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

July 2, 2010

Mr. Paul Freeman
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 NO. 1 - NRC COMPONENT DESIGN BASES INSPECTION REPORT 05000443/2010006

Dear Mr. Freeman:

On May 20, 2010, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at the Seabrook Station, Unit No. 1. The enclosed inspection report documents the inspection results, which were discussed with you and other members of your staff on May 20, 2010, and during a subsequent telephone call with Mr. M. O'Keefe on June 10, 2010.

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 examined the adequacy of selected components and operator actions to mitigate postulated transients, initiating events, and design basis accidents. The inspection involved field walkdowns, examination of selected procedures, calculations and records, and interviews with station personnel.

This report documents two NRC-identified findings that were of very low safety significance (Green). One of these findings was determined to involve a violation of NRC requirements. However, because of the very low safety significance of the violation and because it was entered into your corrective action program, the NRC is treating the violation as a non-cited violation (NCV) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest the 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 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.

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 the public inspection in the NRC Public Docket Room or from the Publicly Available Records component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

Lawrence T. Doerflein

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

Docket Nos.: 50-443
License Nos.: NPF-86

Enclosure: Inspection Report 05000443/2010006
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 the public inspection in the NRC Public Docket Room or from the Publicly Available Records component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

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

Docket Nos. 50-443
License Nos. NPF-86

Enclosure: Inspection Report 05000443/2010006
w/Attachment: Supplemental Information

cc w/encl: Distribution via ListServ

Distribution w/encl: **(via e-mail)**
S. Collins, RA **(R1ORAMAIL Resource)**
M. Dapas, DRA **(R1ORAMAIL Resource)**
D. Lew, DRP **(R1DRPMAIL Resource)**
J. Clifford, DRP **(R1DRPMAIL Resource)**
D. Roberts, DRS **(R1DRSMail Resource)**
P. Wilson, DRS **(R1DRSMail Resource)**
A. Burritt, DRP
L. Cline, DRP
A. Turilin, DRP
C. Douglas, DRP
W. Raymond, DRP, SRI
J. Johnson, DRP, RI
E. Jacobs, DRP, OA
L. Trocine, RI OEDO
D. Bearde, DRS
RidsNrrPMSeabrook Resource
ROPreportsResource@nrc.gov

SUNSI Review Complete: LTD (Reviewer's Initials) ADAMS ML101830127
DOCUMENT NAME: G:\DRS\Engineering Branch 2\Pindale\SB CDBI 2010-006.doc
After declaring this document "An Official Agency Record" it **will** be released to the Public.
To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

OFFICE	RI/DRS		RI/DRS		RI/DRP		RI/DRS	
NAME	SPindale/		WCook/		ABurritt/		LDoerflein/	
DATE	06/28/10		06/25/10		06/30/10		07/2/10	

OFFICIAL RECORD COPY

U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	50-443
License No.:	NPF-86
Report No.:	05000443/2010006
Licensee:	NextEra Energy Seabrook, LLC
Facility:	Seabrook Station, Unit No. 1
Location:	Seabrook, New Hampshire 03874
Dates:	April 26, 2010 – May 20, 2010
Inspectors:	S. Pindale, Senior Reactor Inspector, Team Leader J. Richmond, Senior Reactor Inspector S. Ibarrola, Reactor Inspector J. Lilliendahl, Reactor Inspector C. Baron, NRC Mechanical Contractor G. Skinner, NRC Electrical Contractor
Approved by:	Lawrence T. Doerflein, Chief Engineering Branch 2 Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000443/2010006; 04/26/2010 – 05/20/2010; Seabrook Station, Unit No. 1; Component Design Bases Inspection.

The report covers the Component Design Bases Inspection conducted by a team of four NRC inspectors and two NRC contractors. Two findings of very low risk significance (Green) were identified, one of which was also considered to be a non-cited violation. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using NRC Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). The cross-cutting aspects were determined using IMC 0305, "Operating Reactor Assessment Program." 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.

NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The team identified a finding of very low safety significance involving a non-cited violation of 10 CFR 50, Appendix B, Criterion XI, "Test Control," in that, NextEra did not assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service were identified and performed in accordance with written test procedures. Specifically, the team determined that interlocks between emergency core cooling system valves were not properly tested to demonstrate that the associated valves will perform satisfactorily in service. In response, NextEra entered the issue into the corrective action program and implemented acceptable interim actions to ensure operability.

  The finding is more than minor because it is associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The team determined the finding was of very low safety significance (Green) because it was not a design or qualification deficiency, did not represent a loss of system safety function, and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. This finding did not have a cross-cutting aspect because the most significant contributor of the performance deficiency was not reflective of current licensee performance. (1R21.2.1.1)

- Green. The team identified a finding of very low safety significance for NextEra's failure to take effective or timely corrective actions regarding the battery sizing calculation for safety related battery loading under station blackout (SBO) conditions. Specifically, although NextEra identified that the SBO battery sizing calculation had significant errors, no action was taken to either formally revise the calculation or ensure it was not used. The team also identified additional errors in the existing calculation. In response, NextEra entered the issue into the corrective action program, performed analysis, and confirmed there were no existing operability issues.

The finding is more than minor because it is associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The team determined the finding was of very low safety significance (Green) because it was not a design or qualification deficiency, did not represent a loss of system safety function, and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. Enforcement action did not apply because the performance deficiency did not involve a violation of regulatory requirements. This finding has a cross-cutting aspect in the area of Problem Identification and Resolution, Corrective Action Program Component, because NextEra did not take appropriate corrective actions to address safety issues in a timely manner. Specifically, NextEra did not take action to either formally revise the SBO battery sizing calculation or to ensure that it was not used since identifying deficiencies approximately four years ago. (IMC 0310, Aspect P.1(d)) (1R21.2.1.2)

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R21 Component Design Bases Inspection (IP 71111.21)

.1 Inspection Sample Selection Process

The team selected risk significant components and operator actions for review using information contained in the Seabrook Station, Unit No. 1, Probabilistic Safety Study (PSS) and the U. S. Nuclear Regulatory Commission's (NRC) Standardized Plant Analysis Risk (SPAR) model. Additionally, the Seabrook Station Significance Determination Process (SDP) Phase 2 Notebook (Revision 2.1a) was referenced in the selection of potential components and operator actions for review. In general, the selection process focused on components and operator actions that had a Risk Achievement Worth (RAW) factor greater than 1.3 or a Risk Reduction Worth (RRW) factor greater than 1.005. The components selected were located within both safety-related and non-safety related systems, and included a variety of components such as pumps, breakers, heat exchangers, transformers, and valves.

The team initially compiled a list of components and operator actions based on the risk factors previously mentioned. Additionally, the team reviewed the previous component design bases inspection report (05000443/2007006) and excluded the majority of those components previously inspected. The team then performed a margin assessment to narrow the focus of the inspection to 15 components, four operator actions and four operating experience items. The team's evaluation of possible low design margin included consideration of original design issues, margin reductions due to modifications, or margin reductions identified as a result of material condition/equipment reliability issues. The assessment also included items such as failed performance test results, corrective action history, repeated maintenance, maintenance rule (a)(1) status, operability reviews for degraded conditions, NRC resident inspector insights, system health reports, and industry operating experience. Finally, consideration was also given to the uniqueness and complexity of the design and the available defense-in-depth margins. The margin review of operator actions included complexity of the action, time to complete the action, and extent-of-training on the action.

The inspection performed by the team was conducted as outlined in NRC Inspection Procedure (IP) 71111.21. This inspection effort included walkdowns of selected components, interviews with operators, system engineers and design engineers, and reviews of associated design documents and calculations to assess the adequacy of the components to meet design basis, licensing basis, and risk-informed beyond design basis requirements. Summaries of the reviews performed for each component, operator action, operating experience sample, and the specific inspection findings identified are discussed in the subsequent sections of this report. Documents reviewed for this inspection are listed in the Attachment.

.2 Results of Detailed Reviews

.2.1 Results of Detailed Component Reviews (15 samples)

.2.1.1 Residual Heat Removal System Pump, 1-RH-P-8-B

a. Inspection Scope

The team inspected the 'B' residual heat removal (RHR) pump to verify that it was capable of meeting its design basis requirements. The RHR pump was designed to provide two primary functions; to provide flow during normal plant cool-down and to provide low pressure safety injection flow during postulated accident conditions. Under accident conditions, the RHR pump was designed to take its suction from the refueling water storage tank (RWST) and/or the containment sump and inject water into the reactor coolant system. The RHR pump was also designed to provide a source of suction water to the high pressure safety injection pumps and charging pumps under accident conditions ("piggy-back" operation).

The team reviewed design calculations to verify the adequacy of the pump design. This review included emergency core cooling system (ECCS) calculations to verify that the RHR pump was capable of providing the required flow during accident scenarios and that it would have adequate net positive suction head and vortex allowance. The team reviewed the pump test procedures, acceptance criteria, and recent results to verify that pump testing would ensure adequate performance under the most limiting conditions. The team reviewed emergency procedures associated with the RHR pump to verify that the operators had appropriate directions under postulated accident conditions. Specifically, the team reviewed the operators' response to a leak in the ECCS system during post-accident operation. In addition, the team reviewed the design and testing of the valve interlock control circuits associated with the pump to verify that these valves would operate as required under all postulated accident conditions, including postulated single failures. The team interviewed system and design engineers to determine if there were any recent issues with the pumps or the associated equipment. The team also reviewed a summary of recent maintenance activities and corrective action documents to assess the material condition of the equipment.

b. Findings

Introduction: The team identified a finding of very low safety significance (Green) involving a non-cited violation of 10 CFR 50, Appendix B, Criterion XI, "Test Control," in that NextEra did not assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service were identified and performed in accordance with written test procedures. Specifically, the team determined that certain interlocks associated with ECCS valves were not properly tested to demonstrate that the valves would perform satisfactorily in service.

