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

March 26, 2012

Mr. Paul Freeman
Site Vice President, North Region
Seabrook Nuclear Power Plant
NextEra Energy Seabrook, LLC
c/o Mr. Michael O'Keefe
P.O. Box 300
Seabrook, NH 03874

SUBJECT: SEABROOK STATION - NRC INSPECTION REPORT 05000443/2011010 RELATED TO ALKALI-SILICA REACTION ISSUE IN SAFETY RELATED STRUCTURES

Dear Mr. Freeman:

On January 20, 2012, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Seabrook Station. The enclosed inspection report documents the inspection results, which were discussed at the exit meeting with you and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel. In conjunction with the follow-up of two unresolved items, the focus of this inspection was a review of activities involving NextEra's analysis and evaluation related to addressing the Alkali-Silica Reaction (ASR) issue occurring in safety related and other important to safety concrete structures. As a part of this inspection, we reviewed your original and revised Prompt Operability Determinations (POD) for certain affected structures.

During the exit meeting, Mr. Richard J. Conte, Chief Engineering Branch 1, summarized the findings and observations. In addition, he discussed NRC observations regarding your planned corrective actions and assumptions being made in the NextEra operability determinations. The inspectors concluded that these structures can currently perform their safety related functions despite the observed degradation due to ASR. However the NRC still has concerns associated with long term operability, therefore additional information is needed to determine: 1) how various characteristics of the concrete may be affected by ASR; 2) the related effects on other elements of the structures, such as rebar, due to groundwater in-leakage; and 3) the rate of progression of the ASR in structures at the site. It is our understanding that these specific areas are being addressed in a comprehensive corrective action plan that was still being finalized by your organization at the end of the inspection.

Therefore, we request that you summarize your plans to address the above issues at a management meeting to be conducted April 23, 2012, at NRC Headquarters in Rockville, MD. At the meeting you should be prepared to focus on the following technical issues: 1) describe which applicable American Concrete Institute (ACI) 318 code relationships are affected by ASR and your plans to ensure the applicable licensing and design bases remain valid; 2) describe your comprehensive plans to understand the related effects and overall progression of ASR, its cause, and actions to correct and/or mitigate the issue; and, 3) provide a timeline for key actions, including those to address long term operability, how the degradation affects the design basis, and longer term management of the ASR issue. During the meeting we will discuss your overall corrective action plans, including the documents to be submitted to the NRC on the docket.

Also, the report documents two NRC-identified findings of very low significance (Green) one of which involved a violation of NRC requirements. Because of the very low safety significance, and because they are entered into your corrective action program, the NRC is treating these findings as non-cited violations, consistent with Section 2.3.2 of the NRC Enforcement Policy. If you contest any non-cited violations 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, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at Seabrook Station. In addition, if you disagree with the cross-cutting aspect assigned to 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 Seabrook.

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

Sincerely,

Christopher G. Miller, Director
Division of Reactor Safety

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

Enclosure:
Inspection Report No. 05000443/2011010
w/Attachment: Supplemental Information

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and your plans to ensure the applicable licensing and design bases remain valid; 2) describe your comprehensive plans to understand the related effects and overall progression of ASR, its cause, and actions to correct and/or mitigate the issue; and, 3) provide a timeline for key actions, including those to address long term operability, how the degradation affects the design basis, and longer term management of the ASR issue. During the meeting we will discuss your overall corrective action plans, including the documents to be submitted to the NRC on the docket.

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/RA/

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	50-443
License No.:	NPF-86
Report No.:	05000443/2011010
Licensee:	NextEra Energy Seabrook, LLC
Facility:	Seabrook Station
Location:	Seabrook, NH 03874
Dates:	September 26–September 30, 2011 November 15–17, 2011 (Northbrook, Illinois) November 28–December 1, 2011 January 20, 2012 (Conference Call)
Inspectors:	M. Modes, Senior Reactor Inspector, Region I S. Chaudhary, Reactor Inspector, Region I W. Raymond, Senior Resident Inspector, Seabrook Atif Shaikh, Reactor Inspector, Region III
Accompanied by:	A. Sheikh, Senior Structural Engineer, Office of Nuclear Reactor Regulation (NRR) G. Thomas, Structural Engineer, NRR
Approved by:	Richard J. Conte, Chief Engineering Branch 1 Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000443/2011010; 9/25/2011 – 12/2/2011; Seabrook Station (Problem Identification and Resolution; Follow-up to Operability and Plant Modifications).

