April 30, 2009

Mr. Gene St. Pierre
Vice President, North Region
Seabrook Nuclear Power Plant
FPL Energy Seabrook, LLC
c/o Mr. Michael O'Keefe
P.O. Box 300
Seabrook, NH 03874

SUBJECT: SEABROOK STATION – NRC EVALUATION OF CHANGES, TESTS, AND EXPERIMENTS AND PERMANENT MODIFICATIONS TEAM INSPECTION REPORT 05000443/2009006

Dear Mr. St. Pierre:

On March 26, 2009, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at the Seabrook Station. The enclosed inspection report documents the inspection results, which were discussed on March 26, 2009, with you and other members of your staff.

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

Based on the results of this inspection, no findings of significance were identified.

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 (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 05000443/2009006
w/Attachment: Supplemental Information

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U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	50-443
License No:	NPF-86
Report No:	05000443/2009006
Licensee:	FPL Energy Seabrook, LCC (FPLE)
Facility:	Seabrook Station, Unit No. 1
Location:	Seabrook, New Hampshire 03874
Dates:	March 9, 2009 through March 26, 2009
Inspectors:	K. Mangan, Senior Reactor Inspector, Division of Reactor Safety (DRS), Team Leader S. Pindale, Senior Reactor Inspector, DRS M. Patel, Reactor Inspector, DRS E. Bonney, Reactor Inspector (in-training), DRS J. Rady, Reactor Inspector (in-training), DRS E. Burket, Reactor Inspector (in-training), DRS
Approved by:	Lawrence T. Doerflein, Chief Engineering Branch 2 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000443/2009006; 03/09/2009 - 03/26/2009; Seabrook Station, Unit No. 1; Engineering Specialist Plant Modifications Inspection.

The report covers a two week inspection of the evaluations of changes, tests, or experiments and permanent plant modifications. The inspection was conducted by three region based engineering inspectors, and three inspectors in-training. No findings of significance were identified. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealing Findings

No findings of significance were identified.

B. Licensee-Identified Violations

None.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

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

.1 Evaluations of Changes, Tests, or Experiments (24 samples)

a. Inspection Scope

The team reviewed six 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 Florida Power and Light Energy (FPLE) had been required to obtain NRC approval prior to implementing the change. The team interviewed plant staff and reviewed supporting information including calculations, analyses, design change documentation, procedures, the UFSAR, 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 eighteen 10 CFR 50.59 screenings and applicability determinations for which FPLE had concluded that no safety evaluation was required. These reviews were performed to assess whether FPLE's threshold for performing safety evaluations was consistent with 10 CFR 50.59. The sample of issues inspected that had been screened out by FPLE included design changes, calculations, procedure changes, temporary alterations and setpoint changes.

The team reviewed all the safety evaluations FPLE had performed during the time period covered by this inspection (i.e., since the last modifications inspection). The screenings and applicability determinations were selected based on the risk significance of the associated structures, systems, and components (SSCs).

In addition, the team compared FPLE'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, screenings, and applicability determinations are listed in the attachment.

b. Findings

No findings of significance were identified.

.2 Permanent Plant Modifications (10 samples)

.2.1 Inadvertent ECCS Initiation At-Power Analysis

a. Inspection Scope

The team reviewed a modification that revised the analysis for an inadvertent emergency core cooling system (ECCS) initiation at-power event. In 2005, the NRC issued a license amendment that approved an increase in the licensed core power. The amendment contained a license condition requiring FPLE to evaluate an inadvertent ECCS initiation at-power event either through re-analysis, using an NRC-approved methodology, to show the pressurizer would not become water solid during the event, or qualify the pressurizer power-operated relief valves for steam and water relief. They were currently only qualified for steam relief. The modification FPLE performed to address the licensee condition was the analysis demonstrating that the pressurizer would not become water solid during the inadvertent ECCS initiation at-power event. To meet assumptions in the modification FPLE implemented changes to station procedures and licensed operator training.