Description: The team reviewed the valve interlock circuits between the safety injection (SI) pump minimum flow valves (SI-V-89, 90, 93) and ECCS valves RH-V-35 and RH-V-36 (located between each RHR pump discharge and the suction of the high pressure pumps, and which would be used during piggy-back sump recirculation operation). The three SI pump minimum flow valves were designed to provide a flow path from the SI pumps to the RWST. The valves were designed to be closed by the operators prior to aligning the suction of the SI pumps to the ECCS recirculation mode after a postulated accident by opening valves RH-V-35 and RH-V-36. Closing the SI pump minimum flow valves is required to prevent discharge of radioactive fluid from the containment sump to the RWST. The system design included interlocks to prevent opening of either RH-V-35 or RH-V-36 unless the flow path to the RWST was isolated by closing both SI-V-89 and SI-V-90, or by closing SI-V-93. This valve logic was designed to allow transfer to post-accident ECCS recirculation operation while preventing a potential post-accident release to the RWST in the event of a single failure of a valve or an electrical power supply.

Emergency operating procedure ES-1.3, "Transfer to Cold Leg Recirculation," directs the operators to close all three SI pump minimum flow valves, then to open valves RH-V-35 and RH-V-36 during the transfer to post-accident ECCS recirculation operation. Since the step to close the three SI pump minimum flow valves did not contain a response not obtained provision prior to proceeding to open RH-V-35 and RH-V-36, the team questioned if these interlock circuits were fully tested to ensure that the system would respond as expected under all postulated accident conditions.

Following a review of the associated circuits, NextEra stated that the interlocks for the above ECCS valves were not being fully periodically tested to verify that the valves RH-V-35 and RH-V-36 would be able to open if both SI-V-89 and SI-V-90, or just SI-V-93, failed to close due to a postulated single failure (e.g., power supply failure). NextEra identified that portions of these circuits were being periodically tested for other reasons. The valve in-service test and other motor-operated valve diagnostic activities included testing the active components in the circuits. In fact, the testing demonstrated that at least one path in the parallel circuit was functional as evidenced by the confirmed ability to stroke the RH-V-35 and -36 valves. However, portions of these circuits that could be subject to undetected failures (i.e., broken wire, loose connections, etc.) were apparently not independently tested by those activities. The team identified that a specific undetected failure in the interlock circuit, in conjunction with a postulated single failure of one vital electrical power supply, could result in a condition where the operators could not open either valve RH-V-35 or RH-V-36 from the control room to establish ECCS flow to the safety injection pumps during recirculation mode.

The team noted that UFSAR section 7.1.2.7 documented NextEra's conformance to IEEE Standard 379-1972, "IEEE Trial-Use Guide for the Application of the Single-Failure Criterion to Nuclear Power Generating Station Protection Systems," which stated, in part, that "in the analysis of the effect of each single failure, all potential undetectable failures must be assumed to be in their failed mode." This reference provided the basis for assuming that those portions of the interlock circuit not independently tested would be in a failed mode.

In response to this concern, NextEra initiated AR 00391249. In addition, NextEra indicated that this circuit including the interlock function was adequately tested during pre-operational testing. As a short term action, NextEra issued a Standing Order to operators to provide guidance on alternate options to ensure RH-V-35 and -36 open during the transfer to high pressure recirculation in the event that the interlock fails to operate. NextEra also plans to evaluate periodic testing options to validate the interlock logic. The team found NextEra's actions to be appropriate and concluded that there was reasonable expectation of operability based on existing test results and the actions in the Standing Order.

Analysis: The team determined that the failure to fully periodically test the interlock circuits was a performance deficiency that was reasonably within NextEra's ability to foresee and prevent. The finding was more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure of the untested circuit, in conjunction with a single failure, could affect ECCS function (alignment of high pressure ECCS during recirculation operation). Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function, and was not the result of any willful violation of NRC requirements.

In accordance with NRC Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," a Phase 1 SDP screening was performed and determined the finding was of very low safety significance (Green) because it was not a design or qualification deficiency, did not represent a loss of system safety function, and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. This finding did not have a cross-cutting aspect because the most significant contributor of the performance deficiency was not reflective of current performance.

Enforcement: 10 CFR 50, Appendix B, Criterion XI, "Test Control," requires, in part, that a test program be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is performed in accordance with written test procedures. Contrary to the above, as of May 19, 2010, NextEra did not perform adequate periodic testing to demonstrate that certain interlocks associated with ECCS valves with would perform satisfactorily in the event of a postulated single failure. Because this violation was of very low safety significance (Green) and has been entered into NextEra's corrective action program (AR 00391249), this violation is being treated as a non-cited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000443/2010006-01, Inadequate Test Control of ECCS Valve Interlocks)**

.2.1.2 125 Vdc Battery 1A, 1-EDE-B-1-A, and 125 Vdc Bus 11A, 1-EDE-SWG-11A (2 samples)

a. Inspection Scope

The team reviewed the design, testing, and operation of the 1A station battery and 11A battery 125 Vdc bus to verify that they could perform their design function of providing a reliable source of direct current (DC) power to connected loads under operating, transient, and postulated accident conditions. The team reviewed design calculations to assess the adequacy of the battery's sizing to ensure it could power the required equipment for a sufficient duration and at a voltage above the minimum required for equipment operation. The team reviewed the battery room hydrogen dilution calculation to verify that the hydrogen concentration would stay below flammable limits during normal and postulated accident conditions. The team reviewed battery test results, including discharge tests, to ensure the testing was in accordance with design calculations, plant technical specifications, vendor recommendations, and industry standards; and that the results confirmed acceptable battery performance. Design and system engineers were interviewed regarding the design, operation, testing and maintenance of the battery, bus, and battery charger. The team performed a walkdown of the 1A station battery, 11A bus, battery chargers, and associated distribution panels to assess the material condition of the equipment. Finally, a sample of condition reports was reviewed to ensure NextEra was properly identifying and correcting issues associated with the 1A battery and 11A battery bus.

b. Findings

Introduction: The team identified a finding of very low safety significance (Green) involving the corrective action program requirement for effective and timely corrective actions, in that NextEra did not take effective or timely corrective action regarding the battery sizing calculation for safety related battery loading under station blackout (SBO) conditions. Specifically, although NextEra identified that the SBO battery sizing calculation had significant errors, no action was taken to correct and verify the calculation or ensure it was not used since identifying deficiencies approximately four years ago.

Description: The team reviewed SBC-227, "DC System Evaluation for Station Blackout," which evaluated the adequacy of the four safety related batteries during an SBO. The team questioned several incorrect and non-conservative capacity factors. Capacity factors are provided by the battery manufacturer to convert battery load into battery size (or number of positive battery plates).

NextEra provided the team with condition report (CR) 06-07406, which was created on June 30, 2006. This CR identified the use of non-conservative capacity factors in SBC-227. Although the original calculation showed a positive 1.3 percent margin, when the correct capacity factors were used, the margin dropped to negative 7.9 percent. CR 06-07406 included an evaluation that removed conservatisms in the original calculation and change the methodology for calculating inverter loading. The revised sizing calculation restored the battery margin to positive 3.5 percent.

The team reviewed the evaluation in CR 06-07406 and identified two additional concerns. The evaluation incorrectly neglected the voltage drop from the batteries to the associated safety related inverters. Secondly, the evaluation used measured values for inverter loading instead of calculated values without a basis. The measured values did not account for differences in loading between normal and accident conditions, cyclical loads, instrument error, and inverter voltage tolerance. The team estimated that the magnitude of these additional errors could reduce battery capacity by as much as 5 percent.

CR 06-07406 included a corrective action to formally revise SBC-227. This corrective action was originally set for June 15, 2007. The due date was extended at least twice, and then was changed to a long term corrective action on June 24, 2008. The current due date for the calculation revision was September 24, 2010, and work had not begun to revise the calculation as of the date of this inspection. Procedure PI-AA-205, "Condition Evaluation and Corrective Action," states, "effectively developed corrective actions are critical to the proper resolution of conditions." PI-AA-205 also states, "a critical attribute of an effective corrective action program is timely completion of activities including . . . corrective action(s)." Contrary to PI-AA-205, NextEra did not take effective or timely corrective actions to either formally revise the SBO battery sizing calculation or to ensure that it was not used.

Although NextEra had identified the non-conservative capacity factors in SBC-227, actions were not taken to either revise SBC-227 or to prevent SBC-227 from being used between June 2006 and May 2010. For nearly four years, modifications to the DC system were made without the ability to accurately assess the impact on the battery sizing for SBO. If a significant modification had been made to the DC system, such as adding a large load or jumpering out a cell, then the modification would have been implemented based upon the calculation of record, which was incorrect. In addition, the team identified that the CR 06-07406 evaluation had non-conservative assumptions that likely would have been corrected had the full design calculation revision process been used.

NextEra entered this issue into the corrective action program (AR 391104) and implemented actions to process a timely revision to the SBC-227 calculation. NextEra reviewed the engineering changes to the DC system since 2006 and determined there were no operability issues. The team reviewed NextEra's basis for operability and independently evaluated battery operability. The team similarly concluded that the issues identified did not render either of the batteries inoperable, based on the magnitude of the errors and currently available aging margin.

Analysis: The team determined that the failure to take effective and timely corrective action to correct the SBO battery sizing calculation or to prevent its use was a performance deficiency that was reasonably within NextEra's ability to foresee and prevent. This finding was more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function, and was not the result of any willful violation of NRC requirements.

In accordance with NRC Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," a Phase 1 SDP screening was performed and determined the finding was of very low safety significance (Green) because it was not a design or qualification deficiency, did not represent a loss of system safety function, and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event.

This finding has a cross-cutting aspect in the area of Problem Identification and Resolution, Corrective Action Program Component, because NextEra did not take appropriate corrective actions to address safety issues in a timely manner. Specifically, NextEra did not take action to either formally revise the SBO battery sizing calculation or to ensure that it was not used for almost four years. (IMC 0310, Aspect P.1(d))

Enforcement: This finding does not involve enforcement action because no regulatory requirement violation was identified. No violation of regulatory requirements occurred because, although the batteries are safety related, the affected calculation was for an SBO, which is not a design basis event. Because this finding does not involve a violation of regulatory requirements and has very low safety significance, it was identified as a finding. NextEra took immediate action to verify that engineering changes since 2006 had not invalidated the conclusions of the SBO battery sizing calculation, to verify that the batteries currently have adequate sizing margin, and to enter this issue into their corrective action system (AR 391104). **(FIN 05000443/2010006-02, Inadequate and Untimely Corrective Actions for Station Blackout Calculation Errors)**

.2.1.3 'A' Emergency Diesel Generator (Electrical), 1-DG-1-A

a. Inspection Scope

The team reviewed calculations for both static and transient loading to determine whether the emergency diesel generator (EDG) had sufficient capacity and capability to supply the required accident loads, and whether it could perform within the voltage and frequency limits described in NRC Regulatory Guide 1.9, "Application and Testing of Safety-Related Diesel Generators in Nuclear Power Plants." The team reviewed the generator electrical protective relaying scheme including drawings, calculations, calibration records, and procedures to determine whether the generator was adequately protected and whether its output breaker was subject to spurious tripping. The team reviewed the generator grounding scheme and associated protective relaying to determine whether they were properly coordinated with 4 kV system grounding. The team reviewed maintenance schedules, procedures, and completed work records to determine whether the EDG was being properly maintained. The team reviewed completed surveillances to determine whether the diesel was being tested in accordance with the technical specifications. The team reviewed corrective action histories to determine whether there had been any adverse operating trends. In addition, the team performed a visual inspection of the EDG and its environs to assess material condition and the presence of hazards.

b. Findings

No findings of significance were identified.