This report covers an inspection by regional inspectors and resident staff, with assistance from the Office of Nuclear Reactor Regulation (NRR) structural specialists. Two Green findings were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). The cross-cutting aspects for the findings were determined using IMC 0310, "Components Within Cross-Cutting Areas." Findings for which the SDP does not apply may be Green, or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

Cornerstone: Mitigating Systems

Green. The inspectors identified a finding in that NextEra failed to fully evaluate potential structural and seismic response impacts in accordance with the requirements in NextEra procedure EN-AA-1001 after identifying a degraded and nonconforming condition related to degraded conditions for some safety related structures due to Alkali-Silica Reaction (ASR). Specifically, the evaluation did not consider the following effects due to changed properties of concrete, as reflected in reduced values of the modulus of elasticity as measured directly from concrete core samples: 1) building natural frequency in the dynamic response; 2) performance of anchorages and embedment of systems and components attached to the structures; and, 3) shear strength or capacity of affected structures and the dynamic/flexural response especially those buildings without corresponding shear reinforcement.

The failure to conduct adequate prompt operability determinations per procedure EN-AA-203-1001 for degraded and nonconforming conditions associated with ASR was a performance deficiency relative to a self imposed standard. Specifically, the prompt operability determinations conducted following the identification of ASR in safety-related structures did not completely analyze the effects of the reduced modulus of elasticity on the dynamic and flexural response of the structures to seismic events for certain conditions. This performance deficiency was associated with the design control aspect of the Mitigating Systems cornerstone; and, based on a comparison to Example 3.i of Appendix E of IMC 0612, it was determined to be more than minor. Specifically, the failure to conduct adequate operability determinations adversely affected the Mitigating Systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences because it required an additional evaluation to confirm that the design bases was met. The issue was evaluated using IMC 0609, "Significance Determination Process," and was determined to be of very low safety significance (Green). Specifically, when evaluated under IMC 0609, Attachment 4, the performance deficiency was a design or qualification deficiency confirmed not to result in an actual loss of safety function. The finding had a cross cutting aspect in the area of problem identification and resolution, P.1(c), related to ensuring that issues potentially impacting nuclear safety are thoroughly evaluated. Specifically, NextEra did not fully evaluate conditions adverse to quality, including evaluating the effects of the reduced concrete modulus of elasticity for impact on operability of the affected structures. (Section 4OA5.1.c)

Severity Level IV. The inspectors identified a Severity Level IV non-cited violation (NCV) of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.59(d)(1), "Changes, Tests, and Experiments," because NextEra did not adequately evaluate a "use-as-is" determination, resulting in a defacto design change, for certain ASR impacted safety related structures. Specifically, NextEra did not complete a 10 CFR 50.59 evaluation, to ensure that the identified reduction in concrete modulus of elasticity did not present a more than minimal increase in the likelihood of the occurrence of a malfunction of a structure, system, or component (SSC) important to safety previously evaluated in the updated safety analysis report (USAR) prior to implementing changes to the facility as described in the engineering change EC272057 issued on April 25, 2011.

The failure to evaluate changes to the facility as described in EC272057 was contrary to 10 CFR 50.59(d)(1) and was a performance deficiency warranting a significance evaluation in accordance with the NRC Enforcement Manual for Traditional Enforcement and IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening." The violation was determined to be more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," because it could not reasonably be determined that the changes would not have ultimately required prior NRC approval. In accordance with Section 6.1.d.2 of the NRC Enforcement Policy, this violation is categorized as Severity Level IV because the resulting changes were evaluated by the SDP as having very low safety significance (Green), because it was a design or qualification deficiency confirmed not to result in an actual loss of safety function and because further evaluation determined that the structures remained operable despite the degraded modulus condition. The finding had a cross cutting aspect in the area of human performance – work practices, H.4(b), because NextEra personnel did not follow procedures. Specifically, NextEra personnel did not follow the requirements of Section 5.2.2 of the 5059 Resource Manual when preparing the 50.59 screen for EC272057. (Section 4OA5.2.c)

REPORT DETAILS

Background

In June 2009, NextEra conducted walk downs of structures within the scope of license renewal as part of license renewal application preparations. In June 2010, the License Renewal Application (LRA) was received by the agency. In October 2010, the NRC staff noted that the licensee was beginning to formulate actions associated with both finalizing the operability determination for the control building (CB) and starting an extent of conditions review of other areas that may be subject to the alkali-silica reaction (ASR) degradation.

The ASR is a chemical reaction in concrete, which occurs over time in the presence of water, between the alkaline cement paste and reactive non-crystalline silica that is found in some common coarse aggregates. In the presence of water, the ASR forms a gel that expands, causing micro-cracks that change the physical structural properties[1] of the concrete, including compressive and tensile strength, modulus of elasticity, and Poisson Ratio. At Seabrook the below-grade concrete structures have experienced groundwater infiltration.

In the summer of 2010, NextEra performed an Immediate and Prompt Operability Determination (POD) for the CB "B" electrical tunnel structure based on core samples taken from the building. Inspection Report 05000443/2010004, issued November 1, 2010, documented the NRC review of the POD with no findings.

