UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE RD, STE 102
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May 13, 2024

Bob Coffey
Executive Vice President, Nuclear Division
and Chief Nuclear Officer
Florida Power & Light Company
700 Universe Blvd.
Mail Stop: EX/JB
Juno Beach, FL 33408

SUBJECT: SEABROOK STATION – INTEGRATED INSPECTION REPORT 05000443/2024001

Dear Bob Coffey:

On March 31, 2024, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Seabrook Station. On May 8, 2024, the NRC inspectors discussed the results of this inspection with David Sluszka, Site Vice President, and other members of your staff. The results of this inspection are documented in the enclosed report.

Two findings of very low safety significance (Green) are documented in this report. Two of these findings involved violations of NRC requirements. We are treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2 of the Enforcement Policy.

If you contest the violations or the significance or severity of the violations documented in this inspection 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, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement; and the NRC Resident Inspector at Seabrook Station.

If you disagree with a cross-cutting aspect assignment 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 U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; and the NRC Resident Inspector at Seabrook Station.

This letter, its enclosure, and your response (if any) will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and at the NRC Public Document Room in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 2.390, "Public Inspections, Exemptions, Requests for Withholding."

Sincerely,

Matthew R. Young

Digitally signed by Matthew R. Young
Date: 2024.05.13 09:16:50 -04'00'

Matt R. Young, Chief
Projects Branch 2
Division of Operating Reactor Safety

Docket No. 05000443
License No. NPF-86

Enclosure:
As stated

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SUBJECT: SEABROOK STATION – INTEGRATED INSPECTION REPORT 05000443/2024001 DATED MAY 13, 2024

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U.S. NUCLEAR REGULATORY COMMISSION
Inspection Report

Docket Number:	05000443
License Number:	NPF-86
Report Number:	05000443/2024001
Enterprise Identifier:	I-2024-001-0052
Licensee:	NextEra Energy Seabrook, LLC
Facility:	Seabrook Station
Location:	Seabrook, New Hampshire
Inspection Dates:	January 1, 2024 to March 31, 2024
Inspectors:	T. Daun, Senior Resident Inspector E. Allen, Resident Inspector P. Cataldo, Senior Reactor Inspector S. Elkhiamy, Senior Project Engineer N. Floyd, Senior Reactor Inspector G. Thomas, Senior Civil Engineer D. Werkheiser, Senior Reactor Analyst
Approved By:	Matt R. Young, Chief Projects Branch 2 Division of Operating Reactor Safety

Enclosure

SUMMARY

The U.S. NRC continued monitoring the licensee's performance by conducting an integrated inspection at Seabrook Station, in accordance with the Reactor Oversight Process. The Reactor Oversight Process is the NRC's program for overseeing the safe operation of commercial nuclear power reactors. Refer to https://www.nrc.gov/reactors/operating/oversight.html for more information.

List of Findings and Violations

Failure to Promptly Identify and Correct a Condition Adverse to Quality for Gas Accumulation in Containment Building Spray Piping			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000443/2024001-01 Open/Closed	[P.3] - Resolution	71152A
The NRC inspectors identified a Green finding and associated non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," when the licensee did not identify and correct a condition adverse to quality. Specifically, from April 2023 to September 2023, the licensee did not sufficiently characterize a gas void in a section of 'B' trains residual heat removal (RHR) and containment building spray (CBS) suction piping, and did not ensure, as a corrective action, that the pipe remained sufficiently full to the next scheduled technical specification monthly surveillance.			

Failure to Adequately Evaluate the Reactor Pit Slab During Reanalysis of the Containment Internal Structures to Incorporate Alkali-Silica Reaction Loads			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Barrier Integrity	Green NCV 05000443/2024001-02 Open/Closed	[P.2] - Evaluation	71152A
The NRC inspectors identified a Green finding and associated non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," because the licensee did not verify the adequacy of the reactor pit slab at elevation (-) 44ft-9in for applicable design loads in the revised structural evaluation for the containment internal structures (CIS).			

Additional Tracking Items

Type	Issue Number	Title	Report Section	Status
LER	05000443/2023-004-00	LER 2023-04-00 for Seabrook Station, Manual Reactor Trip due to Main Turbine Electro-Hydraulic Control Oil Level	71153	Closed

PLANT STATUS

Seabrook Station began the inspection period operating at 100 percent rated thermal power and remained at or near full power for the inspection period.

INSPECTION SCOPES

Inspections were conducted using the appropriate portions of the inspection procedures (IPs) in effect at the beginning of the inspection unless otherwise noted. Currently approved IPs with their attached revision histories are located on the public website at http://www.nrc.gov/reading-rm/doc-collections/insp-manual/inspection-procedure/index.html. Samples were declared complete when the IP requirements most appropriate to the inspection activity were met consistent with Inspection Manual Chapter (IMC) 2515, “Light-Water Reactor Inspection Program - Operations Phase.” The inspectors performed activities described in IMC 2515, Appendix D, “Plant Status,” observed risk significant activities, and completed onsite portions of IPs. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel to assess licensee performance and compliance with Commission rules and regulations, license conditions, site procedures, and standards.

REACTOR SAFETY

71111.04 - Equipment Alignment

Partial Walkdown (IP Section 03.01) (6 Samples)

The inspectors evaluated system configurations during partial walkdowns of the following systems/trains:

(1) 'B' residual heat removal system during 'A' residual heat removal system maintenance on January 16, 2024
(2) 'B' service water system during maintenance on the 'A' service water system on January 18, 2024
(3) 'A' containment enclosure ventilation system during the containment enclosure fan (EAH-FN-5B) maintenance on January 24, 2024
(4) Emergency feedwater steam supply following quarterly turbine driven emergency feedwater pump surveillance on February 5, 2024
(5) 'B' 125-volt direct current system during 'B' vital battery maintenance on March 6, 2024
(6) 4160-volt electrical distribution system during 'B' reserve auxiliary transformer maintenance on March 12, 2024

71111.05 - Fire Protection

Fire Area Walkdown and Inspection (IP Section 03.01) (5 Samples)