The team reviewed the modification to ensure it was consistent with the design and licensing bases. The team reviewed calculations and other technical evaluations to assess whether the modification was consistent with assumptions in the design and licensing bases related to the operation of the ECCS and the reactor coolant system. In particular, the team reviewed the emergency procedure changes and associated operator performance to ensure the required time critical operator actions were reasonable and achievable. Accordingly, the team observed crew performance on the Seabrook simulator and reviewed associated training documents. Finally, the team conducted interviews with engineering and operations staff to verify assumptions in the inadvertent ECCS initiation at-power event analysis. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2.2 Reactor Coolant Pump Seal Leak-off Flow Rate Alarm at 1.1 GPM

a. Inspection Scope

The team reviewed a modification which incorporated new alarms for reactor coolant pump (RCP) seal leak-off flow low indication. The modification included the creation of new setpoints in the plant computer and revision of existing system setpoints to provide alarms for RCP leak-off flow low at 1.1 gpm decreasing (new), and low-low at 0.85 gpm (revised). This modification was required in support of the inadvertent ECCS initiation at-power analysis, because emergency procedures had been revised to instruct operators to stop all charging flow to the reactor coolant system in the event of an inadvertent ECCS initiation (RCP seal injection would be isolated). These alarms provide the necessary information to operators to ensure the associated analysis assumptions (initial conditions) regarding seal leak-off flow (1 gpm minimum) would be satisfied. Additionally, the 10 CFR 50.59 screen associated with this modification was reviewed as described in section 1R17.1 of this report.

The team conducted a review to verify that the design bases, licensing bases and performance capability of the reactor coolant and connected systems, including the RCP seal cooling function, had not been degraded by the modification. The team assessed whether the RCP alarm modification was consistent with assumptions in the inadvertent ECCS initiation at-power analysis and the design and licensing bases. The team verified that drawings, calculations, and procedures were properly updated with revised design information and operating guidance. The team also reviewed current RCP seal cooling and leak-off operating parameters to ensure they were being maintained consistent with design and procedure requirements. In addition, the team interviewed the design engineers and reviewed procedures to ensure the applicable instructions were accurate and consistent with the modification specifications. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2.3 Diesel Generator Jacket Water High Temperature Switch Setpoint Change

a. Inspection Scope

The team reviewed a modification that raised the ‘A’ and ‘B’ emergency diesel generator (EDG) jacket water high temperature switch setpoints. The modification was implemented to prevent a spurious trip of the EDG by increasing the margin to trip on jacket water high temperature. The instrumentation is designed to stop the EDG when the jacket water temperature approaches a temperature where vaporization of the coolant could occur. This trip function is bypassed during the emergency modes of EDG operation (i.e., safety injection, loss of power). The modification increased the setpoint from 190 degrees Fahrenheit (°F) to 195° F. The team’s review was performed to verify that the design bases, licensing bases and performance capability of the EDGs had not been degraded by the modification. Additionally, the equivalent 10 CFR 50.59 screen associated with this modification was reviewed in section 1R17.1 of this report.

The team assessed whether the modification was consistent with assumptions in the design and licensing bases. The review included verifying drawings, calculations, calibration instrumentation sheets and calibration procedures were properly updated. Additionally, post-modification testing data was reviewed to verify the operating jacket water temperature range of the EDGs. The team performed a walkdown of the ‘A’ EDG while running to assess operation of the equipment while in service and observe the jacket water temperature trends and maximum temperature. Finally, the team conducted interviews with engineering staff to determine if the EDGs would function in accordance with the design assumptions. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2.4 Cycle 13 Core Design

a. Inspection Scope

The team reviewed the modification that developed and evaluated the fuel loading pattern for the cycle 13 reactor core. The evaluation was done to verify that safety limits and 10 CFR 50.46 limits would not be violated in the event of a design basis accident. FPLE worked with Westinghouse to design the core and complete the evaluation. The team assessed whether the modification used methodologies that had been approved in the FPLE design and licensing bases. In addition, the 10 CFR 50.59 screen associated with this modification was reviewed as described in section 1R17.1 of this report.