On May 12, 2011, Inspection Report 05000443/2011002 identified two non-cited violations (NCV) of very low safety significance related to maintenance rule (Title 10 of the *Code of Federal Regulations* (10 CFR) 50.65 a(1) and b(2)) monitoring of structures. One of the NCVs related NextEra's failure to properly monitor the structural performance of the CB resulting in degraded conditions - 10 CFR 50.65 (a)(1) (NCV 2011-002-01). Also in May 2011, License Renewal Inspection Report 05000443/2011007 (IP71002) reflected an overall inspection result as follows: "Except for Structures Monitoring Program, results support a reasonable assurance determination for license renewal." The structure monitoring program had not addressed the ASR condition.

On August 12, 2011, Inspection Report 05000443/2011003, identified a NCV of very low safety significance related to the untimely operability determinations regarding the extent of condition review for other buildings affected by ASR. The report also identified two unresolved items (URI) related to the operability determinations. Specifically the report identified: 1) the need for additional information related to open operability determinations, one for the CB "B" electrical tunnel, and the other operability determinations for the extent of conditions review for five other areas/structures with evidence of ASR (URI 2011-003-03); and, 2) potential inadequate screening in accordance with 10 CFR 50.59 for accepting the reduced values found on compressive strength and modulus of elasticity for the "B" Electrical Tunnel and the Containment Enclosure Building (URI 2011-003-02).

[1] Material properties defined in the supplemental section of this report

In September 2011, Region I obtained assistance from the Office of Nuclear Reactor Regulation (NRR) through a Task Interface Agreement (TIA) in order to assist in the review of the open PODs.

4. OTHER ACTIVITIES

4OA2 Problem Identification and Resolution (71152 – 1 sample)

Annual Sample: Corrective Actions Associated with Alkali-Silica Reaction in Safety Related Structures

a. Inspection Scope

This review was to assess progress in the development of a corrective plan and implementing schedule to address the ASR degradation issue including: initial assessments of all buildings potentially affected by the problem; root or apparent cause of the problem; control of in-situ testing such as crack mapping/indexing; control of contractor testing and laboratory test facilities in accordance with quality assurance requirements; and any mitigation or long term monitoring actions. The inspectors reviewed laboratory testing to address the ASR degradation with specific focus on the CB ("B" Electrical Tunnel). Laboratory testing was observed during the week of November 14, 2011, to ensure proper sample controls, test preparation, and conduct of the test.

During the week of November 28, 2011, the inspectors reviewed historical documentation from the construction phase of the plant, correlations between the concrete strength value determined by the recent core samples, and the original strength values determined at the time of concrete placement. The licensee's projected plan and schedule for further studies and assessment of the ASR problem were discussed and reviewed with cognizant engineering and management personnel. Inspectors also reviewed the licensee's control of contractors and laboratory facilities used to analyze concrete core samples. The inspectors reviewed the licensee's procedures for administration and control of engineering and testing service vendors and contactors. Additionally, the inspector reviewed the results and documentation of American Society of Mechanical Engineers (ASME) Code Section IWL inspection of the containment.

b. Findings and Observations

No findings were identified. The inspector noted that a comprehensive corrective action was still under development. NextEra classified this issue as a significant condition adverse to quality and was in the progress of completing a root cause analysis, which was scheduled to be completed in February 2012 in order to support an Engineering Evaluation in March 2012. The inspectors noted that NextEra's plans to date did not address some key issues related to ASR that include but are not limited to:

(1) Need for other concrete core testing (i.e., low stress range stiffness damage tests) to assess expansion-to-date or severity of degradation in the critical direction of the thickness with no rebar ties and lesser resistance to expansion;
(2) Basis for the representativeness of concrete core sampling in the buildings for those taken to date and those to be taken, should they occur;
(3) Impact of core boring and re-grouting on the building structural integrity; and
(4) Potential effects of other degradation mechanisms from an "aggressive" groundwater environment along with the presence of ASR.

Methods Used in Evaluating Structural Integrity

The NRC staff noted that the methods used in evaluating structural integrity for the selected buildings were based on the correct design basis code ACI 318-1971. However, the mathematical relationships in this code were based on empirical data, from testing of non-degraded concrete, for determining key ratios that are a part of the design bases and used for determining tensile and shear strength or capacity in addition to compressive strength. These strength values were important in the building loading analysis during normal or upset conditions such as for seismic events. More importantly, while some testing for the modulus of elasticity was done, it was not clear if the plans would result in additional testing of concrete cores for this parameter or any independent testing associated with other key design parameters such as Poisson's Ratio, shear modulus, or bulk modulus. With these parameters known, various strengths or capacities can be determined such as for tensile and shear strength. In addition, the plans that the inspectors reviewed did not address variation in mechanical properties of the concrete in different directions due to ASR cracking nor the effect of the ASR expansion on stresses in the rebar. These parameters were important in order to ensure that the current licensing and design basis was maintained. The licensee representatives agreed to address the assumptions or establish relationships for the current conditions at Seabrook. Accordingly this area is unresolved pending completion of license actions as noted above and further NRC staff review
(URI 05000443/2011010-01, Corrective Actions Associated with calculation methods used to address the ASR Issue)

Control of Contractors/Vendors and Laboratory Testing

The reviewers noted that NextEra had engaged knowledgeable vendors, appropriate consultants, and experts for testing, analysis, and evaluation of the effects of ASR on the serviceability and safety of the affected structures. Also, during the week of November 14, 2011, a Region III inspector reviewed laboratory testing for compressive strength on 15 concrete core samples taken from the CB "B" electrical tunnel in the October 2011 time frame. This testing was being completed to resolve discrepant information for compressive strength testing between two different contractors.