The inspectors evaluated the implementation of the fire protection program by conducting a walkdown and performing a review to verify program compliance, equipment functionality, material condition, and operational readiness of the following fire areas:

(1) Containment enclosure ventilation area (CE-F-1A-Z) on January 22, 2024
(2) 'A' motor control center room (SW-F-1B-A) on January 26, 2024
(3) 345-kilovolt switchyard enclosure (PLT-F-4-0) on March 7, 2024

(4) Fire area transformer yard (PLT-F-3-0) on March 13, 2024
(5) Primary component cooling water pump area (PAB-F-2C-Z) on March 28, 2024

Fire Brigade Drill Performance (IP Section 03.02) (1 Sample)

(1) The inspectors evaluated the onsite fire brigade training and performance during an unannounced fire drill on February 15, 2024

71111.11Q - Licensed Operator Requalification Program and Licensed Operator Performance

Licensed Operator Performance in the Actual Plant/Main Control Room (IP Section 03.01) (1 Sample)

(1) The inspectors observed and evaluated licensed operator performance during turbine driven emergency feedwater pump surveillance testing, reactivity manipulations, crew turn over, and crew briefing activities in the control room on January 23, 2024

Licensed Operator Requalification Training/Examinations (IP Section 03.02) (1 Sample)

(1) The inspectors observed and evaluated licensed operator training conducted in the plant reference simulator on February 20, 2024

71111.12 - Maintenance Effectiveness

Maintenance Effectiveness (IP Section 03.01) (1 Sample)

The inspectors evaluated the effectiveness of maintenance to ensure the following structures, systems, and components remain capable of performing their intended function:

(1) Power supplies for 7300 protection, control, and balance of plant cabinets on March 26, 2024

71111.13 - Maintenance Risk Assessments and Emergent Work Control

Risk Assessment and Management (IP Section 03.01) (5 Samples)

The inspectors evaluated the accuracy and completeness of risk assessments for the following planned and emergent work activities to ensure configuration changes and appropriate work controls were addressed:

(1) Elevated risk during 345-kilovolt line outage on January 31, 2024 and February 1, 2024
(2) Yellow risk during planned maintenance on 4160-kilovolt bus (EDE-SWG-5) and containment enclosure cooling fan (EAH-FN-5A) on February 14, 2024
(3) Emergent work controls during the failure of 'A' main steam isolation valve hydraulic system on February 20, 2024
(4) Risk management actions associated with startup feed pump, trip throttle valve, and switchyard planned testing on February 22, 2024
(5) Emergent work controls during the failure of the 3B reserve auxiliary transformer on March 7, 2024

71111.15 - Operability Determinations and Functionality Assessments

Operability Determination or Functionality Assessment (IP Section 03.01) (3 Samples)

The inspectors evaluated the licensee's justifications and actions associated with the following operability determinations and functionality assessments:

(1) 'A' main steam isolation valve with hydraulic power isolated on February 20, 2024
(2) 'B' diesel generator air start receiver check valve leakage on February 27, 2024
(3) Containment internal structures root cause analyses and Stage 2 structural evaluations with alkali-silica reaction expansion loads on March 28, 2024

71111.24 - Testing and Maintenance of Equipment Important to Risk

The inspectors evaluated the following testing and maintenance activities to verify system operability and/or functionality:

Post Maintenance Testing (IP Section 03.01) (6 Samples)

(1) Service water cooling tower group 'A' pump test following replacement of the pump motor breaker on January 18, 2024
(2) 'A' emergency diesel generator jacket coolant standby circulation pump replacement on January 25, 2024
(3) 'B' coolant charging pump following speed increaser and pump coupling inspection on January 29, 2024
(4) Service water vacuum breaker test following replacement of SW-V-175 on February 9, 2024
(5) 'A' main steam isolation valve hydraulic system following emergent repairs on February 20, 2024
(6) Reserve auxiliary transformer 3B restoration on March 26, 2024

Surveillance Testing (IP Section 03.01) (3 Samples)

(1) 'A' train 4.16-kilovolt loss of voltage protection quarterly surveillance on February 14, 2024
(2) Main steam drain valve operability test on February 21, 2024
(3) Electric emergency feedwater pump quarterly pump test on February 22, 2024

In-service Testing (IP Section 03.01) (1 Sample)

(1) Turbine driven emergency feedwater pump quarterly surveillance on January 23, 2024

71114.06 - Drill Evaluation

Required Emergency Preparedness Drill (IP Section 03.01) (1 Sample)

(1) The inspectors evaluated the conduct of a routine, full participation emergency planning drill on February 7, 2024

OTHER ACTIVITIES – BASELINE

71151 - Performance Indicator Verification

The inspectors verified licensee performance indicators submittals listed below:

IE01: Unplanned Scrams per 7000 Critical Hours (IP Section 02.01) (1 Sample)

(1) For the period January 1, 2023 through December 31, 2023

IE03: Unplanned Power Changes per 7000 Critical Hours (IP Section 02.02) (1 Sample)

(1) For the period January 1, 2023 through December 31, 2023

IE04: Unplanned Scrams with Complications (IP Section 02.03) (1 Sample)

(1) For the period January 1, 2023 through December 31, 2023

71152A - Annual Follow-up Problem Identification and Resolution

Annual Follow-up of Selected Issues (IP Section 03.03) (2 Samples)

The inspectors reviewed the licensee's implementation of its corrective action program related to the following issues:

(1) Review of NextEra's evaluation and corrective actions associated with recurrent voids in the containment building spray piping
(2) Review of NextEra's evaluation and corrective actions for the containment internal structures to include the effects of alkali-silica reaction

71153 – Follow-up of Events and Notices of Enforcement Discretion

Event Report (IP Section 03.02) (1 Sample)

The inspectors evaluated the following licensee event reports (LERs):

(1) LER 05000443/2023-04-00 for Seabrook Station, Manual Reactor Trip due to Main Turbine Electro-Hydraulic Control Oil Level (ADAMS Accession Number ML23270B916). The inspectors determined that the cause of the condition described in the LER was not reasonably within the licensee's ability to foresee and correct, and therefore was not reasonably preventable. No performance deficiency nor violation of NRC requirements was identified. This LER is closed.