The team reviewed the core operating limit report (COLR), technical specifications and the modification to determine if the methodologies used in the core assessment had been previously approved by the NRC for use at Seabrook Station. Additionally the team reviewed post-modification testing, including rod drop testing and low power physics testing, to determine if the new core was operating as anticipated by the design. Design assumptions were reviewed to evaluate whether they were technically appropriate and consistent with the UFSAR. Finally, the team discussed the modification and design basis with reactor engineers to assess the adequacy of the modification. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2.5 Valve Stem Leak-off Abandonment

a. Inspection Scope

The team reviewed the modification that removed the valve stem leak-off piping of several reactor coolant system (RCS), charging system, and residual heat removal system valves. The valve packing configuration was changed from a double pack design to a single pack design and the leak off piping was cut away from the valve and both ends capped. The modification was installed on selected valves that had been identified with a history of packing leaks. The new design was developed to minimize the amount of reactor coolant leakage from the RCS. The team assessed whether the design and licensing bases, and performance capability of valves had been degraded by the modification. In addition, the 10 CFR 50.59 screen associated with this modification was reviewed as described in section 1R17.1 of this report.

The team assessed the adequacy of the modification process including pre- and post-modification walk-downs, design evaluation, welding techniques, and post-modification testing to determine if the modification maintained the licensing and design bases of the Seabrook Station. Modification design assumptions were reviewed to ensure they were technically appropriate and consistent with the UFSAR, and to verify the safety classification of the system had been maintained. Additionally, the team reviewed selected drawings, analysis, and the UFSAR to determine whether they were properly updated with revised design information. Finally, the team discussed the modification and design basis with system engineers to assess the effectiveness of the modification. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2.6 Service Water Piping Repairs 1-SW-1802-15 & 1-SW-1802-03

a. Inspection Scope

The team reviewed the modification that replaced sections of piping in the service water discharge header. The team determined the modification was performed to replace piping that had been subject to pinhole leaks resulting from corrosion. FPLE had previously performed weld overlays to repair the leaks. Subsequently, the licensee decided to replace the section of piping to prevent additional corrosion from occurring. The team assessed whether the design and licensing bases, and performance capability of the service water system had been degraded during the modification. In addition, the 10 CFR 50.59 screen associated with this modification was reviewed as described in section 1R17.1 of this report.

The team assessed the licensee's actions following replacement of the piping to verify the system had been fully restored to service. The team interviewed system engineers, and reviewed post-modification testing to verify that FPLE had appropriately retested the system to ensure it would operate as required. Finally, the team reviewed material installed in the system to verify the safety classification of the system had been maintained. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2.7 P9 Setpoint Change

a. Inspection Scope

The team reviewed a modification that increased the setpoint for the P-9 interlock from 20 percent of rated thermal power (RTP) to 45 percent of RTP. Upon receipt of a turbine trip signal, when the reactor is operating at power levels greater than the P-9 setpoint, the P-9 interlock logic causes an automatic reactor trip. The modification adjusted the setpoint and changed the status light window to reflect the new setpoint value. The review was performed to verify that the design bases, licensing bases and performance capability of the P-9 interlock had not been degraded by the modification. Additionally, the 10 CFR 50.59 screen associated with this modification was reviewed as described in section 1R17.1 of this report.

The team assessed whether the modification was consistent with assumptions in the design and licensing bases related to item II.K.3.10 of NUREG-0737, "Clarification of Three Mile Island Action Plan Requirements." Item II.K.3.10 requires the licensee to demonstrate that the proposed anticipatory trip modification does not affect the probability of a small-break loss-of-coolant accident (LOCA) resulting from a stuck-open power-operated relief valve (PORV). The team reviewed the licensee's analysis which showed that no PORVs would be required to open during a turbine trip at or below 45 percent of RTP. The team reviewed the methodology and design assumptions associated with the analysis to evaluate whether they were technically appropriate and consistent with the NRC approved methodology and UFSAR. Additionally, drawings, calculations, analyses, procedures, and the UFSAR were reviewed to verify they had been properly updated with revised design information and operating guidance. Finally, the team reviewed the post-modification testing and conducted interviews with engineering staff to verify the affected SSCs would function in accordance with the design assumptions. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2.8 Relay Room Battery Chargers 1-SY-BC-3 & 4 Replacements

a. Inspection Scope

The team reviewed a modification that was performed to replace the relay room battery chargers. The relay room battery chargers provide normal steady state DC power to the 345 kV switching station control and protection systems. The 345 kV switching station aligns offsite power to the plant onsite AC and DC power systems. This modification replaced the existing charger with an AC current rating of 19.9 amps with a charger rated at 21.3 amps. The review was performed to verify that the design bases, licensing bases and performance capability of the switching station had not been degraded by the modification. Additionally, the 10 CFR 50.59 screen associated with this modification was reviewed as described in section 1R17.1 of this report.