The testing was conducted at a laboratory in Northbrook, Illinois. All 15 core samples were compression tested. Photographs were taken for all core samples prior to loading for compression test and after fracture. Three cores had small length samples cut from them to be used by Seabrook for further petrography. Sample preparation (capping) was done in accordance with American Society for Testing and Materials (ASTM) C617.

Compression testing was done in accordance with ASTM C39. With respect to laboratory conditions for testing of concrete cores, the inspector verified: 1) organized and clean working area during both sample preparation (measurements and cutting) and compression testing; 2) adequate lighting available at all times; 3) ambient room temperature (~ 68°F) observed during preparation and testing; and 4) core samples were adequately stored and labeled in individual bags.

The inspector observed the care taken to ensure only one core was handled at any given time so as not to confuse cores during measurements, cutting, and testing. With respect to equipment calibration, the inspector verified proper equipment documentation and calibration. With respect to test technician qualifications, the inspector also verified qualification records. The inspector also reviewed the Altran Commercial Grade Dedication Plan.

No concerns were noted with respect to quality control during all aspects of compression testing. All 15 destroyed cores were shipped back to Seabrook including the cut samples to be used for petrography. These results were to be evaluated by NextEra.

4OA5 Other Activities

.1 (Open) Unresolved Item 05000443/2011003-03, Open Operability Determinations for Safety-Related Structures Affected by Alkali-Silica Reaction

a. Inspection Scope

The NRC staff reviewed NextEra actions to develop finalized operability determinations along with the review for extent of conditions. The review included the open aspects as documented in the originating inspection report for which NextEra was to provide additional information related to: 1) effect of the reduced modulus of elasticity on natural frequency of the structures (applied to CB – "B" Electrical tunnel and other structures being evaluated in the extent of conditions review such as for the Containment Enclosure Building (CEB); 2) the effect of the modulus of elasticity on structure flexural response as related to components attached to the structures, such as pipe and cable trays supports and their anchor bolts; 3) related effects from increased flexure of building on the loading and seismic effects on safety related pipes and cable tray supports; and, 4) effect of reduced parameters on the whole building (global) response of the CEB structure to seismic loads including further information of the effect on stress and strain in the concrete and rebar system. With respect to numbers 1 and 2 above, the inspectors reviewed the operability determinations for the below listed safety related structures degraded by ASR. The inspectors verified the basis for why the Radiological Control Area tunnel was confirmed to not be affected by ASR. The inspectors reviewed operability determinations for the following buildings:

- Control Building – "B" Electrical Tunnel,
- Containment Enclosure Building,
- Diesel Generator Fuel Oil Tank Rooms,
- Residual Heat Removal Equipment Vaults, and
- Emergency Feedwater Pump House.

The inspectors utilized site records and interviews to determine the design basis for the safety related structures in addition to those summarized in Sections 3.7 and 3.8 of the Updated Final Safety Analysis Report (UFSAR).

b. Observations

For the open aspects of numbers 1 and 2 above, a finding was identified and addressed in Section 4OA5.1.c. This section also noted a new issue identified by NRC staff related to shear reinforcement for the walls of the CB and the diesel generator building.

The open aspects of numbers 3 and 4 were updated but not completely resolved due to the need to obtain additional information. At the beginning of the inspection, the NRC staff review determined that the initial evaluation for the CEB did not address the open aspects of numbers 1 and 2 above; and, in particular, the response of the entire structure (whole building) to seismic loading comparable to the methods described in UFSAR 3.8. This included how the induced seismic stresses would shift between the concrete and the steel in adjoining sections of the structure. In response, NextEra noted that these issues would be factored into the analytical model (finite element analysis) to reanalyze the CEB using the as-measured worst case elastic modulus applied to ASR-impacted sections.

Revision 1 of the applicable operability determination for the CEB provided additional quantitative and qualitative analysis, for the available information, which addressed groundwater intrusion limited to less than 25 percent of the perimeter of the below grade portion of the building; the effect of the reduced modulus on the natural frequency; and the effect on shear capacity that indicated that the dynamic and flexural response had a minimal effect.

In conclusion, this area remained open pending further developments and completion of licensee actions as noted above and further NRC staff review. While this unresolved item remains open, the NRC staff determined that the affected safety-related structures can currently perform their safety functions. This conclusion was based on the following:

- Conservative safety load factors in controlling load conditions and engineering conservatisms in design provide reasonable expectation that affected structures can perform their safety function, despite the current licensing basis design margin being reduced by the change of mechanical properties;
- Field walk-downs confirm no visible indication of significant deformation, distortion, or displacement of structures, or rebar corrosion;
- Evidence of ASR limited to localized areas in the concrete walls; and

- Progression of ASR degradation occurs slowly based on existing operating experience and published literature, and the licensee continues to monitor.