Notice of Enforcement Discretion (IP Section 03.04) (1 Sample)

(1) The inspectors evaluated the accuracy of the licensee's oral assertions and provided information, implementation of required compensatory measures, and the adequacy of licensee corrective actions and cause determinations surrounding the Notice of Enforcement Discretion EA-24-027, which can be accessed at https://www.nrc.gov/docs/ML2406/ML24065A252.pdf, on March 5, 2024

INSPECTION RESULTS

Very Low Safety Significance Issue Resolution Process: Reserve Auxiliary Transformer 3B Testing	71111.24
This issue is a current licensing basis question and inspection effort is being discontinued in accordance with the Very Low Safety Significance Issue Resolution (VLSSIR) process. No further evaluation is required.	
Description: On March 1, 2024, a fault on the 'B' reserve auxiliary transformer (3B RAT) required replacing the 3B RAT with the onsite spare. The replacement and post maintenance testing was completed on March 25, 2024. The inspectors identified that during the post replacement testing of the 3B RAT, the licensee did not transfer (manually or automatically) unit power supply to the safety-related 4160-volt emergency bus from the normal circuit (unit auxiliary transformer) to the alternate circuit (RAT). The replacement 3B RAT was last load tested as part of the factory acceptance testing performed in 1983. While no loaded test of the 3B RAT was completed after installing, the licensee credited transformer overlap testing combined with a suite of other post maintenance testing to provide reasonable assurance that the RAT could perform its intended function. This testing included transformer winding testing, current transformer testing, as well as testing of the transformer auxiliary support systems such as instrumentation and alarms. Additionally, the licensee did not perform a functional test of the 4160-volt RAT breakers after they were reinstalled into the switchgear but modified their post maintenance testing procedure (MA 3.5) to allow an alternate breaker testing method. This testing involved cycling the breaker in the test position prior to racking the breaker into the switchgear and then visually checking the charging springs, the closing spring charging motor toggle switch, the breaker indicating light, the latching mechanism, and then performing electrical checks to validate the breaker is racked in. The inspectors reviewed the design and construction of the offsite power system. The main generator is connected to the 345-kilovolt switching station through three single-phase generator step-up transformers (GSUs). Two unit auxiliary transformers (UATs) are tapped off the isolated phase bus duct connecting the generator to the GSU transformers. The UATs are the normal source of power to the onsite distribution system when the generator is online. Each UAT is connected to a 13.8-kilovolt switchgear bus and the tertiary winding is connected to one train of 4160-volt emergency switchgear and to one 4160-volt nonessential switchgear lineup. By this arrangement, a separate UAT feeds each emergency bus. The UATs and GSUs provide an immediate access circuit from the preferred power supply (offsite source) to the onsite distribution system, providing power for all loads including all the engineered safety features loads of the unit, when the generator circuit breaker is tripped. Two RATs provide a second immediate access circuit from the preferred power supply (offsite source) to the onsite distribution system, providing electrical power for all loads including all the engineered safety features loads. An automatic fast transfer from the UAT to the RAT will occur to ensure safety-related loads are maintained upon the loss of a UAT. Licensing Basis: NextEra Energy Seabrook's Operating License, Appendix A, Technical Specifications, contains surveillance requirement 4.8.1.1.1.b which states, in part, "Each required independent circuit between offsite transmission network and the Onsite Class 1E Distribution System shall be demonstrated OPERABLE by transferring (manually and automatically) unit power from the normal circuit to the alternate circuit. "Surveillance requirement 4.8.1.1.1.b is modified by footnote" *prohibiting performance during MODE 1 or 2." The reason for the footnote is that, during operation with the reactor critical, performance of this surveillance requirement could cause perturbations to the electrical distribution	

systems that could challenge continued steady state operation and, as a result, unit safety systems.

10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants, Criterion 17 - Electrical power systems states, in part, "Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits designed to minimize the likelihood of their simultaneous failure under operating and postulated accident conditions. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the offsite power circuit."

10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants, Criterion 18 - Inspection and testing of electrical power systems states, in part, "Electric power systems important to safety shall be designed to permit appropriate periodic testing of important areas and features to assess the continuity of systems and the condition of their components. The systems shall be designed with a capability to test periodically the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system."

The Seabrook Station updated final safety analysis report (UFSAR) discusses compliance with General Design Criteria 17 and NRC Regulatory Guide 1.32 in Section 8.2.1.5 and compliance with General Design Criteria 18 in Section 8.2.1.6. It is stated that the electrical power system is designed to permit testing of the operability and functional performance of the transfer scheme which performs the transfer of auxiliary power from the nuclear power unit to the offsite preferred power supplies. It also states that the various automatic and manual transfer schemes for each bus from one source to the other are described in UFSAR Section 8.3.1.1b.3 and that the schemes described can be tested during normal plant operation.

The Seabrook Station UFSAR discusses the automatic and manual transfers associated with 4160-volt distribution system in Section 8.3.1.1b.3 and states, in part, "The opening of the UAT incoming breaker, either manually or automatically, initiates an automatic transfer from UAT to RAT source, provided that the RAT transformer is energized, no fault exists on the bus, and bus voltage is in synchronism with the RAT voltage at the time of transfer initiation or has decayed to an acceptable value. A manually initiated, synchronism check relay-supervised, live bus transfer is provided for transferring a 4160-volt bus from the UAT source to the RAT source, and vice versa, provided the source to which the bus is being transferred is energized, and the two sources are in synchronism."

The inspectors believe that, based on information gathered to date, uncertainty exists in the Seabrook Station licensing basis as to what testing is required following the 3B RAT replacement; specifically, whether Seabrook is required to transfer (manually or automatically) unit power supply to the safety-related 4160-volt emergency bus from the UAT to the RAT to demonstrate the full operation sequence that brings the 3B RAT into operation. To resolve this uncertainty, it would be necessary to conduct further investigation to determine the scope and level of detail of the staff's original licensing review, in addition to other relevant information, to resolve the licensing basis.