The team reviewed the relay room battery charger breaker coordination curves and cable sizing calculations to ensure that the design limits were not exceeded as a result of increase in the charger current draw. The team also reviewed the design assumptions associated with battery charger to evaluate whether they were technically appropriate and consistent with the UFSAR. The team verified that drawings, calculations, analyses, procedures, and the UFSAR were properly updated with revised design information and operating guidance. Post-modification testing was reviewed to determine if the test results showed the affected SSCs would function in accordance with the design assumptions. Finally, the team interviewed the responsible design engineer and walked down the battery chargers to detect any potentially abnormal installation conditions. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2.9 369 Line Protection Relay System Replacement

a. Inspection Scope

The team reviewed a modification that replaced the existing protective relay systems on the 369 Line with a new protection scheme. The 369 Line is a 345 kV transmission line that provides offsite power to Seabrook Station. The new protection scheme was designed to incorporate the same design functions that were part of the original system. The team verified the design bases, licensing bases and performance capability of the protective relays had not been degraded by the modification. Additionally, the 10 CFR 50.59 screen associated with this modification was reviewed as described in section 1R17.1 of this report.

The team assessed whether the replacement of the protective relay system associated with 369 Line was consistent with assumptions in the design and licensing bases. The team reviewed relay system coordination, relay settings and calculations to verify that the design limits were not exceeded. The team verified drawings, calculations, analyses, procedures, and the UFSAR were properly updated with revised design information and operating guidance. Finally, the team reviewed the post-modification testing and conducted interviews with engineering staff to verify the affected SSCs would function in accordance with the design assumptions. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The team reviewed a modification that increased the steam dump arming setpoint from a 10 percent decrease in turbine power to a 15 percent decrease in turbine power. This setpoint change was implemented to avoid arming the steam dump loss-of-load interlock during quarterly performance testing of the high pressure turbine control valve when the unit is operating at 100 percent power. The team verified the design bases, licensing bases and performance capability of the loss-of-load interlock had not been degraded by the modification. Additionally, the 10 CFR 50.59 screen associated with this modification was reviewed as described in section 1R17.1 of this report.

The team reviewed the Westinghouse evaluation performed for FPLE that evaluated the increase in the interlock setpoint. The team verified that component sizing, design transients, and margin to trip evaluations concluded that the design limits and licensing bases were not exceeded, and that approved methodologies and assumptions were used in the evaluation. The team also verified drawings, calculations, analyses, procedures, and the UFSAR were properly updated with revised design information and operating guidance. The team reviewed the post-modification testing and conducted interviews with engineering staff to verify the affected SSCs would function in accordance with the design assumptions. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (IP 71152)

a. Inspection Scope

The team reviewed a sample of condition reports associated with 10 CFR 50.59 and plant modification issues to determine whether FPLE was appropriately identifying, characterizing, and correcting problems associated with these areas and whether the planned or completed corrective actions were appropriate. The condition reports reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

4OA6 Meetings, including Exit

The team presented the inspection results to Mr. Gene St. Pierre and other members of FPLE's staff at an exit meeting on March 26, 2009. The team verified that this report does not contain proprietary information.

ATTACHMENT

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

G. St. Pierre	Site Vice President
M. O'Keefe	Licensing Manager
V. Brown	Sr. Licensing Analyst
N. Mccafferty	Plant Engineering Manager
J. Sobotia	Design Engineering Supervisor
M. Collins	Design Engineering Manager
W. Alcusky	Project Engineer
D. McGonigle	Design Engineering
F. Arn	System Engineer
R. Belanger	Principal Engineer
P. Brangiel	System Engineer
P. Brown	Component Engineer
J. Finnigan	Operations Procedure Writer
P. Gurney	Reactor Engineering Supervisor
G. Kann	Project Engineer
D. Kelly	EOP Coordinator
C. Mello	Design Engineer
R. Belanger	Design Engineer
C. Finnegan	Design Engineer
R. Goodridge	Design Engineer
T. Schulz	Design Engineer