.2.1.4 Motor for Emergency Feedwater Pump, 1-FW-P-37-B

a. Inspection Scope

The team reviewed electrical load flow calculations to determine whether the motor-driven emergency feedwater pump motor had adequate voltage to perform its function under degraded voltage conditions. The team reviewed the motor protective relaying scheme, including drawings, calculations, and procedures to determine whether it was adequately protected, and whether it was subject to spurious tripping. The team reviewed the motor control scheme to verify that automatic starting functions were consistent with the design bases. The team reviewed maintenance schedules, procedures, and completed work records to determine whether the motor was being properly maintained. The team reviewed corrective action histories to determine whether there had been any adverse operating trends. In addition, the team performed a visual inspection of the motor to assess material condition and the presence of hazards.

b. Findings

No findings of significance were identified.

.2.1.5 4160 Vac Bus E5, 1-EDE-SWG-5

a. Inspection Scope

The team reviewed bus loading calculations to determine whether the 4160 Vac bus and breakers were applied within their specified capacity ratings under worst case accident loading and grid voltage conditions. The team reviewed the design of the 4160 Vac bus degraded voltage protection scheme to determine whether it afforded adequate voltage to safety related devices at all voltage distribution levels. This included a review of degraded voltage relay setpoint calculations, motor starting and running voltage calculations, and motor control center control circuit voltage drop calculations. The team reviewed procedures and completed surveillances for calibration of the degraded voltage relays to determine whether acceptance criteria were consistent with design calculations, and to determine whether the relays were performing satisfactorily. The team reviewed operating procedures to determine whether the limits and protocols for maintaining offsite voltage were consistent with design calculations. The team reviewed schematic diagrams and calculations for 4160 Vac bus protective relays to ensure that equipment was adequately protected, loads were not subject to spurious tripping, and to determine whether proper coordination was maintained. The team reviewed NextEra's response to NRC Generic Letter 2006-02, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power," to determine whether current procedures for maintaining the availability of offsite power were appropriate. The team reviewed recent corrective action documents and completed maintenance and testing records to determine whether there were any adverse operating trends. In addition, the team performed a visual inspection of the 4160 Vac safety buses to assess material condition and the presence of hazards.

b. Findings

No findings of significance were identified.

.2.1.6 4 kV – 480 Vac Transformer, 1-EDE-X-5-D

a. Inspection Scope

The team reviewed load flow calculations to determine whether the capacity of the transformer was adequate to supply worst-case accident loads. The team reviewed the transformer protective relaying scheme including drawings, calculations, calibration records, and procedures to determine whether the transformer was adequately protected and whether it was subject to spurious tripping. Maintenance schedules, procedures, and completed work records were also reviewed to determine whether the transformer was being properly maintained. The team reviewed corrective action histories to determine whether there had been any adverse operating trends. In addition, the team performed a visual inspection of the transformer cubicle and its environs to assess material condition and the presence of hazards.

b. Findings

No findings of significance were identified.

.2.1.7 Primary Component Cooling Water System Instrumentation

a. Inspection Scope

Primary component cooling water (PCCW) system instruments 1-CC-TE-2271, PCCW heat exchanger outlet temperature, and 1-CC-L-2272, PCCW head tank level, were inspected as a representative sample to ensure the PCCW instruments were capable of performing their design functions. Specifically, 1CC-TE-2271 is designed to prevent operation of a PCCW pump if system temperature is too high, and 1-CC-L-2272 is designed to initiate a PCCW containment isolation on low head tank level. The team reviewed the instrument logic and completed surveillance test results to verify temperature and head tank level indications would provide the required system response; and that the instruments were being calibrated in accordance with the design values. The team interviewed system and design engineers to ensure appropriate assumptions had been used in associated setpoint calculations. The setpoint calculations were reviewed to verify that the indication and actuation settings were correct and based on appropriate design conditions such as maximum expected transient system temperature and minimum expected transient head tank level. The Updated Final Safety Analysis Report (UFSAR), technical specifications, design basis documents, and emergency procedures were reviewed to ensure that design and licensing bases assumptions were met. Condition reports and surveillance test results were reviewed to verify that potential degradation was identified and corrected. Finally, a walkdown was performed to assess the material condition of the instruments and to verify that the installed configuration would support the design basis functions under transient and postulated accident conditions.

b. Findings

No findings of significance were identified.

.2.1.8 Containment Building Spray Valve, 1-CBS-V-11

a. Inspection Scope

The team inspected containment building spray (CBS) injection valve 1-CBS-V-11 as a representative sample to ensure the CBS injection valves were capable of performing their design function. The team reviewed the valve operating logic and completed surveillance test results to verify valve controls would function to provide the desired response to an initiation signal. The team interviewed system and design engineers to ensure appropriate assumptions had been used in associated valve calculations. The valve capability calculations were reviewed to verify that the thrust and torque limits and actuator settings were correct and based on appropriate design conditions such as maximum expected differential pressures. The UFSAR, technical specifications, design basis documents, and emergency procedures were reviewed to ensure that design and licensing bases assumptions were met. Condition reports were reviewed to verify that potential degradation was identified and corrected. Finally, a walkdown was conducted to assess the material condition of the valve and to verify that the installed configuration would support its design basis function under transient and postulated accident conditions.

b. Findings

No findings of significance were identified.

.2.1.9 Reactor Coolant System Power Operated Relief Valve, 1-RC-PCV-456-A

a. Inspection Scope

The team reviewed the mechanical design, testing, and operation of the 'A' power-operated relief valve (PORV), RC-PCV-456-A, to assess whether the PORV could perform its design functions. The PORV is a pilot operated solenoid valve, which requires reactor coolant system pressure to open. The team reviewed applicable portions of the UFSAR, the reactor coolant system design basis document, the technical specifications and associated bases, drawings, and procedures to identify design basis requirements for the PORV. Specifically, the PORV design basis functions included plant pressure control at normal operating temperature and pressure, reactor vessel low-temperature over-pressure protection, and to provide a flow path for primary side feed and bleed operations using the emergency operating procedures.

Surveillance test and operating procedures were reviewed to assess whether the PORV was appropriately tested and operated within required design limits and whether tests adequately verified component functionality. The team compared recent as-found test and inspection results to established acceptance criteria to evaluate the as-found conditions and assess whether those conditions conformed to design basis assumptions and regulatory requirements. Maintenance records were reviewed to assess whether the maintenance was sufficient and whether those activities were performed in accordance with established procedures, vendor recommendations, environmental qualification requirements, and industry standards. The team reviewed drawings and design calculations to verify calculation inputs and assumptions were reasonable and appropriate. The team's review included PORV steam and water relief capacity at expected plant operating conditions, PORV seat leakage history, and assessment of any adverse impact due to seat leakage. In addition to the mechanical review, the team also assessed whether the PORV solenoid would have adequate minimum DC voltage to operate under worse case 125 Vdc battery loading conditions.

The team interviewed design and system engineers regarding the design, operation, testing, and maintenance of the PORV, including recent test results, and operating and maintenance history. Finally, the team reviewed recent system health reports, maintenance work orders, and corrective action documents to determine whether there were any adverse operating trends.

b. Findings

No findings of significance were identified.

.2.1.10 'A' Diesel Generator Air Handling System

a. Inspection Scope

The team reviewed the mechanical design, testing, and operation of the diesel air handling (DAH) system for the 'A' EDG to assess whether it could perform its design functions. The team reviewed applicable portions of the UFSAR, the EDG building heating and ventilation system design basis document, the Technical Requirements Manual, drawings, and procedures to identify the design basis requirements of the DAH system. Specifically, the DAH design basis functions included EDG room air temperature control to less than the equipment maximum operating temperature limit, while the EDG was running fully loaded and the ambient outside air temperature was at the plant design limit. The team reviewed design calculations and operating procedures to assess the adequacy of the DAH system to maintain EDG room temperature. Additionally, the team reviewed NextEra's compensatory ventilation procedure to assess the ability of pre-approved compensatory actions for selected degraded conditions.

The team performed field walkdowns of the 'A' DAH system to independently assess the material condition of the associated equipment. In addition, the team interviewed licensed operators, and design and system engineers regarding the design, operation, testing, and maintenance of the DAH system, including recent operating and maintenance history. Finally, the team reviewed recent system health reports, maintenance work orders, and corrective action documents to determine whether there were any adverse operating trends.

b. Findings

No findings of significance were identified.

.2.1.11 Reactor Coolant Pump Thermal Barrier Cooling

a. Inspection Scope

The team reviewed the mechanical design, testing, and operation of the reactor coolant pump (RCP) thermal barrier cooling (RCPTBC) system to assess whether it could perform its design functions. The team reviewed applicable portions of the UFSAR, the PCCW system design basis document, the Technical Requirements Manual, drawings, and procedures to identify the design basis requirements of the RCPTBC system. The RCPTBC system is a closed loop cooling system located inside the primary containment building and is cooled by the PCCW system. The RCPTBC provides a barrier function between the reactor coolant system and the PCCW system to prevent an RCP thermal barrier leak (e.g., reactor coolant leak into the RCPTBC closed loop) from entering into the PCCW system (a primary containment barrier function). The RCPTBC also functions to transfer heat from the RCP thermal barriers to the PCCW system, and to manually isolate an individual RCP thermal barrier, should a leak occur. The team reviewed design calculations and operating procedures to assess the adequacy of the RCPTBC system to perform the required design basis functions.