This unresolved item related to operability of ASR affected safety related buildings remained open for NextEra to evaluate ASR effect on cable and pipe loadings (number 3) and evaluate ASR effect on the CEB whole building response (number 4).

c. Finding Related to Operability Determinations and Functionality Assessments - Inadequate Operability Determinations

Introduction. The inspector identified a finding in that NextEra failed to fully evaluate potential structural and seismic response impacts in accordance with the requirements in NextEra Procedure EN-AA-1001 after identifying degraded and nonconforming condition related to reduced concrete modulus of elasticity due to ASR degradation for safety related structures. The evaluation did not consider the following effects due to changed properties of concrete as measured directly from building concrete core samples: building natural frequency in the dynamic response; performance of anchorages and embedment of systems and components attached to the structures; and shear strength or capacity of affected structures and the dynamic/flexural response especially for those building walls without corresponding shear reinforcement.

Description. NextEra analysis of concrete cores samples taken following the April 2011 determination that certain below grade concrete walls in safety related structures were affected by ASR, indicated a reduced modulus of elasticity and compressive strength. Although the compressive strength reduction was viewed by NextEra as slight and acceptable, the lowest measured modulus was about 40 percent less than the design value of 3,620 kpsi.

NextEra completed operability determinations for certain affected safety-related concrete structures as required by NextEra Procedure EN-AA-203-1001, "Operability Determinations/Functional Assessments." In accordance with the Procedure EN-AA-203-1001, an operability determination must include: identification of current licensing basis functions and performance requirements as listed in the UFSAR; identification of the minimum design basis values necessary to satisfy the structure, system, or component (SSC) design basis safety functions; and evaluation of the effects of the degraded condition on the ability of the SSCs to meet its specified function and performance requirements.

During the week of September 26, 2011, NRC staff determined that the completed operability determinations were not sufficient in that they did not address the impact of the degraded condition on key aspects of the structure design as described in UFSAR. Specifically, NextEra failed to address the ASR induced effects of the reduced modulus of elasticity on seismic dynamic and flexural response in the following areas:

- Building natural frequency in the dynamic response;
- Performance of anchorages and embedment of systems and components attached to the structures affected by ASR; and

- Shear capacity of affected walls especially for those buildings without corresponding shear reinforcement such as for the CB and the emergency diesel generator building.

NextEra performed additional reviews and updated the operability determinations for the affected areas in response to these concerns, on October 14, 2011. The licensee determined that the structures and other affected systems and components remained functional for design basis conditions but were degraded.

The NRC reviewed the updated operability determinations and associated calculations determining that the additional areas needing evaluation were addressed and that the structures remained "operable but degraded." The previous determination indicated that the evaluated structures were "operable." Specifically, NextEra used quantitative and qualitative information with respect to the degraded concrete conditions as noted below.

With respect to dynamic response and the change in the natural frequency of the structures, licensee's additional evaluation determined that the shift in natural frequency was minimal and remained well above the ground response peak frequency range such that the response of the structures remained rigid. With respect to the ability of the equipment anchors and embedment to perform their function, the licensee's additional evaluation noted that there was no appreciable impact. The licensee also determined that the impact on the flexural capacity of seismic buildings with respect to shear stress was minimal, and the resultant stresses on the steel and concrete remained below the design stress limits with margin.

Following review, the inspector determined there was a reasonable expectation that the structural integrity remained intact under design loads, and the buildings remained operable but degraded. NextEra continued to review the degraded concrete issue within the corrective action program, including the effects on the long term reliability of the structures.

Analysis. The inspectors determined that NextEra's failure to conduct adequate prompt operability determinations per Procedure EN-AA-203-1001 for degraded and nonconforming conditions associated with ASR was a performance deficiency relative to a self imposed standard. Specifically, the operability determinations conducted following identification of ASR in safety-related structures did not completely analyze the effects of the reduced modulus on the dynamic and flexural response of safety related structures to seismic events along with the effect on attached systems and components. This performance deficiency was associated with the design control aspect of the Mitigating Systems cornerstone; and, based on a comparison to Example 3.i of Appendix E of IMC 0612, it was determined to be more than minor. The issue was evaluated using IMC 0609, "Significance Determination Process," and was determined to be of very low safety significance (Green). The finding had a cross cutting aspect in the area of problem identification and resolution, P.1(c), related to ensuring that issues potentially impacting nuclear safety are thoroughly evaluated. NextEra did not thoroughly evaluate conditions adverse to quality, including evaluating the effects of the reduced concrete modulus for impact on operability of the affected structures.

Enforcement. Because this finding does not involve a violation and has very low safety significance, it is identified as **FIN 05000443/2011-10-02, Incomplete Operability Determination for Degraded Concrete Structures Housing Safety-Related Equipment**.