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

None

LIST OF DOCUMENTS REVIEWED

10 CFR 50.59 Evaluations

07-001, UFSAR Methodology Change For the Use of Additional RG 1.92 Combination of Modal, Rev. 0
07-003, DNB Related UFSAR Changes UFCR 07-033, Rev. 0
07-004, Non-LOCA Safety Analysis UFSAR Corrections UFCR 07-036, Rev. 0
07-005, 100 Year Environs Temperature Extremes UFCR 07-042, Rev. 0
08-003, Appendix R Time Critical Operator Actions and RCP Cooling, Rev. 0
08-004, Emergency Diesel Engine High Temp. Protection Circuit Modification, Rev. 0

10 CFR 50.59 Screened-out Evaluations

2007-057, Add step to open PORV block valves if closed to isolate PORV leakage, dated 1/26/2007
2007-112, Removal of OP∆T and OT∆T Auto and Manual Rod Withdrawal Block, Rev. 0
2007-182, Interim RWST Level Guidance, dated 4/4/2007
2008-131, CVCS Controller Logic Modifications, dated 3/14/2008
2008-232, Provided Method to Open RHR Cross Tie Valves in OS1201.10, dated 4/16/2008
2008-281, TS Bases Change 08-05 (Containment Building Spray), dated 4/29/2008
2008-344, OS1014.08, New Procedure to Provide Guidance to Ops to Lower the SFP Level to a BWST, dated 6/3/2008
2008-354, Fuel Consumption Exceeds Storage Tank TS Capacity, dated 6/6/2008
2008-094, Elimination of Unnecessary Radiation Monitor Channels, dated 2/20/2008

Modification Packages

02DCR002, Valve Stem Leak-off Abandonment, DCN 15
05DRC015, Inadvertent ECCS Initiation At-Power Analysis, Rev. 00
06DCR004, P9 Setpoint Change, DCN 03
06DCR009, 369 Line Protection Relay System Replacement, DCN 07
06MSE080, Reactor Coolant Pump Seal Leak Off Flow Rate Alarm at 1.1 GPM, Rev. 00
06MSE093, DG Jacket Water High Temperature Switch Setpoint Change, Rev. 1
07DCR010, Cycle 13 Core Design, DCN 01
07MMOD512, Relay Room Battery Chargers 1-SY-BC-3 & 4 Replacements, DCN 05
08MSE211, SW Pipe Modification, Rev. 0
08MMOD506, Steam Dump System Loss of Load Interlock Setpoint Change, DCN 01

Calculations & Analysis

C-S-1-24006, Inadvertent ECCS Actuation At-Power Non-LOCA Analysis, Rev. 0
C-S-1-50037, Turbine Trip without a Reactor Trip P-9 Setpoint Evaluation, Rev. 0
C-S-1-57002, DG Jacket Water Outlet Temperature, Rev. 0
C-S-1-57057, RWST Level Loops Instrument Uncertainties, Setpoints and TS Numerical Values, Rev. 00
DS39140, Measurement Uncertainty, Rev. 4
ECA99115110, DG-TS-CHTA Change 185F to 190F, Rev. A
EE-08-008, Engineering Evaluation for Spent Fuel Pool Water Level Management during Dry Fuel Storage Operations, Rev. 00 & 01

Conditions Reports (* denotes NRC identified during this inspection)

00005142*	05-05861	07-05144	08-08668
00192489*	05-12639	07-05668	08-08673
00192510 *	06-01470	07-06187	08-09200
00192519*	06-04377	07-08124	08-09316
00192540*	06-05656	07-10346	08-10220
00192614*	06-14211	07-12784	08-11350
00193326*	06-15259	08-00673	09-00984
00193404*	07-00390	08-02983	09-01705
00193542*	07-03741	08-04854	
00193556*	07-04915	08-05693	