The team interviewed licensed operators, and design and system engineers regarding the design, operation, testing, and maintenance of the RCPTBC system, including recent test results, and operating and maintenance history. Additionally, the team reviewed recent system health reports, maintenance work orders, and corrective action documents to determine whether there were any adverse operating trends.

b. Findings

No findings of significance were identified.

.2.1.12 Service Water System Strainers, 1-SW-S-10/11

a. Inspection Scope

The team inspected the service water (SW) system strainers to verify that they were capable of meeting their design basis requirements. The SW strainers were designed to prevent large objects from entering the downstream equipment and potentially clogging or damaging the SW heat exchangers. The strainers were also designed to allow adequate flow to the downstream components under both normal and postulated accident conditions.

The team reviewed design calculations to verify the adequacy of the strainer design. This review included the SW system flow calculation to verify that appropriate pressure drop inputs were included. The strainer differential pressure alarm setpoint analysis was also reviewed to verify the operators would be alerted if a strainer became clogged. The team interviewed system and design engineers to determine if there were any recent issues with the strainers or the associated piping and to verify the results of periodic inspections of the strainers. The team also reviewed a summary of recent maintenance activities and corrective action documents, and performed a walkdown of the strainers and associated piping and valves to assess the material condition of the equipment.

b. Findings

No findings of significance were identified.

.2.1.13 Supplemental Emergency Power System Diesel Generator (Mechanical), 1-SEP-DG-2-B

a. Inspection Scope

The team inspected the 'B' supplemental emergency power system (SEPS) diesel generator to verify that it was capable of meeting its functional requirements. The SEPS diesel generator is non-safety-related, and was designed to provide a backup source of power in the event that the safety-related EDGs were not available. Although the SEPS diesel generator can be used during a station blackout event, it is not credited as a station blackout power source.

The team reviewed the design calculation associated with the diesel fuel capacity to verify the capability of the diesel generator to operate for 24 hours without refueling. The team reviewed the design of the electric starting system, the diesel generator controls, the heating, cooling, and ventilation systems, and the associated breakers. Post-modification testing and periodic tests were reviewed to determine whether the SEPS diesel was capable of supplying its required loads. The team interviewed system and design engineers to determine if there were any recent issues with the SEPS diesel generator or its associated systems and to verify the results of periodic testing of the diesel generator. In addition, the team reviewed NextEra's evaluation of operating experience related to the use of ultra-low sulfur diesel and bio-diesel fuels in the SEPS diesel generator. Finally, the team reviewed a summary of recent maintenance activities and performed a walkdown of the diesel generator and associated systems to assess the material condition of the equipment.

b. Findings

No findings of significance were identified.

.2.1.14 Primary Component Cooling Water System Heat Exchanger, 1-CC-E-17-B

a. Inspection Scope

The team inspected the 'B' PCCW system heat exchanger to verify that it was capable of meeting its design basis requirements. The PCCW heat exchanger was designed to transfer heat from the PCCW system to the service water system during normal, transient, and postulated accident conditions. The heat exchanger was designed to limit the PCCW system supply temperature to specific values under normal and postulated transient conditions.

The team reviewed design calculations to verify the capability of the heat exchanger to transfer the required heat load during postulated accident conditions. This review included the component cooling thermal performance calculation to verify that the PCCW system supply temperature would be acceptable under the most limiting system flow and heat load conditions. The team interviewed system and design engineers to determine if there were any recent issues with the heat exchanger and to verify the results of periodic heat exchanger inspections. The team reviewed the design of the control valves associated with the heat exchanger to verify that the required PCCW flow and temperature would be maintained under design conditions. In addition, the team reviewed the PCCW inventory controls and the capability of the operators to detect a loss of system inventory and provide make-up water. The team also reviewed a summary of recent maintenance activities and corrective action documents, and performed a walkdown of the heat exchangers to assess the material condition of the equipment.

b. Findings

No findings of significance were identified.

.2.2 Detailed Operator Action Reviews (4 samples)

The team assessed manual operator actions and selected a sample of four operator actions for detailed review based upon risk significance, time urgency, and factors affecting the likelihood of human error. The operator actions were selected from a PSS ranking of operator action importance based on RRW and RAW values. The non-PSS considerations in the selection process included the following factors:

- Margin between the time needed to complete the actions and the time available prior to adverse reactor consequences;
- Complexity of the actions;
- Reliability and/or redundancy of components associated with the actions;
- Extent-of-actions to be performed outside of the control room;
- Procedural guidance to the operators; and
- Amount of relevant operator training conducted.

.2.2.1 Establish Feed and Bleed Cooling Given Loss of Main Feedwater, Including Emergency Feedwater and Startup Feed Pump

a. Inspection Scope

The team reviewed the operator action to establish feed and bleed cooling of the reactor coolant system in response to a transient with complete loss of main feedwater and emergency feedwater (e.g., loss of secondary cooling). The team reviewed the PSS to determine how quickly the operators were credited with completing critical operator tasks of actuating safety injection and opening the pressurizer power-operated relief valves to prevent steam generator dryout. The team reviewed the associated emergency and abnormal operating procedures to ensure the operators were provided with clear guidance to perform the action as credited in the Seabrook design and licensing bases. The team evaluated the available time margins to perform the actions to verify the reasonableness of NextEra's operating procedures and risk assumptions. The team conducted a walkdown of the associated annunciators and instrumentation on the main control room panels. In addition, the team observed operator responses during a simulator scenario and interviewed the operators on indications and responses, to assess operator knowledge of and ability to perform the required procedural actions.

b. Findings

No findings of significance were identified.

.2.2.2 Cooldown and Depressurize the Reactor Coolant System to Minimize Reactor Coolant Pump Seal Leak in a Station Blackout

a. Inspection Scope

The team reviewed the operator action to initiate a reactor coolant system cooldown by manually dumping steam at a maximum rate using the atmospheric steam dump valves. The team reviewed the bases of the assumptions used to determine the time required to take appropriate manual action. The team conducted interviews with operators to assess operator knowledge of and ability to operate applicable equipment, and to verify that the action could be accomplished in the required time. The team performed a walkdown of the associated areas to assess equipment material condition; and to ensure the areas, equipment, and instrumentation were accessible. The team interviewed licensed and non-licensed operators and observed an in-field operator job performance measure to assess operator ability and familiarity with performing the backup local manual action. The team reviewed emergency and abnormal operating procedures to verify that the procedures provided clear steps to complete the manual action. In addition, the team observed operator responses during a simulator scenario.

b. Findings

No findings of significance were identified.

.2.2.3 Align Alternate Cooling Water to Charging Pump

a. Inspection Scope

The team reviewed the manual operator action to locally align alternate cooling to the charging pumps during a transient with a loss of the primary component cooling water system. The team reviewed the bases of the assumptions used to determine the time required to take appropriate local manual action to allow the standby charging pump to restart and restore reactor coolant pump seal cooling before the heat up of seal injection above 230°F. The team interviewed licensed and non-licensed operators and observed an in-field operator job performance measure to locally align fire water and demineralized water system valves to evaluate the operators' ability to perform the required actions. In addition, the team walked down the associated piping and valves to assess material condition as well as any likelihood of cognitive or execution errors. The team reviewed emergency and abnormal operating procedures to verify that the procedures provided clear steps to complete the manual action. The team reviewed the CRs and completed surveillances associated with this operator action to assess the overall health of the affected equipment.

b. Findings

No findings of significance were identified.

.2.2.4 Locally Throttle Emergency Feedwater Flow to Steam Generators and Throttle Emergency Feedwater Recirculation Flow During a Station Blackout

a. Inspection Scope

The team reviewed the operator action required to manually control turbine-driven emergency feedwater (EFW) pump flow following a station blackout by opening EFW minimum flow recirculation valves and throttling EFW flow control valves. The team reviewed the PSS studies to determine when and how quickly operators were credited with gaining control of EFW flow to prevent steam generator overfill. The team reviewed the bases of the assumptions used to determine the time required to take appropriate manual action. The team interviewed operators to assess operator knowledge of and ability to operate applicable equipment. The team observed an in-field operator job performance measure to evaluate the operators' ability to perform the required actions. The team reviewed associated operating and emergency procedures to ensure this action could be performed as credited. The team performed a walkdown of the associated areas to ensure the areas, equipment, and instrumentation were accessible.

b. Findings

No findings of significance were identified.

.2.3 Review of Industry Operating Experience and Generic Issues (4 samples)

The team reviewed selected operating experience issues for applicability at Seabrook Station. The team performed a detailed review of the operating experience issues listed below to verify that NextEra had appropriately assessed potential applicability to site equipment and initiated corrective actions when necessary.

.2.3.1 NRC Information Notice 2007-34, Operating Experience Regarding Electrical Circuit Breakers

a. Inspection Scope

The team evaluated NextEra's applicability review and disposition of NRC Information Notice (IN) 2007-34. The NRC issued this IN to inform licensees of operating experience regarding low-, medium-, and high-voltage circuit breakers. In particular, the IN discussed problems caused by deficient maintenance and configuration control practices. The team performed an independent review for the specific issues that were identified as being applicable to Seabrook Station for which NextEra performed a detailed review. The team reviewed NextEra's evaluation, interviewed engineers, and conducted walkdowns of selected components.

b. Findings

No findings of significance were identified.

.2.3.2 NRC Information Notice 2008-20, Failures of Motor Operated Valve Actuator Motors with Magnesium Alloy Rotors

a. Inspection Scope

The team evaluated NextEra's applicability review and disposition of NRC IN 2008-20. The NRC issued this IN to inform licensees of failures and corrective actions for motor-operated valve actuator motors due to corrosion of the magnesium alloy rotors. The team assessed NextEra's evaluation of this potential condition by reviewing specific CRs, reviewing results of motor inspections, and conducting interviews with engineering personnel.

b. Findings

No findings of significance were identified.

.2.3.3 NRC Information Notice 2010-09, Importance of Understanding Circuit Breaker Control Power Indications

a. Inspection Scope

The team reviewed the applicability and disposition of NRC IN 2010-09. The NRC issued this IN to alert licensees to issues with circuit breaker control power, as they related to the failure of a non-safety breaker to open at H.B. Robinson Steam Electric Plant on March 28, 2010. The team reviewed NextEra's evaluation of the issue described in the IN. Specifically, the team reviewed NextEra's CRs and actions documented to address this issue. The team interviewed plant personnel to discuss breaker control power design and indication. This review included monthly surveillance procedures, which require verification of control power availability.

b. Findings

No findings of significance were identified.