.2 (Closed) Unresolved Item 05000443/2011003-02, 50.59 Evaluation for Accepting Reduced Modulus of Elasticity in Certain Safety-Related Structures Affected by Alkali-Silica Reaction

a. Inspection Scope

As part of the review of this unresolved item, the inspectors continued to review EC272057, dated April 25, 2011, for adequacy in which the engineering change (EC) was a design change to address reduced concrete modulus of elasticity in the CB electric tunnel and the containment enclosure building. The review was to determine if only a 10 CFR 50.59 screening was adequate to accept "as-is" conditions for this concrete material property. The inspector reviewed NextEra's revocation of this EC.

b. Observations

This issue was closed based on the revocation of the EC, and on the Severity Level IV NCV, as noted below.

c. Finding Related to Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications - Inadequate 50.59 Screen Evaluation for EC272057

Introduction. The inspectors identified a Severity Level IV NCV of 10 CFR 50.59(d)(1), "Changes, Tests, and Experiments," because NextEra did not adequately evaluate a "use-as-is" determination for the ASR impacted Category I concrete structures. Specifically, NextEra did not complete a 10 CFR 50.59 evaluation, to ensure that the identified reduction in concrete modulus of elasticity did not present a more than minimal increase in the likelihood of the occurrence of a malfunction of a SSC important to safety previously evaluated in the updated safety analysis report (USAR) prior to implementing changes to the facility as described in the engineering change EC272057 issued on April 25, 2011.

Description. On April 25, 2011, NextEra issued EC272057, "Concrete Modulus of Elasticity Evaluation," to address the reduced concrete modulus in the CB, the "B" electric tunnel, the containment enclosure building, the diesel generator fuel oil tank rooms, the residual heat removal equipment vaults, and emergency feedwater pump house. EC272057 dispositioned the degraded condition as "use-as-is" and incorporated the degraded condition into the design basis. In a safety evaluation screen for EC272057, NextEra concluded the change did not require a complete evaluation per 10 CFR 50.59(c)(2) because adequate design margin existed and there was no adverse affect on an UFSAR described design function.

10 CFR 50.59 requires licensees to evaluate whether NRC approval is required for proposed changes to the facility. The Seabrook 5059 Resource Manual defines the process for completing 10 CFR 50.59 evaluations for changes, tests, and experiments completed at Seabrook. It includes a screening process that defines criteria used to determine whether a full 10 CFR 50.59 evaluation must be performed for each applicable change, test, or experiment. NextEra screened EC272057 in accordance with the guidance in the 5059 Resource Manual and concluded that the change did not require a full evaluation per 10 CFR 50.59(c)(2) because adequate design margin existed and there were no adverse affects on the UFSAR described design functions.

The inspectors reviewed EC272057 and determined that NextEra's 50.59 Screen for EC272057 did not correctly address "adverse affects" as described in Section 5.2.2 of the 5059 Resource Manual. The concrete modulus of elasticity is a design value specified in both the Seabrook UFSAR and the ACI 318 – 1971 Building Code for the applicable plant structures. The inspectors determined that the reduced modulus of elasticity caused by the ASR could have had an "adverse affect" on the flexural and dynamic response of the impacted structures and, as such, required further evaluation per 10 CFR 50.59(c)(2 (ii) and (iv). The criterion c(2)(ii) and (iv) deal with the change resulting in more than minimal increase in the likelihood of occurrence or in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR. In response to the inspectors' concerns regarding the adequacy of the 10 CFR 50.59 evaluation, NextEra rescinded the design change EC272057 from the design basis on September 22, 2011, and initiated additional evaluations of the ASR affected structures.

NextEra personnel did not complete the 10 CFR 50.59 screen properly because they misunderstood the guidance in the 50.59 Resource Manual regarding the need to screen in changes in design parameters which impact the design function acceptance criteria (Resource Manual Section 5.2.2).

Analysis. The inspectors determined that the failure to evaluate changes to the facility as described in EC272057 was contrary to 10 CFR 50.59(d)(1) and was a performance deficiency warranting a significance evaluation in accordance with the NRC Enforcement Manual for Traditional Enforcement and IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening." The violation was determined to be more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," because the inspector could not reasonably determine that the changes would not have ultimately required prior NRC approval.

Violations of 10 CFR 50.59 are dispositioned using the traditional enforcement process instead of the SDP because they are considered to be violations that could potentially impede or impact the regulatory process. However, if possible, the underlying technical issue is evaluated under the SDP to determine the severity of the violation. In this case, for Mitigating Systems, the inspector determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings." The issue was determined to be of very low safety significance (Green) because it was a design or qualification deficiency confirmed not to result in an actual loss of safety function, because further evaluation determined that the structures remained operable despite the degraded modulus condition. In accordance with Section 6.1.d.2 of the NRC Enforcement Policy, this violation is categorized as Severity Level IV because the resulting changes were evaluated by the SDP as having very low safety significance (Green). Upon removal of EC272057 from the design basis on September 22, 2011, the issue no longer required an evaluation per 10 CFR 50.59(a)(2).