Significance: Senior Reactor Analysts (SRA) performed a bounding analysis of the risk associated with a failure of the 3B RAT. This analysis assumed a non-recoverable loss of the 3B RAT at-power. No credit for use of diverse and flexible coping strategies equipment was assumed. This resulted in a bounding risk assessment of <1.0E-6/year delta core damage frequency, had a performance deficiency been identified. In addition, the SRAs and inspectors assessed the risk achievement worth for the 3B RAT provided by the licensee, which is a reasonable measure of risk increase if the 3B RAT was failed. Using this value (1.004), the SRAs estimated the core damage frequency <1.0E-6/year, which is consistent with the NRC assessment and therefore would be considered very low safety significance. The inspectors concluded that the change in risk was very low, that adequate safety margin is retained, that sufficient defense-in-depth is maintained, that there is adequate opportunity for monitoring, and the issue's safety significance is very low. The inspectors used the guidance from IMC 0612, Appendix B, Issue Screening Directions, effective August 9, 2023, Block 8 VLSSIR process, and determined this issue met the criteria for closure. No further evaluation is required. Technical Assistance Request: A technical assistance request was not initiated. Corrective Action Reference: 02480634

Failure to Promptly Identify and Correct a Condition Adverse to Quality for Gas Accumulation in Containment Building Spray Piping			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000443/2024001-01 Open/Closed	[P.3] - Resolution	71152A
The NRC inspectors identified a Green finding and associated non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," when the licensee did not identify and correct a condition adverse to quality. Specifically, from April 2023 to September 2023, the licensee did not sufficiently characterize a gas void in a section of 'B' train RHR and CBS suction piping, and did not ensure, as a corrective action, that the pipe remained sufficiently full to the next, scheduled technical specification monthly surveillance.			
<u>Description</u>: On June 18, 2023, licensee staff performed ultrasonic testing (UT) of emergency core cooling system (ECCS) piping high points to verify the piping sections were sufficiently full. During performance of ECCS-OT027, "Ultrasonic Testing of ECCS and CBS Points," locations CBS-1 ('A' train ECCS containment sump suction) and CBS-4 ('B' train ECCS containment sump suction) were identified with voids exceeding the respective technical specification surveillance procedure acceptance criteria of 3-5/8 inches and 4-5/8 inches, respectively. The inspectors noted that these acceptance criteria were identified in OX1456.02, "ECCS Monthly System Verification," Revision 25. The CBS-1 location as-found void size was 5-1/4 inches and CBS-4 location as-found void size was 6-3/4 inches. These 'A' and 'B' train piping segments, which were tested sequentially, were declared inoperable by licensee staff and subsequently vented to restore operability. These segments serve as suction sources for both RHR and CBS pumps, from the containment recirculation sump. The licensee staff performed a follow-up UT measurement identified at 4-1/4 inches. The inspectors observed that OX1456.02 directed engineering staff to evaluate the impact on operability, as well as to determine if a gas sample should be taken. On September 18, 2023, ECCS-OT027 was again performed, and			

an unacceptable gas void of 7 inches was discovered at location CBS-4. The inspector noted that licensee staff refilled and vented this portion of the CBS system, resulting in an as-left UT measurement of 4 inches.

The inspectors found that applicable technical specification requirements for the ECCS and CBS systems require them to be operable prior to entering Mode 4. The inspectors reviewed the monthly void verifications for these locations since plant restart from an April 2023 refueling outage and found the following as-found and as-left void sizes in Table 1:

Table 1: Summary of Monthly CBS Void location UT Values

Date	CBS-1 as-found	CBS-1 as-left	CBS-4 as-found	CBS-4 as-left
April 2023	3-5/8	3-5/8	4-3/32	4-3/32
May 2023	2-9/16	2-9/16	4-3/4	3-1/4
June 2023	5-1/4	2	6-3/4	4-1/4
July 2023	1	1	0	0
Aug. 2023	0	0	4	4
Sept. 2023	0	0	7	4-1/4

Note: CBS-1 Limit: 3-5/8 and CBS-4 Limit: 4-5/8

The inspectors noted that the licensee had identified corrective actions for the CBS-1 and CBS-4 voids discovered in June 2023, under action requests (ARs) 02460317 and 02460307, respectively. These actions were in addition to the monthly UT verifications, as well as fill and venting activities, if needed. They also included void trending, evaluating strategy changes, adding a high point at the bend of the 'B' train pipe where it is determined to be the highest point, performing a GOTHIC analysis to find a more realistic 'A' train limit for void size and pump operability, or performing a sloped piping calculation on CBS-1 to increase the margin.

However, the inspectors determined the licensee did not implement the existing corrective actions contained in procedure OX1456.02, to ensure these piping locations were sufficiently full during the intervening monthly surveillance intervals from April 2023 to September 2023. For example, the CAUTION statement prior to Steps 4.2.5 and 4.3.5, requires operators to take appropriate actions to “bring the system (ECCS voids) into compliance with the allowable values.” The inspectors noted that the recurrent void trends from April 2023 to September 2023, were addressed by fill and vent activities to values equal to, or just under, the acceptance criteria. These acceptance criteria corresponded to an operability limit associated with two percent void requirement, as identified in applicable design basis calculations, performed following the issuance of Generic Letter 2008-01, “Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems.” Additionally, the inspectors found that procedure OX1456.02 directed engineering staff to perform evaluations of voids that exceeded the acceptance criteria. The inspectors also found that the approach of removing only enough of the void to meet acceptance criteria was not evaluated by engineering staff who had initiated and planned broader corrective actions to better preclude voiding at these locations.