Drawings
1-AAH-B20004, Administration & Service Bldg. HVAC Air Flow Detail, Rev. 9
1-BRS-B20857, Boron Recovery System Storage Detail, Rev. 6
1-BRS-D20857, Boron Recovery System Storage Detail, Rev. 5
1-CBS-B20233, Containment Spray System, Rev. 32
1-CS-B20722, Chemical & Volume Control Sys. Heat Exchanger Detail, Rev. 13
1-CS-B20724, Chemical & Volume Control Sys. Letdown Degassifier Detail, Rev. 16
1-CS-B20725, Chemical & Volume Control Charging System Detail, Rev. 26
1-DG-B20466, EDG Cooling Water System Train ‘B’ Detail, Rev. 15
1-FP-B20272, Fire Protection Valve Details, Rev. 10
1-FP-D20271, Fire Protection Details. Rev. 19
1-NHY-310857, Sht. E93-8e, ‘A’ EDG Monitor Circuit Schematic Diagram, dated 1/09
1-NHY-805059, Fuel Storage Building Plan, Rev. 12
1-RC-B20846, Reactor Coolant System Pressurizer, Rev. 14
1-RH-B20660, Residual Heat Removal System Overview, Rev. 3
1-RH-B20663, Residual Heat Removal System Train B Cross-Tie Detail, Rev. 19
1-RH-D20662, Residual Heat Removal System Train A Detail, Rev. 20
1-RH-D20663, Residual Heat Removal System Train B Cross-Tie Detail, Rev. 17
1-SF-B20480, Spent Fuel Pool Cooling and Clean-up System Overview, Rev. 6
1-SF-B20483, Spent Fuel Pool Cooling and Clean-up System Detail, Rev. 13
1-SF-D20483, Spent Fuel Pool Cooling and Clean-up System Detail, Rev. 11
1-SF-D20484, Spent Fuel Pool Cooling and Clean-up System Detail, Rev. 7
9763-F-805085, Fuel Storage Building Sections “A-A” & “B-B” Gen. Arrangement, Rev. 6

Procedures
1-S-S-1-E-0009, Station Requirements for Torque/Bolting Material Substitution, Rev. 9
CP 4.1, Effluent Surveillance Program, Rev. 17
E-0, Reactor Trip or Safety Injection, Rev. 46
ECA-0.0, Loss of All AC Power, Rev. 37
ES-1.1, SI Termination, Rev. 34
F-0.5, Containment, Rev.19
FR-I.1, Response to High Pressurizer Level, Rev. 24
FR-Z.1, Response to High Containment Pressure, Rev. 22
IN1652.410, MS-U-500 Steam Dump Control Calibration, Rev. 04
IS1630.410, DGA-T-CTHA Calibration, Rev. 7
IS1632.410, DGB-T-CTHA Calibration, Rev. 7
L00157J, Job Performance Measure - Stopping All Charging Pumps in Response to High Pressurizer Level and Inadvertent SI in Progress, Rev. 1
MS03-01-01, Installation of AMEX-10/WEKO Seals in Service Water Piping, Rev. 01
MS0517.03, Flange Maintenance, Rev. 3
MS0517.12, Application of Repair and Protective Coating(s), Rev. 04
OS1000.15, Refueling Outage Cooldown, Rev. 02
OS1001.05, Reactor Coolant Pump Operations, Rev. 08
OS1006.04, Operation of the Containment Spray System, Rev. 08
OS1008.01, Chemical and Volume Control System Makeup Operations, Rev. 10
OS1014.08, Spent Fuel Transfer to a Boron Waste Storage Tank, Rev. 00
OS1015.10, Refueling Canal and Cavity Drain, Rev. 06

OS1046.15, Emergency Closure and Alternate Power to 345 kV Switchyard SF_6 Breaker Compressors and Heaters, Rev. 04
OS1201.01, RCP Malfunction, Rev. 14
OS1201.10, Shutdown LOCA, Rev. 08
OX1426.20, Diesel Generator '1A' 18 Month Operability and Engineered Safeguards Pump and Valve Response Time Testing Surveillance, Rev. 05
OX1426.22, Emergency Diesel Generator 1A 24 Hour Load Test and Hot Restart Surveillance, performed 07/02/08
RS1748, Subcritical Physics Testing Using SRWM, Rev. 00
SM 7.12, Radiological Effluent Quality Assurance Program, Rev. 01
SM 7.20, Control of Time Critical Actions, Rev. 01