.2.3.4 NRC Information Notice 94-66, Overspeed of Turbine-Driven Pumps Caused by Governor Valve Stem Binding

a. Inspection Scope

The team evaluated NextEra's applicability review and disposition of NRC IN 94-66, including Supplement 1 to the IN. The NRC issued IN 94-66 to alert licensees to recent problems regarding binding of governor valves for turbine-driven pumps that have resulted in overspeed trips. Supplement 1 was issued to alert licensees to a potential problem with some licensee actions taken to prevent binding of the valve stems of turbine governor valves and the resulting overspeed trips of the associated turbine-driven pumps. The team assessed NextEra's evaluation of this potential condition by reviewing Nuclear Safety Engineering Report NS95-06. The team also interviewed plant personnel and reviewed an associated modification (MMOD 95-574) that changed the packing material for the governor valve stem.

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 problems that NextEra had previously identified and entered into the corrective action program. The team reviewed these issues to verify an appropriate threshold for identifying issues and to evaluate the effectiveness of corrective actions. In addition, CRs written on issues identified during the inspection were reviewed to verify adequate problem identification and incorporation of the problem into the corrective action system. The specific corrective action documents that were sampled and reviewed by the team are listed in the attachment.

b. Findings

No findings of significance were identified with the exception of the finding discussed in Section 1R21.2.1.2.

4OA6 Meetings, Including Exit

The team presented the inspection results to Mr. Paul Freeman and other members of NextEra staff at an exit meeting on May 20, 2010, and during a subsequent telephone conversation with Mr. M. O'Keefe on June 10, 2010. The team reviewed proprietary information, which was returned to NextEra at the end of the inspection. The team verified that none of the information in this report is proprietary.

ATTACHMENT

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

R. Belanger	Design Engineer
P. Brown	MOV Component Engineer
V. Brown	Senior Licensing Analyst
S. Corcoran	System Engineer
R. Dean	I&C Engineer
S. Fournier	Design Engineer
L. Hansen	System Engineer
R. Jamison	Design Engineer
D. Kelly	Operations EOP Engineer
G. Kim	PRA Engineer
G. Kotkowski	Electrical Design Supervisor
K. Letourneau	Design Engineer
D. McGonigle	Design Engineer
V. Patel	Design Engineer
T. Schulz	Design Engineer
K. Shea	System Engineer
T. Waechter	Operations Assistant Manager

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

NCV	05000443/2010006-01	Inadequate Test Control of ECCS Valve Interlocks (Section 1R21.2.1.1)
FIN	05000443/2010006-02	Inadequate Corrective Actions for Station Blackout Calculation Errors (Section 1R21.2.1.2)

LIST OF DOCUMENTS REVIEWED

Calculations and Evaluations:

00689, Seismic Evaluation of Crain Hoist Chains, 12/3/86
4.3.05.10F, CBS Hydraulic Analysis, Rev. 10
4.3.05.30F, CBS System Setpoints, Rev. 3
4.3.05.31F, RWST Vortex Studies, Rev. 3
4.3.07.26F, Thermal Barrier Loop, Rev. 6
4.3.07.27F, PCCW Instrument Setpoints, Rev. 7

4.3.07.28F, Thermal Barrier Head Tank, Rev. 3
4.3.07.29F, Thermal Barrier Relief Valve Discharge Rates, Rev. 0
4.3.07.33F, Thermal Barrier Hydraulic Characteristics of TB Pumps, Rev. 1
4.3.07.37F, Thermal Barrier Heat Exchanger Performance, Rev. 1
4.3.07-21F, PCCW Water Volume and Head Tank, Rev. 4
4.3.7-59F, PCCW Maximum/Minimum Component Cooling Water Temperature, Rev. 1
4.3.8-72F, Proto-Flo Model of Seabrook Station Service Water System, Rev. 4
6.01.41.04, Battery Room Hydrogen Dilution, Rev. 2
6.01.47.13, Average Temperature in EDG Rooms, Rev. 2
9763-3-ED-00-02-F, Voltage Regulation, Rev. 11
9763-3-ED-00-04-F, 4160 Vac Grounding Resistor and Transformer Sizing, Rev. 3
9763-3-ED-00-13-F, Diesel Generator Grounding, Rev. 2
9763-3-ED-00-14-F, Batteries, Chargers, and Motor Feeds, Rev. 13
9763-3-ED-00-23-F, Protective Relay Coordination and Miscellaneous Relay Settings, Rev. 5
9763-3-ED-00-28-F, Motor Control Circuit Protection, Rev. 7
9763-3-ED-00-31-F, 480 Vac Coordination, Rev. 3
9763-3-ED-00-32-F, Diesel Generator Relay Settings, Rev. 5
9763-3-ED-00-34-F, UPS Loading Class 1E, Rev. 7
9763-3-ED-00-43-F, DC Short Circuit Calculation, Rev. 3
9763-3-ED-00-44-F, 125 Vdc Breaker Coordination, Rev. 2
9763-3-ED-00-66-F, Control Circuit Voltage Drop, Rev. 4
9763-3-ED-00-83-F, Diesel Generator Loading, Rev. 8
9763-5-SP-1F, PCCW Level Error Analysis, Rev. 7
C-S-1-23704, Allowable Leakage from Safety Related Air Supplies, Rev. 3
C-S-1-28009, PCCW Heat Loads and Flow Rates for Various Operating Modes, Rev. 0
C-S-1-57017, PCCW Heat Exchanger Outlet Temperature Uncertainties, Rev. 3
C-S-1-57057, RWST Level Loops Instrument Uncertainties, Setpoints, and TS Values, Rev. 0
C-S-1-80903, Motor-Operated Valve Differential Pressure Calculations, Rev. 0
C-S-1-80904, Motor Operated Valve Sizing, Rev. 2
C-S-1-83610, SW-PDIS-8258/8259 Setpoint Change, Rev. 0
C-S-1-87901, Diesel Generator Room Average Temperature, Rev. 1
C-S-1-E-0130, RWST Time to Vortex, Rev. 2
C-S-1-E-0161, EDG Maximum Allowable Fuel Oil Consumption Rate, Rev. 16
C-X-1-27801, Minimum Stored Fuel for SEPS Diesel Generators, Rev. 0
EE 93-21, Compensatory Actions for Safety-Related HVAC Systems and Components, Rev. 5
EE 95-07, Pressure Locking and Thermal Binding of Gate Valves, Rev. 2
EE-04-024, Operator Action Response Times Assumed in the UFSAR, Rev. 4
FP-57747, Garrett PORV Environmental and Seismic Testing, Rev. 1
MSVCS-FAG-09, Temperatures in the Diesel Generator Building, Rev. 1
NAH-1795, LTOP Over-Pressure Protection Report, 1/83
NSS-220-04, Environmental Qualification of Electrical Equipment, Rev. 3
NSS-220-06, EQ Report for Garret (PORV) Solenoid Operated Pilot Valve, Rev. 3
PM Basis Document for CC-V-395-MOV2, MOV CC-V-395 Diagnostics Testing, 11/3/03
SBC-128, Technical Specifications – Setpoints and Allowable Values, Rev. 14
SBC-227, DC System Evaluation for Station Blackout, Rev. 2
SBC-535, Seabrook ECCS Performance During Post-LOCA Conditions and for DBAs, Rev. 7
SBC-646, PORV Relief Capacity, Rev. 0
SBC-987, PORV COMS (LTOP) Setpoints for 20 EFPY, Rev. 0

Seabrook EOP Setpoint Study, Rev. 10
YAEC1664, Station Blackout Evaluation, Rev. 4

Completed Surveillance, Maintenance and Modification Testing:

01168495-01, EDG 1A Operability and Pump and Valve Response Time Testing (11/13/09)
01169163-01, 4 kV Switchgear Inspection Testing and Preventive Maintenance (10/27/09)
01169173-01, 4 kV Degraded Voltage Surveillance (10/28/09)
01169266-01, EDE-SWG-5-E360-0801, 4 kV Breaker and Control Circuit Inspection (9/10/09)
01170545 01, RHR Pump 'B' Comprehensive Pump Test (10/12/09)
01172570-01, EDE-SWG-5-E360-0804, 4 kV Breaker Refurbishment (9/4/09)
01172571-01, EDE-SWG-5-E361-0804, 4 kV Breaker Refurbishment (9/16/09)
01173195 01, EDE-US-62-R900-1357 Trip Checks (7/28/09)
01173288-01, Agastat Relay Inspection (8/14/09)
01185933-01, EDE-SWG-5-E361-0801, 4 kV Breaker and Control Circuit Inspection (9/16/09)
01186162-01, 4 kV Undervoltage Surveillance (10/29/09)
01194806-01, 4 kV Degraded Voltage Surveillance (3/19/10)
01196761-01, Bus 5 Node A53 4 kV Loss of Voltage Surveillance (4/19/10)
01196762-01, 4 kV Degraded Voltage Surveillance (4/19/10)
0234849, MOV Diagnostic Testing Summary Report for 1-CBS-V-11 (10/7/03)
0422877, 480 Vac Unit Substation Transformer Inspection (1/9/06)
0500322, 480 Vac Substation Bus 62 Power Factor Testing (1/8/06)
0707863, 480 Vac Unit Sub Inspection (4/13/08)
0712388, Protective Relay PM (7/21/08)
0712392, Unit Substation Relay PM (7/21/08)
0832579, 4 kV Breaker Swapout and Inspection – UAT/SWG 5 (10/27/09)
0832582, 4 kV Breaker Swapout and Inspection – RAT/SWG 5 (10/17/09)
1-CC-L-2272-3-CAL-1, PCCW Head Tank Level Loop Train 'A' Calibration (6/27/08 and 1/21/10)
1-CC-T-2271-CAL-1, PCCW Loop 'B' Supply Header Temp Calibration (4/15/05 and 4/25/08)
1-EDE-B-1-X-BAT3-E32, Battery Service Test (2/26/04, 8/10/05, 7/27/06, 7/25/07, 8/6/08, 1/8/09, and 4/24/10)
1-EDE-B-1-X-BAT4-E33, Battery Discharge Test (4/23/99, 10/4/01, 7/18/02, 7/27/06, 3/21/07, 4/19/07, and 8/23/07)
1-EDE-BC-1-A-BATC-E35, Battery Charger Capacity Test (9/16/09)
1-EDE-I-1-A-E362-0607, 125 Vdc Breaker Inspection (6/5/06)
1-EDE-PP-113-A, DC Breaker Inspection (8/11/09)
1-EDE-SWG-11-A, 125 Vdc Switchgear Inspection (10/16/06)
1-RH-OT005-000, RHR Quarterly Flow and Valve Stroke Test (3/26/08, 6/28/08, 9/9/08, 12/2/08, 2/24/09, 5/26/09, 8/26/09, 11/24/09, and 3/3/10)
CBS-0T017, Train 'A' CBS Valve Stroke Test (1/20/10)
CBS-0T018, CBS Train 'A' 18 Month Valve Position Indication and Status (7/17/09)
DAH-T-5529-CAL-1, T-5529 DG Room Temperature Control Calibration (1/11/10)
DAH-T-5530-CAL-1, T-5530 DG Room Temperature Control Calibration (1/27/10)
ED-OS004, Monthly Surveillance of 13.8 kV and 4160 Vac Breaker Charger (12/26/09)
Operations Surveillance Log Admin-10-0057, Outside Air Temperature (4/28/10)
OS0443.108, FP-P-374, Fire Protection Booster Pump 18 Month Operability Test (10/28/09)
OS1402.03, Train 'A' CCP Oil Cooler Alternate Cooling Supply Flow Test (3/16/09)