Corrective Actions: The licensee continued with the complex troubleshooting process to determine the source of the voids, consistent with the support/refute matrix developed under AR 02460307. ECCS-OT027 is performed on a monthly periodicity and no voids have been detected in CBS-1 or CBS-4 since October 2023. The monthly surveillance procedure was also revised in October 2023, which provided additional void removal capabilities through

available valve and system alignments conducted under procedure change request (PCR) 02467932. Procedures are also being developed to aid in sampling of the gas accumulation if additional voids are detected in the future, as directed in AR 02478078. Corrective Action References: 02460307, 02478078
Performance Assessment: Performance Deficiency: The inspectors determined that, from April 2023 to September 2023, the licensee did not sufficiently characterize a gas void in a section of 'B' train RHR and CBS suction piping or fully refill the location consistent with their procedures to ensure the piping remained sufficiently full to the next, scheduled technical specification monthly surveillance. This was determined to be a performance deficiency. Screening: The inspectors determined the performance deficiency was more than minor because if left uncorrected, it would have the potential to lead to a more significant safety concern. Specifically, failure to identify and correct recurrent gas voiding in CBS suction piping was a condition adverse to quality and resulted in the inoperability of the 'B' train of the CBS and RHR systems. Significance: The inspectors assessed the significance of the finding using IMC 0609 Appendix A, “The Significance Determination Process for Findings At-Power.” The inspectors determined that this finding is of very low safety significance (Green) since it did not affect the design or qualification of a mitigating structure, system, or component and it did not represent the loss of any system, function, or train of equipment for greater than 24 hours or the technical specification allowed outage time. Cross-Cutting Aspect: P.3 - Resolution: The organization takes effective corrective actions to address issues in a timely manner commensurate with their safety significance. Specifically, the licensee did not implement appropriate corrective actions to resolve recurrent gas accumulation in the ECCS and CBS systems that had yielded unpredictable results during monthly UTs.
Enforcement: Violation: 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action,” requires in part that, “measures shall be established to assure conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment and non-conformances are promptly identified and corrected.” Contrary to the above, from April 24, 2023, to September 18, 2023, the licensee failed to promptly identify and correct a condition adverse to quality. Specifically, gas accumulation was allowed to remain in the CBS system suction piping which likely resulted in void sizes that rendered the 'B' trains of CBS and RHR systems inoperable on September 19, 2023. Enforcement Action: This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy.

Failure to Adequately Evaluate the Reactor Pit Slab During Reanalysis of the Containment Internal Structures to Incorporate Alkali-Silica Reaction Loads			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Barrier Integrity	Green NCV 05000443/2024001-02 Open/Closed	[P.2] - Evaluation	71152A
The NRC inspectors identified a Green finding and associated non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control" because the licensee did not verify the adequacy of the reactor pit slab at elevation (-) 44ft-9in for applicable design loads in the revised structural evaluation for the CIS.			

Description: The reactor pit area is a normally inaccessible area inside the containment building that provides an access point for the reactor under vessel area and a pathway for in-core instrumentation cabling to reach the underside of the reactor vessel. The reactor pit area is a seismic category I concrete structure and is categorized as part of the CIS. During the fall 2021 refueling outage (designated OR21), the licensee identified spalling of the concrete floor slab and bowing of the adjacent metal deck grating located in the reactor pit area North end at elevation (-) 44ft-9in.

Licensee staff performed field inspections and walkdowns of the reactor pit areas in 2014, 2017, and 2018, and identified the presence of structural indications (e.g., cracking and gaps) suggestive of potential ASR effects in multiple walls. The NRC documented a finding (FIN 2021004-02 (ML22040A204)) because the licensee did not evaluate these structural indications in accordance with their structural monitoring program which would have accounted for added load demands from potential ASR related building deformation. In response to the finding, licensee staff performed a Stage 2 structural evaluation of the CIS that revised the original structural design calculations to include additional loading resulting from assumed ASR expansion in the containment foundation and to account for thermal loads explicitly via finite element modeling. The licensee completed the structural evaluation on November 7, 2023, and concluded the CIS condition was in accordance with the Seabrook design and licensing bases.

The inspectors reviewed the Stage 2 structural evaluation documents, one for the reactor pit area and the other for the remainder of the CIS above the reactor pit area. The inspectors also reviewed the licensee's root cause studies which modeled and evaluated load combination scenarios of operating thermal conditions in the reactor pit area and ASR expansion levels in containment foundation and containment internals structures to estimate a limiting amount and location of ASR expansion in the structure. The inspectors observed that the licensee incorporated ASR loading in accordance with the NRC-approved methodology document for all structural components of the CIS except for the reactor pit floor slab located at elevation (-) 44ft-9in. This one-foot-thick slab provides a landing for personnel to access the in-core instrumentation tubing below via a ladder and supports a ventilation duct attached on the underside.

The inspectors discussed their observation regarding the omission of the slab with licensee staff and noted that this slab was not analyzed for the applicable design loads (e.g., ASR, seismic, thermal) in the revised structural design calculation. The inspectors observed the licensee staff's review of the original design calculation showed this slab was not credited as part of the seismic lateral load path and did not contribute to the overall seismic capacity of the structure. However, the inspectors considered that the slab is part of the CIS, a seismic category I structure, and is to maintain its integrity because it is credited to support the attached ductwork and not impact the safety-related reactor in-core instrumentation located below it.

Per the Structures Monitoring Program ASR methodology document, the structural evaluation of the CIS structure should have included any additional ASR load demands for comparison to the total structural load in accordance with the design code of record, ACI 318-71, as amended by License Amendment 159, to ensure that all components / members of the structure remained in compliance with the code. The inspectors determined that not evaluating the applicable design loads for the reactor pit slab, including the added demands from ASR, in the revised structural evaluation for the CIS was a performance deficiency.

Corrective Actions: Based on discussions with the inspectors, licensee staff generated a condition report and initiated a prompt operability determination to confirm the integrity of the reactor pit slab at elevation (-) 44ft-9in for the unusual load combination with a safe shutdown earthquake. The inspectors reviewed the analysis to determine the slab would perform its function and not impact the safety-related equipment located underneath it. The licensee planned corrective actions to perform a detailed analysis of the reactor pit slab for the applicable load combinations and consideration of reclassifying the slab as “seismic category II over category I equipment.” The licensee previously generated AR 02481629 to evaluate a future preventative retrofit for protection of components below the slab in the event of concrete spalling.

Corrective Action References: 02483088, 02483585

Performance Assessment:

Performance Deficiency: The inspectors determined that the failure to evaluate the applicable design loads for the reactor pit slab, including the postulated added demands from ASR, in the revised structural evaluation for the CIS in accordance with the Structures Monitoring Program and the NRC-approved methodology document was a performance deficiency.