The finding had a cross cutting aspect in the area of human performance – work practices, H.4(b), because NextEra personnel did not follow procedures. Specifically, NextEra personnel did not address "adverse effects" as required by Section 5.2.2 of the 50.59 Resource Manual when preparing the 10 CFR 50.59 screen for EC272057.

Enforcement. Title 10 CFR 50.59, "Changes, Tests, and Experiments," Section (d)(1) states, in part, that the licensee shall maintain records of changes in the facility or procedures, and that the records must include a written evaluation that provides the bases for the determination that the change does not require a license amendment pursuant to paragraph 10 CFR 50.59(c)(2). Contrary to the above, from April 25 to September 22, 2011, NextEra did not provide an evaluation that adequately documented why the reduced concrete modulus of elasticity in Category I structures did not present a more than minimal increase in the likelihood of occurrence of a malfunction of a SSC important to safety previously evaluated in the USAR. Because this failure to properly evaluate a proposed change is of very low safety significance and has been entered into the licensee's Corrective Action Program (CR1647722), this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy.
(NCV 05000443/2011010-03, Failure to Properly Complete a 50.59 Screen).

4OA6 **Meetings, Including Exit**

On September 30 and December 2, 2011, the inspectors presented the interim results of this inspection to Mr. P. Freeman, Site Vice President, and Seabrook Station staff. The inspectors also confirmed with NextEra that no proprietary information was retained by inspectors during the course of the inspection.

On January 20, 2012, a final exit meeting was conducted and led by Mr. Richard J. Conte, Chief Engineering Branch No. 1. Others involved in this conference are noted on the list of contacts. During the meeting, the NRC staff's final disposition of the unresolved items and new findings were summarized. Other comments and questions were communicated to NextEra management with respect to the ASR problem in safety related structures.

ATTACHMENT: SUPPLEMENTARY INFORMATION

SUPPLEMENTARY INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel
B. Brown, Supervisor, Civil Engineering
V. Brown, Senior Licensing Analyst
K. Browne, Plant General Manager
J. Esteves, Plant Engineering
P. Freeman, Site Vice President
P. Gurney, Reactor Engineering Supervisor
M. Collins, Manager, Design Engineering
M. O'Keefe, Licensing Manager

Key Participants for Teleconference of January 20, 2012

NextEra Attendees:
Paul Freeman, Site Vice President
Mike O'Keefe, Licensing Manager
Mike Collins, Design Engineering Manager
Rick Cliché, License Renewal Project Manager
Ted Vassallo, Design Engineering
Paul Willoughby, Licensing
Ken Chew, License Renewal
Al Griffith, Public Communications

NRC Staff:
Christopher Miller, Division of Reactor Safety, Region I
Richard Conte, Division of Reactor Safety, Region I
Suresh Chaudhary, Division of Reactor Safety, Region I
Art Burritt, Division of Reactor Projects, Region I
Bill Raymond, Division of Reactor Projects, Region I
John Lamb, Division of Operating Reactor Licensing, NRR
Abdul Sheikh, Division of License Renewal, NRR
George Thomas, Division of Engineering, NRR
Raj Auluck, Division of License Renewal, NRR

LIST OF ITEMS OPENED, CLOSED, DISCUSSED, AND UPDATED

Opened/Closed:

05000443/2011-010-01	URI	Adequacy of Corrective Actions Associated with Calculation Methods for Alkali-Silica Reaction Issue
05000443/2011-010-02	FIN	Inadequate Operability Determination for Degraded Concrete Structures Housing Safety-Related Equipment
05000443/2011-010-03	NCV	Failure to Properly Complete a 50.59 Screen for EC272057

Closed:

05000443/2011-003-02	URI	Review of 50.59 screening to accept-as-is reduced values for concrete properties in safety related structures.

Updated:

05000443/2011-003-03	URI	Prompt Operability Determination for Safety Related Structures affected by ASR.

Definitions
(American Concrete Institute (ACI) Terminology)

Poisson's ratio, v: The ratio of transverse strain (perpendicular to the applied load) to the axial strain (in the direction of the applied load).

Modulus of Elasticity, E: The ratio of the normal stress to the corresponding strain for tensile or compressive stress below the proportional limit of the material.

Shear Modulus or Modulus of Rigidity, G: The ratio of unit shearing stress to the corresponding unit shearing strain.

Bulk Modulus, K: The ratio of the change in average stress to the change in unit volume.

Note: The above parameters are four simplified elastic constants defining a material exhibiting "elastic" behavior.

Other Definitions:

Stress: The force per unit area (compressive/tensile, transverse or shear).

Strain: In a given direction (transverse or axial) is the change in dimension under load to the original dimension in the direction under consideration.