OS1402.04, Train 'B' CCP Oil Cooler Alternate Cooling Supply Flow Test (11/8/08)
OX1401.09, Reactor Vent Paths Cold Shutdown Surveillance (11/17/86 and 5/18/87)
PT-16.2, Thermal Barrier Cooling System Preoperational Test (9/11/85)
RC-OT010, RCS Vent Paths Cold Shutdown and 18 Month Surveillance (9/1/09)
RC-P-405-CAL-1, RC Wide Range Pressure Calibration (12/10/08)
RC-T-413-A-CAL-1, Wide Range RCS Hot Leg Calibration (12/3/09)
RSS-OT002, 18 Month Remote Safe Shutdown System Operability Test (10/3/09)
RSS-OT003, 18 Month Remote Safe Shutdown System Operability Test (10/5/09)
SI-OT005, Safety Injection Test – Group B Pump Test (12/23/09)
SI-OT007, Safety Injection Valve Position Indication Verification (12/23/08)

Corrective Action Documents:

196431*	391249*	011511	115242	202348
196515*	391320*	011817	116093	205702
222070*	000154	012617	132462	205706
222071*	000234	012884	139098	206418
222226*	000762	013282	142250	206450
222228*	001203	014215	142452	208483
222230*	002011	015710	146042	209470
222235*	002637	015863	150039	209576
222236*	002804	016278	152068	209860
222237*	003569	017322	154918	210424
222239*	003638	017975	155125	211950
222245*	004391	018212	155563	213351
222271*	004786	019928	155815	214863
222372*	004974	021693	157637	215561
222637*	005438	032680	160548	216429
222930*	005520	042960	162662	219518
222938*	005697	066834	173930	219521
222952*	006156	076838	176865	219759
223023*	006330	077102	177704	219994
223037*	006556	079382	181538	220051
223213*	007324	080929	192356	221158
223224*	007325	081782	192766	221515
223224*	007653	081911	193524	221937
391073*	009539	085089	195935	222857
391104*	010952	087602	201399	
391237*	011219	090958	202081	
391242*	011492	099859	202202	

* Document written as a result of inspection effort.

Drawings:

1NHY-250000, Data Sheets for Motor and Air Operated Valves and Dampers, Rev. 72
1NHY-300219, Service Environment Chart, Sh. 1, Rev. 27
1NHY-300219, Service Environment Chart, Sh. 2, Rev. 22
1NHY-301104, SW and CW Pmp House 460 Vac MCC 1/2-E514 One Line Diagram, Rev. 15
1NHY-301105, SW and CW Pmp House 460 Vac MCC 1/2-E614 One Line Diagram, Rev. 10
1NHY-301704, 480 Vac Unit Substation Bus 1-E64 & 2-E64 One Line Diagram, Rev. 11
1NHY-301705, Cooling Tower Swgr. Room 460V MCC 1/2-E513 One Line Diagram, Rev. 17
1NHY-301706, Cooling Tower Swgr. Room 460V MCC 1/2-E641 One Line Diagram, Rev. 11
1NHY-310002, Unit Electrical Distribution One Line Diagram, Rev. 41
1NHY-310007, 4160 Vac Switchgear Bus 1-E5 One Line Diagram, Rev. 20
1NHY-310008, 4160 Vac Switchgear Bus 1-E6 One Line Diagram, Rev. 18
1NHY-310010, Diesel Generator DG-1A and DG-1B One Line Diagram, Sh. 1, Rev. 15
1NHY-310010, Diesel Generator DG-1A and DG-1B One Line Diagram, Sh. 2, Rev. 4
1NHY-310014, 480 Vac Unit Substation Buses 1-E61 and 1-E62, One Line Diagram, Rev. 18
1NHY-310023, EDG Building 460 Vac MCC 1-E511 One Line Diagram, Rev. 21
1NHY-310024, Control Building 460 Vac MCC 1-E512 One Line Diagram, Sh. 1, Rev. 28
1NHY-310026, Control Building 460 Vac MCC 1-E531 One Line Diagram, Rev. 31
1NHY-310027, Control Building 460 Vac MCC 1-E521 One Line Diagram, Rev. 28
1NHY-310028, Control Building 460 Vac MCC 1-E522 and 1-E622 One Line Diagram, Rev. 13
1NHY-310029, EDG Building 460 Vac MCC 1-E611 One Line Diagram, Rev. 17
1NHY-310030, Control Building 460 Vac MCC 1-E612 One Line Diagram, Sh. 1, Rev. 31
1NHY-310032, Control Building 460 Vac MCC 1-E621 One Line Diagram, Sh. 1, Rev. 28
1NHY-310032, Control Building 460 Vac MCC 1-E631 One Line Diagram, Sh. 1, Rev. 28
1NHY-310041, 125 Vdc and 120 Vac Instrument Buses One Line, Rev. 16
1NHY-310042, 125 Vdc Vital Distribution One Line Diagram, Sh. 1, Rev. 16
1NHY-310042, 125 Vdc Vital Distribution One Line Diagram, Sh. 2, Rev. 3
1NHY-310051, 480 Vac Unit Substation Bus 1-E53 One Line Diagram, Rev. 15
1NHY-310052, 480 Vac Unit Substation Bus 1-E63 One Line Diagram, Rev. 15
1NHY-310057, Control Building 460 Vac MCC 1-E512 One Line Diagram, Sh. 2, Rev. 15
1NHY-310058, Control Building 460 Vac MCC 1-E612 One Line Diagram, Sh. 2, Rev. 13
1NHY-310066, Control Building 460 Vac MCC 1-E515 One Line Diagram, Rev. 9
1NHY-310067, Control Building 460 Vac MCC 1-E615 One Line Diagram, Rev. 9
1NHY-310102, EDG 1-DG-1A Ground Protection Schematic Diagram, Sh. HA1b, Rev. 9
1NHY-310102, EDG 1-DG-1A Three Line Diagram, Sh. HA1a, Rev. 11
1NHY-310102, Transformer 1-EDE-X-5D Close Schematic, Sh. A83b, Rev. 7
1NHY-310102, Transformer 1-EDE-X-5D Protection Schematic, Sh. A83d, Rev. 2
1NHY-310102, Transformer 1-EDE-X-5D Trip Schematic, Sh. A83c, Rev. 3
1NHY-310102, 4160 Vac Bus 1-E5 DG1A Close Schematic, Sh. A54c, Rev. 13
1NHY-310102, 4160 Vac Bus 1-E5 DG1A Protection Schematic, Sh. A54e, Rev. 11
1NHY-310102, 4160 Vac Bus 1-E5 DG1A Protection Schematic, Sh. A54f, Rev. 7
1NHY-310102, 4160 Vac Bus 1-E5 DG1A Three Line Diagram, Sh. A54a, Rev. 14
1NHY-310102, 4160 Vac Bus 1-E5 DG1A Three Line Diagram, Sh. A54b, Rev. 9
1NHY-310102, 4160 Vac Bus 1-E5 DG1A Trip Schematic, Sh. A54d, Rev. 8
1NHY-310102, 4160 Vac Bus 1-E5 PTs Three Line Diagram, Sh. A53a, Rev. 13
1NHY-310102, 4160 Vac Bus 1-E5 RAT Close Schematic, Sh. A52b, Rev. 16
1NHY-310102, 4160 Vac Bus 1-E5 RAT Protection Schematic, Sh. A52e, Rev. 6