Screening: The inspectors determined the performance deficiency was more than minor because if left uncorrected, it would have the potential to lead to a more significant safety concern. Specifically, unanalyzed loading of the slab and additional loads from ASR challenged the structural integrity of the slab where failure could impact in-core instrumentation and result in reactor coolant system leakage.

Significance: The inspectors assessed the significance of the finding using IMC 0609 Appendix A, “The Significance Determination Process for Findings At-Power.” The inspectors determined the finding to be of very low safety significance (Green), because the structure, specifically the reactor pit slab, maintained its functionality based on a subsequent supporting evaluation.

Cross-Cutting Aspect: P.2 - Evaluation: The organization thoroughly evaluates issues to ensure that resolutions address causes and extent of conditions commensurate with their safety significance. The licensee did not evaluate the design loads for postulated ASR on the reactor pit slab or demonstrate the impacts of slab failure were acceptable until inspectors inquired in this regard.

Enforcement:

Violation: 10 CFR 50, Appendix B, Criterion III, “Design Control”. The design control measures shall provide for verifying or checking the adequacy of design, such as by the

performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. Contrary to the above, from November 7, 2023 to present, the licensee did not verify the adequacy of the structural design of the reactor pit slab located at elevation (-) 44ft-9in to ensure it would perform its design function and not impact safety-related equipment. Enforcement Action: This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy.

Observation: Review of Evaluation and Corrective Actions for the Containment Internal Structures to Include the Effects of Alkali-Silica Reaction	71152A

An NRC inspector accompanied by a senior civil engineer (structural) from the NRC Office of Nuclear Reactor Regulation completed an onsite inspection at the Seabrook Station from March 24 to March 29, 2024 to review the licensee's performance to monitor reinforced concrete structures affected by ASR and to provide for corrective actions. A senior technical advisor for civil/structural engineering from the NRC Office of Nuclear Regulatory Research also provided remote support and reviews.

The licensee completed root cause studies of observed structural indications (such as spalling and cracks) and applied the results of these studies as inputs into their Stage 2 structural evaluations for the CIS. The inspectors reviewed the root cause studies, Stage 2 structural evaluations, and corrective actions associated with the CIS (action requests 02408546, 02416500, and 02471465). The inspectors considered whether the licensee's evaluations and corrective actions were in conformance with the standards in their Structures Monitoring Program, NRC-approved methodology document, and corrective action program procedures.

<u>Root Cause Studies for the CIS</u>

The NRC documented a Green finding in the 4Q2021 NRC integrated inspection report 05000443/2021004 (ML22040A204) regarding an instance in 2017, where licensee staff did not document and evaluate conditions observed as indicative of potential impacts from ASR identified in the CIS reactor pit area in accordance with NextEra Engineering Department Standard 36180, "Structural Monitoring Program." In response to the finding, the licensee generated a finite element model for both thermal (heat transfer) and stress analyses and conducted a series of parametric studies to identify the likely causes of the observed structural distress in various areas of the CIS. The licensee developed a global model for the entire CIS and localized models for areas of observed distress. Licensee staff documented that no typical visual symptoms of ASR (pattern cracking, dark staining, gel) were observed in the accessible CIS surfaces and that the CIS atmosphere is dry with low humidity levels. The following conditions were evaluated:

- *Reactor pit area:* Spalling of the concrete reactor cavity pit floor slab at EL (-) 44ft-9in and buckled angle supporting adjacent metal walkway grating, gap between wall and slab, horizontal cracks along walls and circumferential cracks on the floor beneath the reactor vessel.

- *EL (-) 26ft fill mat slab and adjacent to the sump at azimuth 80^o:* Circumferential / intermittent radial surface cracking in the concrete fill mat slab; cracking in the 2ft thick walls adjacent to the sump at azimuth 80°, and gaps between the bottom of these walls and the fill mat slab.

- *Personnel elevator with base at EL (-)26 ft azimuth 335°:* Buckling of an enclosure plate and gap on the southwest corner of the elevator shaft; gaps between the steel angle column base plates and supporting grout; and minor cracking and spalling of the grout beneath the base plates in southwest and southeast corners of the shaft.

The inspectors noted these parametric studies analyzed different combinations and bounding scenarios of in-situ load conditions (e.g., thermal and ASR) and evaluated whether the results reasonably correlated with the observed distress (in-situ loads are actual loads not amplified by safety factors). Specifically, the licensee's studies determined that the distress in the reactor pit area, including the slab at EL (-) 44ft-9in, can be primarily attributed to thermal loading and cycling fatigue, stress concentrations, and a possible contribution of combined ASR expansion and swelling in the containment building foundation of up to 0.2 mm/m in each direction (i.e., up to 0.6 mm/m volumetric). The inspectors noted the foundation is up to 10 feet thick and located beneath the containment liner and fill mat. The studies showed that ASR expansion beyond this estimate would have produced more severe cracking in the reactor pit walls considering the in-situ temperature conditions in the CIS. The studies concluded that observed distresses near the sump and personnel elevator areas were likely due to effects other than ASR. Even though there are no characteristic visual ASR symptoms on the inner surfaces of the fill mat or the reactor pit walls, the studies provided monitoring recommendations, which the inspectors found were being tracked for implementation, to monitor for the potential effects of ASR in these components and to further support their upper-bound estimates of expansion in the containment building foundation.

The inspectors concluded that the licensee's root cause studies were of appropriate technical detail to develop insights and provide for upper limit estimates of possible ASR/swelling expansion that correlated with field observations of observed distresses in the CIS. The inspectors also determined that the recommended monitoring locations and actions to detect future indications of ASR effects, if any, were incorporated into Seabrook's Structures Monitoring Program to be implemented on a 3-month and 18-month frequency.