Compressive Strength: Capacity of a material or structure to withstand axial pushing forces. When the limit of compressive strength is reached, materials fail.

Tensile Strength: Capacity of a material or structure to withstand axial pulling forces. When the limit of tensile strength is reached, materials fail.

Shear Strength: Capacity of a material or structure to withstand forces parallel to a surface area that could cause sliding failure of the material. When the limit of shear strength is reached, materials fail.

Bond Strength: The resistance to separation of mortar and concrete from reinforcing and other materials with which it is in contact.

LIST OF DOCUMENTS REVIEWED

Prompt Operability Determination (POD) AR 581434, Reduced Concrete Modulus of Elasticity below Grade in 'B' Electrical Tunnel Exterior Walls, Revision 0, June 27, 2011, and Revision 1, October 14, 2011

POD AR 1664399, Reduced Concrete Modulus of Elasticity Below Grade in Containment Enclosure Building, RHR Equipment Vaults, EFW Pump House, and Diesel Generator Fuel Oil Tank Rooms, Revision 0, June 27, 2011, and Revision 1, October 14, 2011

Calculation C-S-1-10163, Rev. 0, Fundamental Frequency of ASR Effected Walls, October 14, 2011

Calculation C-S-1-10159, Rev. 0, 'B' Electrical Tunnel Transverse Shear Evaluation Supplement to Calculation CD-20

Calculation C-S-1-10150, Rev. 0, Effects of Reduced Modulus of Elasticity – 'B' Electrical Tunnel Exterior Walls

Calculation CD-20-CALC, UE Control and Diesel Generator Building Design of Material and Walls below Grade for Electrical Tunnel and the Control Building (Original Design Calculation)

Drawings for Control Building Concrete (Electrical Tunnel) 9763-F-111342, 9763-F-111343 and 9763-F-111345

EC 145305, Condition Assessment of Control Building Concrete

AR1641413, Evaluation of Containment with Craze Cracking in Concrete, April 20, 2011

AR1644074, Concrete Test Results for Containment Enclosure Building, April 21, 2011

AR 574120, Preliminary Test Results of Control Building Concrete

AR 581434 Test Results from Control Building Concrete Modulus Testing (Results of petrographic analysis of four of the 12 CB cores identified the presence of moderate to severe ASR in the concrete)

EC250348, Revision 002, Condition Assessment of Building Concrete

AR 01625775, Revision 000, Petrographic Analysis of Concrete Cores from Seabrook Station

System Description No. SD-66, Revision 2, System Description for Structural Design Criteria for Public Service Company of New Hampshire, Seabrook Station, Unit Nos. 1 and 2, 3/02/84.

Seabrook UFSAR, Revision 12, Section 3.8.4, Other Seismic Category 1 Structures

Letter dated 6-29-2011 from Richard Plasse, USNRC, to Mr. Paul Freeman, NextEra Energy Seabrook, LLC – Request for Additional Information for the Review of Seabrook Station License Renewal Application (Specifically Follow-up to RAI B2.1.31-1 on pages 2-3) (ML11178A3380)

NextEra Energy Letter SBK-L-11154 to USNRC dated 8-11-2011, Docket No. 50-443, Seabrook Station Response to Request for Additional Information – NextEra Energy Seabrook License

NextEra Energy Letter SBK-L-11063 to USNRC dated 4-14-2011, Docket No. 50-443, Seabrook Station Response to Request for Additional Information – NextEra Energy Seabrook License Renewal Application Request for Additional Information – Set 13 (Specifically Responses to Follow-up to RAI B.2.1.31-1 and -2 on pages 4-7) (ML11108A1310)

NextEra Energy Letter SBK-L-10204 to USNRC dated 12-17-2010, Docket No. 50-443, Seabrook Station Response to Request for Additional Information – NextEra Energy Seabrook License Renewal Application Aging Management Programs (Specifically Responses to RAI B.2.1.31-1, -2 and -3 on pages 36-39) (ML1035405340)

LIST OF ACRONYMS

AR	Action Request
ACI	American Concrete Institute
ASR	Alkali-Silica Reaction
ASME	American Society of Mechanical Engineers
CB	Control Building
CEB	Containment Enclosure Building
CFR	Code of Federal Regulations
CR	Corrective Action
DRS	Division of Reactor Safety
EC	Engineering Change
EN	Procedural Notice for Engineering Department
FIN	Finding
IMC	Inspection Manual Chapter
IP	Inspection Procedure
KSI	Kilo-pounds per square inch
LRA	License Renewal Application
NCV	Non-Cited Violation
NRC	U.S. Nuclear Regulatory Commission

NRR	Office of Nuclear Reactor Regulation
OD	Operability Determination
POD	Prompt Operability Determination
psi	Pounds per square inch (absolute)
PSIG	Pounds per square inch (gage)
RCA	Radiological Controlled Area
SDP	Significance Determination Process
SR	Safety Related
SSC	Structure, System, or Component
TIA	Task Interface Agreement
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
USAR	Updated Safety Analysis Report