1NHY-310102, 4160 Vac Bus 1-E5 RAT Three Line Diagram, Sh. A52a, Rev. 5
1NHY-310102, 4160 Vac Bus 1-E5 RAT Trip Schematic, Sh. A52c, Rev. 7
1NHY-310102, 4160 Vac Bus 1-E5 RAT Trip Schematic, Sh. A52d, Rev. 5
1NHY-310102, 4160 Vac Bus 1-E5 UAT Close Schematic, Sh. A51b, Rev. 13
1NHY-310102, 4160 Vac Bus 1-E5 UAT Protection Schematic, Sh. A51e, Rev. 2
1NHY-310102, 4160 Vac Bus 1-E5 UAT Three Line Diagram, Sh. A51a, Rev. 6
1NHY-310102, 4160 Vac Bus 1-E5 UAT Trip Schematic, Sh. A51c, Rev. 8
1NHY-310102, 4160 Vac Bus 1-E5 UAT Trip Schematic, Sh. A51d, Rev. 4
1NHY-310102, 4160 Vac Feed to 1-EDE-X-5D Three Line Diagram, Sh. A83a, Rev. 5
1NHY-310107, 125 Vdc Bus Distribution Panel PP-112A Schedule, Sh. E87a, Rev. 16
1NHY-310107, 125 Vdc Panel 1-PP-111A, Sh. 90, Rev. 12
1NHY-310107, 125 Vdc Panel 1-PP-112A, Sh. 78, Rev. 16
1NHY-310107, 125 Vdc Panel 1-PP-113A, Sh. 74, Rev. 10
1NHY-310107, 125 Vdc Switchboard 1-SWG-11A, Sh. 46, Rev. 5
1NHY-310844, Emergency Feed Water Pump 1-P-37B Close Schematic, Sh. A80b, Rev. 10
1NHY-310844, Emergency Feed Water Pump 1-P-37B Protection Schematic, Sh. A80d, Rev. 2
1NHY-310844, Emergency Feed Water Pump 1-P-37B Three Line Diagram, Sh. A80a, Rev. 4
1NHY-310844, Emergency Feed Water Pump 1-P-37B Trip Schematic, Sh. A80c, Rev. 4
1NHY-310857, EDG 1-DG-1A Start Circuit No. 1 Schematic Diagram, Sh. E93/8a, Rev. 9
1NHY-310857, EDG 1-DG-1A Start Circuit No. 1 Schematic Diagram, Sh. E93/8b, Rev. 7
1NHY-310857, EDG 1-DG-1A Start Circuit No. 2 Schematic Diagram, Sh. E93/8c, Rev. 9
1NHY-310857, EDG 1-DG-1A Stop Circuit Schematic Diagram, Sh. E93/8d, Rev. 8
1NHY-310882, Pressurizer Pressure Control Valve Control Wiring Diagram, Sh. 19e, Rev. 0
1NHY-310882, Pressurizer Pressure Control Valve Control Wiring Diagram, Sh. 19f, Rev. 0
1NHY-310882, Pressurizer Pressure Control Valve Schematic, Sh. 19a, Rev. 6
1NHY-310882, Pressurizer Pressure Control Valve Schematic, Sh. 19b, Rev. 8
1NHY-310895, Thermal Barrier Pump Control Wiring Diagram, Sh. B4Me, Rev. 1
1NHY-310895, Thermal Barrier Pump Schematic, Sh. B4Ma, Rev. 2
1NHY-310900, Containment Spray Valve V11 Cable Schematic, Sh. 63, Rev. 7
1NHY-310900, Containment Spray Valve V11 Control Wiring, Sh. 63a, Rev. 0
1NHY-310900, Containment Spray Valve V11 Schematic, Sh. 61, Rev. 7
1NHY-310900, Containment Spray Valve V11 Switch Development, Sh. 62, Rev. 6
1NHY-310928, EDG Room Return Air Damper Control Wiring Diagram, Sh. 6e, Rev. 0
1NHY-310928, EDG Room Return Air Damper Schematic, Sh. 6a, Rev. 0
1NHY-310928, EDG Room Return Air Damper Schematic, Sh. 6b, Rev. 0
1NHY-310928, EDG Room Return Air Damper Schematic, Sh. 6c, Rev. 0
1NHY-310928, EDG Room Return Air Damper Schematic, Sh. 6d, Rev. 1
1NHY-310928, EDG Room Return Fan Schematic, Sh. 3a, Rev. 10
1NHY-310928, EDG Room Return Fan Schematic, Sh. 3b, Rev. 9
1NHY-310928, EDG Room Return Fan Schematic, Sh. 3c, Rev. 9
1NHY-310928, EDG Room Return Fan Schematic, Sh. 3d, Rev. 9
1NHY-310928, EDG Room Return Fan Schematic, Sh. 3e, Rev. 2
1NHY-310928, EDG Room Supply Fan Schematic, Sh. 1a, Rev. 15
1NHY-310928, EDG Room Supply Fan Schematic, Sh. 1b, Rev. 11
1NHY-310928, EDG Room Supply Fan Schematic, Sh. 1c, Rev. 11
1NHY-310928, EDG Room Supply Fan Schematic, Sh. 1d, Rev. 11
1NHY-310928, EDG Room Supply Fan Schematic, Sh. 1e, Rev. 4
1NHY-503274, CC – Thermal Barrier Isolation Valves Logic Diagram, Rev. 7

1NHY-503274, Thermal Barrier Isolation Valve Logic Diagram Loop, Rev. 7
1NHY-503278, PCCW Head Tank Level Isolation Signal Logic Diagram, Rev. 9
1NHY-503284, CC – RC Pump Thermal Barrier Coolant Pumps Logic Diagram, Rev. 4
1NHY-503284, Thermal Barrier Pump Logic Diagram Loop, Rev. 4
1NHY-503763, RH – Recirculation Valves Logic Diagram, Rev. 10
1NHY-503901, RH – Pumps P-6A & B Disch MOV to RWST Train 'B' Logic Diagram, Rev. 8
1NHY-503911, SI – Motor Operated Valves Logic Diagram, Rev. 6
1NHY-506170, CBS – Control Loop Diagram, Rev. 17
1NHY-506191, PCCW Head Tank Control Loop Diagram, Rev. 15
1NHY-506202, CC – RC Pump Thermal Barrier Coolant System Control Loop Diagram, Rev. 9
1NHY-506202, Thermal Barrier Control Loop, Rev. 9
1NHY-506203, CC – RC Pump Thermal Barrier Control Loop Diagram, Rev. 5
1NHY-506203, Thermal Barrier Control Loop, Rev. 5
1NHY-506381, DAH Control Loop Diagram, Rev. 12
1NHY-506798, SI – Pumps 6A & B Control Loop Diagram, Rev. 15
1NHY-509037, Process Control Block Diagram, Rev. 8
1NHY-509038, RCS Cold Over-Pressurization Control, Rev. 5
CBS-B20233, Containment Spray System, Rev. 35
CBS-D20233, Containment Spray System, Rev. 32
CC-B20204, Component Cooling Loop 'A' Overview, Rev. 4
CC-B20209, Primary Component Cooling Thermal Barrier Loop, Rev. 10
CC-B20211, Component Cooling Loop 'B' Detail, Rev. 21
DAH-B20624, Diesel Generator Building Air Handling, Rev. 12
DS-C-67970, Solenoid Power Operated Relief Valve, Rev. D
FP 20684, EDG Voltage Dip Curves, Rev. 0
ILD-1-CC-L02272, Instrument Loop PCCW Loop 'B' Head Tank TK-198 Level, Rev. 0
ILD-1-CC-T02271, Instrument Loop PCCW Loop 'B' Supply Header Temperature, Rev. 1
LBC-1-CBS-040225, Containment Building Spray System – VAS Logic by Computer, Rev. 2
PID-1-CC-D20204, Primary Component Cooling Loop 'A' Overview, Rev. 4
PID-1-CC-D20205, Primary Component Cooling Loop 'A' Detail, Rev. 25
PID-1-CC-D20206, Primary Component Cooling Loop 'A' Detail, Rev. 16
PID-1-CC-D20207, Primary Component Cooling Loop 'A' Detail, Rev. 12
PID-1-CC-D20208, Primary Component Cooling Loop 'A' Detail, Rev. 3
PID-1-CC-D20209, Primary Component Cooling Thermal Barrier Loop Detail, Rev. 9
PID-1-CC-D20210, Primary Component Cooling Loop 'B' Overview, Rev. 3
PID-1-CC-D20212, Primary Component Cooling Loop 'B' Detail, Rev. 13
PID-1-CC-D20213, Primary Component Cooling Loop 'B' Detail, Rev. 14
PID-1-CC-D20214, Primary Component Cooling Loop 'B' Detail, Rev. 7
PID-1-CC-D20215, Primary Component Cooling Loop 'B' Detail, Rev. 6
PID-1-DF-B20196, Plant Drainage Oil/Water Separation System 2 Detail, Rev. 7
PID-1-SI-B20445, Safety Injection System Overview, Rev. 4
PID-1-SI-B20446, Safety Injection System Intermediate Head Injection System Detail, Rev. 17
PID-1-SI-B20447, Safety Injection System High Head Injection System Detail, Rev. 15
PID-1-SI-B20449, Safety Injection System Intermediate Head Injection System, Sh. 2, Rev. 17
PID-1-SI-B20450, Safety Injection System Low Head Injection System Detail, Rev. 14
PID-1-SW-B20792, Service Water System Nuclear Overview, Rev. 6
PID-1-SW-B20794, Service Water System Nuclear Detail, Rev. 33
PID-1-SW-B20795, Service Water System Nuclear Detail, Rev. 37

PID-1-SW-B20796, Service Water System Nuclear Detail, Rev. 5
PID-1-WLD-B20221, Waste Processing Liquid Drains–RHR Equipment Vaults 1 and 2, Rev. 10
SI-B20446, Safety Injection Intermediate Head System, Rev. 18
SI-B20448, Safety Injection Low Head System, Rev. 19
SI-B20449, Safety Injection Low Head System, Rev. 16
SK-EC145164-2000, Service Water Strainer Basket Modification – 1SW-S-10, Rev. 0

Design Basis Documents:

DBD-CC-01, Primary Component Cooling Water System, Rev. 4
DBD-DAH-01, Diesel Generator Building Heating and Ventilation Systems, Rev. 2
DBD-DG-01, Emergency Diesel Generator – Mechanical, Rev. 4
DBD-ED-06, 480 Vac System, Rev. 0
DBD-EFW-01, Emergency Feed Water System, Rev. 5
DBD-EQ-01, Environmental Qualification of Electrical Equipment, Rev. 2
DBD-RC-01, Reactor Coolant System, Rev. 2
DBD-RH-01, Residual Heat Removal System, Rev. 1
DBD-SW-01, Service Water System, Rev. 5
Seabrook Station Updated Final Safety Analysis Report, Rev. 13
TR 24-3.7.10, Area Temperature Monitoring, Rev. 115

Miscellaneous:

10CAR018, Modify SW Strainers to Prevent Debris from Impacting the PCCW HXs, 3/9/10
CEM#98-060, Review of TS/Procedures and Response to NRC GL 96-01, 2/19/98
E-0 Background, Reactor Trip or Safety Injection, HP-Rev. 2
ECA-0.0 Background, Loss of all AC Power, HP-Rev. 2
Engineering Department Standard 36180, Structural Monitoring Program, Rev. 0
FPL Letter L-2006-073 to NRC, NRC Generic Letter 2006-02 60-Day Response, 4/3/06
FPLE Letter NYN-03069, License Amendment Request 03-01, Electrical Power, 8/25/03
FPLE Letter NYN-04005, Response to RAI, License Amendment Request 03-01, 7/29/04
FR-H.1 Background, Response to Loss of Secondary Heat Sink, HP-Rev. 2
NS 95-006, Turbine-Driven Pump Overspeed Due to Governor Valve Stem Binding, 12/13/95
NUREG-0896, SER - Operation of the Seabrook Station Units 1 and 2, March 1983
Operating Training Detailed Systems Text, Component Cooling System, Rev. 9
Operator Logs, Unit Journal Day & Mid Shift, 10/1/09 to 11/6/09, 3/15/08 to 4/1/08
PSNH Letter SBN-427, Open Items Responses, 1/20/83
PSNH Letter SBN-899, Seabrook Station Voltage Regulation Study, 11/21/85
PSNH Letter SBN-903, Resolution of Power Systems Branch Confirmatory Items, 11/27/85
SB-20885, PSNH Letter – Request for Additional Information, 5/13/86
SBU-74794, UE Letter – RCP Thermal Barrier Leakage, 6/27/83
SBU-97831, UE Letter – RCP Seal - Thermal Barrier, 2/19/86
Simulator Examination, Demonstrative Examination #50, Rev. 4
Standard Operating Order 10-007, Bus E5 Degraded Voltage Relay Reset Condition, 5/3/10
Standing Operating Order 10-009, Technical Requirement 24 Interim Guidance, 5/12/10
System Health Report, 125 Vdc System, Fourth Quarter 2009
System Health Report, Diesel Generator Building Air Handling, Fourth Quarter 2009

System Health Report, Reactor Coolant System, Fourth Quarter 2009
WCAP-16755-NP, Operator Time Critical Action Program Standard, Rev. 0

Modifications & 10 CFR 50.59 Reviews:

DCN 65-0272B, Provide Safety Grade Manual Controls for ASDVs, 9/19/88
DCR 02-008, Reduced ECCS Pump Head Requirements, 5/6/02
EC-145024, Reconciliation of Methods - RWST Minimum Submergence Levels, Rev. 0
ECA 81-107178, Thermal Barrier Cooling System Flow Rate increase, Rev. A
ECA 98/118030, PCCW TCVs, Rev. C
ECA 99-110282, Thermal Barrier Cooling Loop Redesign, Rev. A
MMOD 95-0632, AB/AS/ASC Setpoints and AB Time Delay Addition, 5/20/98
MMOD 95-574, EFW Terry Turbine - Packing Material Changes, Governor Valve Stem, Rev. 0

Procedures:

E-0, Reactor Trip or Safety Injection, Rev. 48
E-1, Loss of Reactor or Secondary Coolant, Rev. 38
E-3, Steam Generator Tube Rupture, Rev. 40
ECA-0.0, Loss of All AC Power, Rev. 40
ECA-2.1, Uncontrolled Depressurization of All Steam Generators, Rev. 33
ES-0.1, Reactor Trip Response, Rev. 35
ES-1.3, Transfer to Cold Leg Recirculation, Rev. 26
ES1850.001, Check Valve Performance Monitoring Program, Rev. 5
F-0.3, Heat Sink (H), Rev. 20
FR-H.1, Response to Loss of Secondary Heat Sink, Rev. 33
FR-S.1, Response to Nuclear Power Generation/ATWS, Rev. 29
IS1616.22X, CC-L-2292-X PCCW Head Tank Level Calibration, Rev. 5
IS1616.411, CC-T-2271 PCCW Supply Header Temperature Control Calibration, Rev. 10
IS603.095, PORV Switch/Solenoid Repair and Gasket Replacement, Rev. 3
JPM L0050J, Locally Operate the EFW Flow Control Valves, Rev. 6
JPM L0113J, FR-H.1 Bleed and Feed, Rev. 5
JPM L0134J, Align Alternate (Firewater) Cooling to CCP Lube Oil Cooler, Rev. 3
JPM L0135J, Align Alternate (Demin Water) Cooling to CCP Lube Oil Cooler, Rev. 3
JPM L0174J, ASDV Local Operation (RCA), Rev. 0
LS0557.18, ABB K-1600 Breaker Refurbishment, Rev. 3
LS0558.03, 4 kV Motor - Routine Testing Inspection and Preventive Maintenance, Rev. 1
LS0563.09, Diesel Generator Underfrequency Relay Preventive Maintenance, Rev. 1
LS0569.01, Inspection and Testing of Limitorque Valve Actuators, Rev. 5
LS0569.07, Corrective Maintenance of Limitorque Actuators, Types SB-0, SB-1, SB-3, Rev. 3
LS0569.13, Miscellaneous MOV Procedures and Guidelines, Rev. 2
LS0569.16, Diagnostic Testing of Rising Stem MOVs, Rev. 6
LS0569.17, Diagnostic Testing of Rising Stem MOVs Using the Torque Thrust Cell, Rev. 3
LS0569.20, Lubrication PM and Starter Inspection for MOV Actuators, Rev. 14
LX0556.04, Station Battery Service Test, Rev. 3
LX0556.05, Station Battery Performance Discharge Test, Rev. 2
LX0563.07, 4 kV Bus Degraded Voltage Protection Monthly Surveillance, Rev. 3
MA 4.8, Control of Scaffolding, Rev. 9

MA 7.4, Breaker Refurbishment, Rev. 1
MCR Alarm Response D4266, Thermal Barrier Cooling Flow High, Rev. 3
MCR Alarm Response D7079, DG Room Temp High, Rev. 2
MS0519.14, Crosby Power Operated Relief Valve Maintenance, Rev. 4
MS0599.47, Erection of Scaffolding, Rev. 0
NAP-412, Operational Decision Making, Rev. 8
OP 9.2, Transient Response Procedure User's Guide, Rev. 14
OS0443.108, FP-P-374, Fire Protection Booster Pump 18 Month Operability Test, Rev. 1
OS-05-01-02, SEPS Demonstration Test on Bus E6, Rev. 0
OS1000.04, Plant Cooldown From Hot Standby to Cold Shutdown, Rev. 21
OS1001.15, PORV Block Valve Operation, Rev. 0
OS1002.02, Operation of Letdown, Charging and Seal Injection, Rev. 22
OS1012.02, Thermal Barrier Cooling Water System Fill and Vent, Rev. 8
OS1012.08, Thermal Barrier Cooling Water System Operation, Rev. 10
OS1023.54, Diesel Generator Building Ventilation System Operation, Rev. 8
OS1023.74, Maintenance of Safety Related HVAC Systems - Compensatory Ventilation, Rev. 5
OS1212.01, PCCW System Malfunction, Rev. 13
OS1246.02, Degraded Vital AC Power (Plant Operating), Rev. 9
OS1402.03, Train 'A' CCP Oil Cooler Alternate Cooling Supply Flow Test, Rev. 1
OS1402.04, Train 'B' CCP Oil Cooler Alternate Cooling Supply Flow Test, Rev. 1
OX1400.02, Remote Safe Shutdown System 18 Month Operability Check, Rev. 6
OX1400.09, Reactor Vent Paths Cold Shutdown and 18 Month Surveillance, Rev. 7
OX1426.20, EDG 18 Month Operability Surveillance Test, Rev. 12
OX1456.81, Operability Testing of IST Valves, Rev. 10
OX1456.86, Operability Testing of IST Pumps, Rev. 2
OX1461.01, SEPS Full Load Testing Surveillance, Rev. 2
OX1461.04, SEPS Monthly Availability Surveillance, Rev. 2
OX1461.05, SEPS Annual Availability Surveillance, Rev. 1
PI-AA-205, Condition Evaluation and Corrective Action, Rev. 8
PR 7.1, Emergency Operating Procedure Writer's Guide, Rev. 0
SM 7.20, Control of Time Critical Actions, Rev. 3

<u>Vendor Manuals & Specifications</u>:

3379001-11, Service Bulletin - Fuels for Cummins Engines, 3/20/07
41-348.11C, ABB Type SA-1 Generator Differential Relay, Rev. C
82-4101-A02, Thermal Barrier Heat Exchanger Specifications, Rev. 2
BOP FR-1, Functional Requirements and Design Criteria – CCW System, Rev. 0
FP-56024, Thermal Barrier Heat Exchanger, Rev. 1
FP-57748, Garrett PORV Solenoid Pilot Valve EQ Data Package, Rev. 1
IM 33-51963, 480 Vac Unit Substation Instruction Manual, Rev. 4
J12.01.99, Industrial Batteries and Chargers, 3/10/00
Specification 238-34, Thermal Barrier Cooling Pumps, Rev. 5
VMAN FP-56336, Thermal Barrier Water Pumps, Rev. 0
VMAN FP-57526, PORV Solenoid/ Power Operated Relief Valves, Rev. 1

Work Orders:

0119009	0821150	1170342
0121562	0821151	1189035
0301479	0821955	1191476
0636176	1168036	

LIST OF ACRONYMS

AC	Alternating Current
AR	Action Report
CBS	Containment Building Spray
CFR	Code of Federal Regulations
CR	Condition Report
DAH	Diesel Air Handling
DC	Direct Current
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EFW	Emergency Feedwater
EOP	Emergency Operating Procedure
FIN	Finding
IEEE	Institute of Electrical and Electronics Engineers
IMC	Inspection Manual Chapter
IN	Information Notice
IP	Inspection Procedure
kV	kilo-Volts
kW	kilo-Watts
MOV	Motor Operated Valve
NCV	Non-cited Violation
NRC	Nuclear Regulatory Commission
PCCW	Primary Component Cooling Water
PORV	Power-Operated Relief Valve
PSS	Probabilistic Safety Study
RAW	Risk Achievement Worth
RCP	Reactor Coolant Pump
RCPTBC	Reactor Coolant Pump Thermal Barrier Cooling
RHR	Residual Heat Removal
RRW	Risk Reduction Worth
RWST	Refueling Water Storage Tank
SBO	Station Blackout
SDP	Significance Determination Process
SEPS	Supplemental Emergency Power System
SI	Safety Injection
SPAR	Standardized Plant Analysis Risk
SW	Service Water
UFSAR	Updated Final Safety Analysis Report
Vac	Volts, Alternating Current
Vdc	Volts, Direct Current