<u>Stage 2 Structural Evaluations for the CIS</u>

On December 19, 2023, licensee staff closed the prompt operability determination for the CIS based on their CIS Stage 2 structural evaluations and established threshold monitoring of key areas. In tracking AR 02471465, licensee staff stated they re-evaluated the CIS for all design loads, including ASR and swelling. The licensee staff documented the CIS meets the evaluation criteria of ACI 318-71 as amended by License Amendment 159 for all UFSAR load combinations with the ASR load amplified by a threshold factor of 1.3 to account for potential future ASR expansion.

The inspectors reviewed the structural evaluations and noted they were separated into two calculations: one for the CIS reactor pit area (pit walls and base but excludes the pit slab at EL (-) 44ft-9in), and the second for the CIS superstructure above the reactor pit area (concrete walls, slabs and columns including and above the 4 ft thick fill mat). The reactor pit area is below EL (-) 30 ft and the CIS superstructure is above EL (-) 30 ft.

The inspectors found that the CIS structural analyses were performed using appropriate 3D finite element models using ANSYS 15.0 computer software for current licensing basis design loads and load combinations stipulated in UFSAR Table 3.8-14. This included an estimated ASR expansion load of 0.2 mm/m in each direction (applied as 0.6 mm/m volumetric strain) in the containment building foundation mat and amplified by a threshold factor of 1.3 to account for potential future ASR expansion. The ASR expansion value of 0.2 mm/m was based on an upper limit estimate determined from the root cause studies of observed distress in the CIS reactor pit area and reactor pit slab at EL (-) 44ft-9in. The evaluation also assessed that an adequate isolation gap remained between the CIS and adjacent containment building. The inspectors verified the licensee performed the evaluations in accordance with their Structures Monitoring Program and Stage 2 analysis process described in the NRC-approved methodology document and met the acceptance criteria for structural demands and stability consistent with the Seabrook current licensing basis as amended by License Amendment 159.

The inspectors determined that the structural evaluations specified appropriate threshold monitoring locations and limits consisting of a combination of quantitative and qualitative monitoring that, except for one location near azimuth 270° sump under consideration for alternate monitoring, have been incorporated in the Structures Monitoring Program and will be implemented on a 3-month schedule for items located outside the secondary shield wall and an 18-month schedule for items located in the reactor pit area. The inspectors noted the monitoring frequencies were consistent with the specified threshold monitoring interval for Stage 2 structures as required by the Structures Monitoring Program.

However, the inspectors noted the licensee did not evaluate the structural adequacy for the applicable design loads and load combinations, including the additional demands from ASR, of the reactor pit slab at EL (-) 44ft-9in in the revised structural evaluation for the CIS reactor pit area. This was a performance deficiency, because the functionality of safety-related components located/supported underneath the slab could be adversely affected and is documented as NCV 2024001-02 in this report.

<u>Corrective Actions Associated with the CIS</u>

The inspectors found that the licensee plans to confirm the temperature input assumption utilized in their revised structural evaluations for the CIS following data collection that should be available from an installed data logger during the fall 2024 refueling outage. Licensee staff previously installed thermocouples in the reactor pit area at EL (-) 44ft-9in with a data logger to collect temperature data during the spring 2022 outage. If found to be significantly different, the temperature data could result in revisions to ASR expansion estimates used in the revised CIS evaluations.

The Stage 2 CIS analyses used a design input of 0.2mm/m for ASR expansion in each direction in the containment foundation, which the inspectors' observed interfaces with the above fill mat as part of the CIS. This value was determined to be an upper limit estimate based on the root cause studies to explain the observed distresses. The inspectors noted that ASR expansion in the foundation can potentially impact the evaluation of the containment liner and the evaluation of the containment building. The inspectors noted that the containment building evaluation, performed as a Stage 1 structural evaluation (FP 101113, Evaluation of Containment for ASR Effects, Revision 1, dated 09/01/2020), did not apply ASR expansion to the containment building foundation. The licensee generated AR 02482677 to determine whether the estimated ASR expansion should also be applied to the existing containment building evaluation. The inspectors noted that there is no immediate safety concern resulting from this because the existing containment Stage 1 evaluation has a threshold factor of 1.8, indicating adequate margin exists for future ASR expansion, and the licensee's preliminary evaluation of the impact of the potential 0.2 mm/m expansion in the foundation indicated that the liner strain would remain within the allowable limit. The inspectors noted the licensee's plan to verify the temperature input assumption used for the CIS and evaluate the impacts of assumptions used in other analyses was appropriate.

EXIT MEETINGS AND DEBRIEFS

The inspectors verified no proprietary information was retained or documented in this report.

- On April 12, 2024, the inspectors presented the containment building spray piping voids inspection results to Jeff Sobotka, Engineering Director, and other members of the licensee staff.
- On May 8, 2024, the inspectors presented the integrated inspection results to David Sluszka, Site Vice President, and other members of the licensee staff.

DOCUMENTS REVIEWED

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
71111.04	Corrective Action Documents	02476921		
		02476938		
		02478044		
	Corrective Action Documents Resulting from Inspection	02476709		
	Drawings	1-MAH-B20495	Miscellaneous Air Handling PAB & Containment Enclosure Ventilation Area Detail	Revision 18
		1-MS-B20582	Main Steam Emergency Feedwater Pump Supply	Revision 21
		1-NHY-310002	Unit Electrical Distribution One Line Diagram	Revision 11
		1-NHY-310042	125VDC Vital Distribution System	Revision 5
		1-NHY-310107	125VDC Bus 11B 1-SWG-11B	Revision 14
		1-SW-O20795	Service Water Nuclear Detail	Revision 05
	Miscellaneous	FP32364	Industrial Batteries and Chargers	Revision 19
	Procedures	ON1090.12	Equipment Long Term Layup	Revision 05
		OS1023.66	Containment Enclosure Ventilation System Operation	Revision 22
	Work Orders	40894415		
		40895407		
		40953805		
71111.05	Corrective Action Documents	02465524		
		02477381		
		02479305		
		02482318		
	Corrective Action Documents Resulting from Inspection	02480751		
	Drawings	FP62296	Fire Protection Pre-Fire Strategies	Revision 45
	Fire Plans	ISP-24-4327		
	Miscellaneous	CMP-23-4795	Transient Combustible Materials Permit	08/25/2023

