ-15-E PM and Inspection, Rev. 2
MS0523.65, Charging Pump Bearing and Mechanical Seal Maintenance, Rev. 1
MX0539.66, B-EDG Mechanical Governor Venting and Setup after Replacement, Rev. 1
OS1000.01, Heatup from Cold Shutdown to Hot Standby, Rev. 30
OS1000.02, Plant Startup from Hot Standby to Minimum Load, Rev. 22
OS1000.03, Plant Shutdown from Minimum Load to Hot Standby, Rev. 19
OS1000.11, Post Trip to Hot Standby, Rev. 8
OS1001.15, PORV Block Valve Operation, Rev. 1
OS1016.05, Service Water Cooling Tower Operation, Rev. 20
OS1023.75, Operation of Feedwater Isolation Valves' Temporary Heating, Rev. 3
OS1026.01, Operation of DG 1A, Rev. 19
OS1026.03, Operating DG 1A Jacket Water Cooling Water System, Rev. 9
OS1026.11, Operating DG 1B Jacket Water Cooling Water System, Rev. 9
OX1426.01, DG 1A Monthly Operability Surveillance, Rev. 25
OX1426.05, DG 1B Monthly Operability Surveillance, Rev. 24
OX1426.20, Diesel Generator 1A 18 Month Operability and Engineered Safeguards Pump and Valve Response Time Testing Surveillance, Rev. 18
OX1426.21, Diesel Generator 1B 18 Month Operability and Engineered Safeguards Pump and Valve Response Time Testing Surveillance, Rev. 14

OX1426.22, Emergency Diesel Generator 1A 24 Hour Load Test and Hot Restart Surveillance, Rev. 13
OX1426.23, Emergency Diesel Generator 1B 24 Hour Load Test and Hot Restart Surveillance, Rev. 12
OX1426.26, DG 1A Semiannual Operability Surveillance, Rev. 14
OX1426.27, DG 1B Semiannual Operability Surveillance, Rev. 15
OX1426.28, Simultaneous Start of Both Emergency Diesel Generators 1A and 1B – Ten Year Operability Surveillance Test, Rev. 2
OX1431.03, Main Control Valve Quarterly Test, Rev. 24
OX1436.07, Main Feedwater System Valve, Cold Shutdown Operability Tests and 18 Month Position Verification, Rev. 6
OX1456.81, Operability Testing of IST Valves, Rev. 6
RS07-01-01, Main Turbine Control Valve Testing, Rev. 1

<u>Work Orders</u>

00627627	0803990	0836824
01198138	0803991	0836825
01198488	0809987	0836827
01210514	0821860	0836828
01207753	0836819	0841789
0803812	0836821	40068717
0803869	0836822	
0803870	0836823	

<u>Miscellaneous</u>

08-9085599, Platinum (Pt) Incore Detector Assembly, dated 10/18/10
4.16 kV Distribution System Detailed System Text, Rev. 4
51-9156705, EQ Review for Seabrook Pt-ICDAs, dated 10/18/10
51-9159575, Use of Seabrook Incore S/N PTICD-0006, dated 10/18/10
Ashcroft Reference Bulletin BM-1, Bimetal Thermometers, Series EI, Grade A (1%)
ASME Section XI, Ch. 28.38, IWA-5250: Corrective Action, Interpretation XI-1-92-31
ASTM D6677-01, Standard Test Method for Evaluating Adhesion by Knife
DRR 94-0109, SW Strainer Cover Bolting and Gasket Improvements, Rev. 3
NAH-07-32, Westinghouse Letter - Turbine Valve Testing Data Review, dated 9/24/07
SSPC-SP 12/NACE No. 5, Surface Preparation and Cleaning of Metals by Waterjetting Prior to Recoating, July 2002

<u>Surveillance and Modification Acceptance Tests</u>

IN1652.410, MS-U-500 Steam Dump Control Calibration, completed 4/15/08
KBU7317, Quality Conformance Test Procedure and Test Documentation for Valve Control Module 6N372, performed 10/8/09
LS0563.11, Testing of Agastat 125 VDC (7000 Series) TDPU Timing Relays, performed 1/23/08 and 3/4/08
OX1456.81, Operability Testing of IST Valves: Valve Stroke Time and Remote Position/Status Light Data: 1-FW-V-30, performed 1/21/08
UT Examination Report: SW-1814-1-156-24, performed 10/27/09

Vendor Manuals
FP25277, DG Governor Modification Instruction Manual, Rev. 2
FP52936, Charging/Safety Injection Pumps Operations Manual, Rev. 23
FP53455, Primary Component Cooling Pump, Rev. 11
Mobil DTE 700 Series, Product Specifications and Properties, dated 6/07
Mobil DTE 790 Series, Product Specifications and Properties, dated 11/06
PDS I-06, Mobil Product Data Sheet – Mobil DTE 797 Oil

LIST OF ACRONYMS

ADAMS	Agencywide Documents Access and Management System
ASME	American Society of Mechanical Engineers
CBS	Containment Building Spray
CFR	Code of Federal Regulations
COLR	Core Operating Limits Report
CR	Condition Reports
DRS	Division of Reactor Safety
EDG	Emergency Diesel Generator
FIVs	Feed Water Isolation Valves
FSAR	Final Safety Analysis Report
IMC	Inspection Manual Chapter
LAR	License Amendment Request
MSIV	Main Steam Isolation Valve
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
PARS	Publicly Available Records
PMT	Post Modification Test
RAI	Request for Additional Information
RAT	Reserve Auxiliary Transformer
SDP	Significance Determination Process
SFP	Spent Fuel Pool
SI	Safety Injection
SLIV	Severity Level IV
SW	Service Water
TS	Technical Specifications
UAT	Unit Auxiliary Transformer
UFSAR	Updated Final Safety Analysis Report