<u>Work Orders</u>

0209446	0618070	0640131	0731185
0336026	0618071	0641429	0809987
0536750	0628730	0722901	0821859
0617099	0638373	0727087	0840544

<u>Vendor Manuals</u>
Allen Bradley Bulletin 836-837 11906076 - Pressure and Temperature Switches, dated 11/69
Rockwell Automation Vendor Guidance on Drawing 11906212, dated 12/1/08

<u>Miscellaneous</u>
'A' EDG Jacket Water Temperature 24 Hour Run Data, performed 11/29/06 - 12/1/06
'A' EDG Jacket Water Temperature 24 Hour Run Data, performed 6/4/08 - 6/6/08
'B' EDG Jacket Water Temperature (Running) Data, performed 1/1/09 - 3/15/09
'B' EDG Jacket Water Temperature 24 Hour Run Data, performed 1/3/07 - 1/5/07
'B' EDG Jacket Water Temperature 24 Hour Run Data, performed 9/10/08 - 9/12/08
ASME N-302, Tack Welding Section III, Division 1
ASME N-416-1, Alternative Pressure Test Requirement for Welding Repairs or Installation of Replacement Items by Welding, Class 1,2 and 3 Section XI, Division 1
B8337, RCP 'A' No. 1 Seal Leakoff Flow Low (VPRO) - Computer Alarm Display
D4602, RCP 'A' No. 1 Seal Leakoff Flow Low (VPRO) - Computer Alarm Display
DBD-PB-01, Design Basis Document – Plant Barriers, Rev. 03
DW-00-022 (Direct Work Request Form), ERG Feedback, 8/2/00
Updated Final Safety Analysis Report, Seabrook Unit 1
Letter NAH-93-217, Westinghouse to Mr. W. A. DiProfio, NAESCo, Seabrook Station (Inadvertent ECCS Actuation at Power), dated 6/30/93
Letter NF-NA-07-49, Westinghouse to Mr. J.Perryman, FPL, Impact of Incorrect RCCA Trip Curve Assumed in Non-LOCA Safety Analysis, dated 6/29/07
Letter NYN-96058, North Atlantic Energy Service Corporation to NRC, Response to Request for Additional Information for Generic Letter 95-07, dated 8/15/96
Letter, NRC to Mr. M. E. Warner, Seabrook Station, FPLE, (Issuance of Amendment – 5.2 Percent Power Uprate), dated 2/28/05
Letter, Westinghouse to Mr. S. Hale, FPL Energy – Seabrook, RCP Inadvertent Safety Injection Analysis Licensing Report, dated 10/13/05
Regulatory Guide 1.92, Combining Modal Responses and Spatial Components in Seismic Response Analysis, Rev. 1

Regulatory Guide 1.92, Combining Modal Responses and Spatial Components in Seismic Response Analysis, Rev. 2
RES-90-740 (Request for Engineering Services), RHR System, Rev. 00
Spent Fuel Pool Level History Graph, for time period 7/1/08 – 10/2/08
Standing Operating Order 07-002, RWST Interim Level Limitation, dated 4/4/07
Technical Specifications, Seabrook - Unit 1, Amendment 115
UFCR No. 07-036, (UFSAR Change Request) Non-LOCS Safety Analysis UFSAR Corrections, Rev. 0
UFCR No. 07-042, (UFSAR Change Request) Extreme Outside Air Minimum and Maximum Hourly Temperature Change, Rev. 00
UFRC 05-032, Clarification of Valve Packing Configurations, Rev. 0

LIST OF ACRONYMS

AC	Alternating Current
ADAMS	NRC Document System
CFR	Code of Federal Regulations
COLR	Core Operating Limit Report
DC	Direct Current
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
FPLE	Florida Power and Light Energy
GPM	Gallons Per Minute
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Inspection Report
kV	Kilo-volt
LOCA	Loss-of-Coolant Accident
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PARS	Publicly Available Records
PORV	Power Operated Relief Valve
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RTP	Rated Thermal Power
SSC	Structures, Systems and Components
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report