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
		Fire-24-034	Fire Patrol	01/26/2024
	Procedures	FP-AA-104-1004	Control of Combustibles and Ignition Sources	Revision 4
		FP-CP-557		Revision 02
		FP2.2A	Control of Combustible Materials	Revision 24
		OS1200.00	Response to Fire or Fire Alarm Actuation	Revision 28
		OS1200.00A	Fire Hazards Analysis for Affected Area/Zone - Appendix A	Revision 26
		OX0443.06	Deluge And Pre-action Sprinkler Valve 18 Month Actuation Test, Flow And System Alarms Test	Revision 19
	Work Orders	40826770		
		40966695		
71111.11Q	Procedures	AD-AA-100-1006	Procedure and Work Instruction use and Adherence	Revision 25
		OP-AA-100-1000	Conduct of Operations	Revision 39
71111.12	Corrective Action Documents	02468533		
		02479437		
	Procedures	ER-AA-100-2002	Maintenance Rule Program Administration	Revision 14
71111.13	Corrective Action Documents	02479158		
		02479435		
		02479436		
	Drawings	1-NHY-310002	Unit Electrical Distribution One Line Diagram	Revision 49
	Procedures	OP-AA-102-1003	Guarded Equipment	Revision 46
		OP-AA-104-1007	Online Aggregate Risk	Revision 8
		OX1446.01	AC Power Source Weekly Operability Surveillance	Revision 27
	Work Orders	40811315		
		40905805		
		40965618		
71111.15	Corrective Action Documents	02408546		
		02479436		
		02479446		
		02480016		
		02408546		
	Work Orders	40879954		
		40965618		
71111.24	Corrective Action	02468308		

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
	Documents	02479152		
		02479436		
		02480060		
	Drawings	1-NHY-310857, Sheet 82	DG-1A JKT Cool STBY Circ Pump 1-P-120A	Revision 7
		PID-1-CS-B20725	Chemical and Volume Control Charging System	Revision 33
	Engineering Changes	EC14152		
		EC298710		
		EC299475		
		FP62322		
		FP98333		
	Procedures	ES1850.001	Check Valve Performance Monitoring Program	Revision 15
		LX0536.06	4.16kV Loss of Voltage Protection Quarterly Surveillance	Revision 11
		LX0536.07	4.16kV Bus Degraded Voltage Protection Quarterly Surveillance	Revision 10
		MA3.5	Post Maintenance Testing	Revision 28
		MA3.5	Post Maintenance Testing	Revision 28
		MA3.5	Post Maintenance Testing	Revision 28
		MS0517.43	Piping Installation And Maintenance	Revision 4
		MS0523.29	Inspection and Repair of Lube Oil Pump Coupling	Revision 4
		MS0539.28	A-EDG Coolant Recirculation, Draining and Refilling	Revision 20
		OS1046.04	345-kilovolt Operations	Revision 63
		OS1046.79	Breaker Racking Operations	Revision 5
		OX1416.01	Monthly Service Water Valve Verification	Revision 14
		OX1416.05	Service Water Cooling Tower Pumps' Quarterly and 2 Year Comprehensive Test	Revision 30
		OX1430.03	Main Steam Drain Valve Operability Tests	Revision 10
		OX1436.02	Turbine Driven Emergency Feedwater Pump Quarterly And Monthly Valve Alignment	Revision 37
		OX1436.03	Electric EFW Pump Quarterly, 18 Month, 30 Day Cold Shutdown, and Comprehensive Pump Test, and Monthly Valve Stroke Verification Surveillance	Revision 30

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
		OX1456.01	Charging Pump A&B Quarterly Flow and Valve Stroke Test	Revision 33
		OX1456.86	Operability Testing Of IST Pumps	Revision 31
	Work Orders	40794366		
		40808876		
		40854497		
		40869450		
		40873904		
		40875289		
		40877946		
		40878489		
		40883560		
		40946791		
		40951737		
		40965618		
		40966727		
71114.06	Miscellaneous	CFD 24-01	Seabrook Emergency Preparedness	1/31/2024
	Procedures	EPDP-03	Emergency Preparedness Performance Indicators	Revision 27
71152A	Calculations	Calculation CI-32	Containment Internals – Fill Mat at EL -26'-0" Design of Radial ad Shear Reinforcement	Revision 6
		Calculation CI-34	Containment Internals - Design of Reactor Cavity Pit	Revision 2
		Calculation CI-46	Containment Internals - Design of Primary Shield Wall	Revision 5
		Calculation CI-62	Containment Internals – Evaluation of Analysis of Containment Internals	Revision 1
		Calculation CS-23	Liner Strain - Containment Structure	Revision 1
		FP101532	Reanalysis of Containment Internal Structures (CIS) [except Reactor Pit]	Revision 0
		FP101536	Reanalysis of Containment Internal Structures (CIS) – Reactor Pit	Revision 0
	Corrective Action Documents	02460307		
		02467135		
		02469935		
		02481381		

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
	Corrective Action Documents Resulting from Inspection	02477416		
		02478078		
		02482093		
		02482677		
		02482694		
		02483034		
		02483088		
	Engineering Evaluations	EE-98-002	Evaluation of ECCS High Points	Revision 6
		FP101483	POD Evaluation of Containment Internal Structure due to possible ASR expansion of the containment pit and containment foundation	Revision 0
		FP101517	Root Cause Analysis of structural distress observed in the Containment Internal Structures (CIS) reactor pit area	Revision 0
		FP101518	Root Cause Analysis of Containment Internal Structures (CIS) structural distress observed in the walls adjacent to the sump at Azimuth 80	Revision 0
		FP101519	Root Cause Analysis of structural distress observed in the Containment Internal Structures (CIS) personnel elevator area	Revision 0
	Miscellaneous	FP101196	Methodology for the Analysis of Seismic Category 1 with concrete Affected by Alkali-Silica Reaction	Revision 3
		SMPM	Seabrook Station Structures Monitoring Program Manual	Revision 16
	Work Orders	40867851-02		
		40947667-02		
71153	Corrective Action Documents	02463450		