Inspector and Tech Reviewer Areas to focus on:

1. **Cover letter messages and request**
2. **Executive Summary (ES) for appropriateness and need**
3. **Length of Open URI section 4OA5.2**
4. **Summary of POD assumptions issues from TIA in ES and 4OA5**
5. **What to address in plans is only in cover letter (management integration of inspector observations to TIA)**
6. **No immediate safety concerns in ES only.**
7. **List of ACRONYMS and Reference list ???????**

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

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 on January 20th 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. The focus of this inspection was activities surrounding your development actions related to the Alkali-Silica Reaction (ASR) problem occurring in safety related structures and other structures of regulatory importance (covered by the maintenance rule). In particular, we reviewed your Prompt Operability Determinations for certain structures based on best available information. At the beginning of the inspection report period, we noted some areas that still needed to be addressed based on available information and NextEra satisfactorily addressed them with revisions to the documents.

On January 20, 2011 a final exit meeting was conducted and lead by Mr. Richard J. Conte, Chief Engineering Branch No. 1 of my staff. During the meeting, my staff summarized the change in status of the new findings and our plans to issue a Task Interface Agreement between Region I and the Office of Nuclear Reactor Regulation simultaneously with this report. The TIA was placed in the public document room **(ADAMS Accession No. MLXXXXXXX)**. The purpose of the Task Interface Agreement was for the NRR staff to identify the review criteria in evaluating the operability determination for the "B" Electrical Tunnel affected by ASR (part of the Control Building) in assistance to the Region I staff by addressing questions we had on the matter.

Also on January 20th, we focused on and summarized observations on your plans with respect to the unwritten assumptions in your operability determinations. The NRC staff noted that these

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determinations listed no assumptions in the applicable sections and that the design basis code ACI 318-1971 was based on empirical data for determining certain parameters that were a part of the design bases. Also Poison's ratio on concrete cores were not being tested or evaluated and this ratio was in the UFSAR. The assumption of this empirical data was that the relationships were for ASR free concrete. Specific areas for which your plans do not address unwritten assumptions being made in the prompt operability determinations were list in section 4OA5.2

In consultation with our technical reviewers in headquarters and to address the current shortcomings on unwritten assumptions for your operability determinations, we have determine that your plans do not sufficiently provide information related to: 1) condition assessment (extent and characterization); 2) cause of the ASR as it impacts current degradation and operability; 3) estimate of expansion to date and current expansion rate; 4) interim structural assessment as it impacts current operability vs. longer term structural assessment; and, longer term monitoring ensure operability in the near future vs. longer term of the duration of the license (1-2 years vs. longer); and, 5) short term mitigation or needed remedial actions. This is in distinction to your overall comprehensive plan for the problem.

Accordingly, we request that you provide your plans to address the above issues within 30 days of the date of this inspection report. We noted that, from the exit meeting of January 20th you have agreed to this request and to review the report in 15 days and let us know of your plans to honor our request or identify the need for a management meeting. We further request that, should a management meeting be needed on these issues, it should be conducted within 30 days of the date of this report and a final response time will be negotiated at the management meeting. If your root cause evaluation scheduled for Feb. 2012 and the associate corrective action plan for this significant condition adverse to quality addresses the above, please use them to respond to our request.

Also, the report documents two NRC-identified findings of very low significance (Green) that were determined to involve 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 Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States 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 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 (ADAMS). 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. 05000443/201110
w/Attachment: Supplemental Information

cc w/encl: Distribution via ListServ

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

Sincerely,

Christopher Miller, Director
Division of Reactor Safety

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

Enclosure:
Inspection Report No. 05000443/201110
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	50-443
License No.:	NPF-86
Report No.:	05000443/2011010
Licensee:	NextEra Energy Seabrook, LLC
Facility:	Seabrook Station
Location:	Seabrook, NH 03874
Dates:	September 25 – September 30, November 15-17, 2011 (Illinois) November 28-29, 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, NRR G. Thomas, Structural Engineer, NRR
Approved by:	Richard J. Conte, Chief Engineering Branch 1 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000443/2011010; 9/25/2011 – 12/2/2011; Seabrook Station (IP 7111115 and IP7111117).

This report covers an inspection by regional inspectors, and resident staff, with assistance from 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". The cross-cutting aspects for the findings were determined using IMC 0310, "Components Within Cross-Cutting Areas." Findings for which the Significance Determination Process 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 regarding NextEra's operability determinations for Category I structures with a reduced concrete modulus of elasticity caused by alkali-silica reaction in the concrete. NextEra did not adequately evaluate with available information the effects of the reduced concrete modulus with respect to key aspects of structural design as described in the Updated Final Safety Analysis Report (UFSAR). Specifically, NextEra did not initially fully evaluate the effects of the reduced modulus on the dynamic response of Category I structures to **seismic events relative to global response**; the changes in the structural natural frequency; and, the effects on attached systems, components and anchors. **Further, NextEra did not adequately evaluate the adequacy of shear capacity of electric tunnel concrete walls without shear reinforcement to resist lateral forces during seismic events.**

The failure to fully evaluate the degraded and nonconforming concrete modulus condition as required by procedure EN-AA-203-1001 was a performance deficiency. The performance deficiency was associated with the Mitigating Systems cornerstone and was determined to be more than minor based on a comparison with Appendix E.3.i of IMC 0612 because it adversely affected the cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events in order to prevent core damage. The issue was evaluated using IMC 0609, "Significance Determination Process" (SDP), 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. Specifically, 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. (Section 1R15)

Severity Level IV. The inspectors identified a non-cited violation of 10 CFR 50.59(d)(1) because NextEra did not provide an evaluation that adequately documented why implementing a design change to address an identified reduction in the concrete modulus of elasticity for several Category I concrete structures, 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 UFSAR. Specifically, NextEra issued EC272057 on April 25, 2011, to address reduced concrete modulus of elasticity in the control building electric tunnel and the

containment enclosure building, but did not complete a 10 CFR 50.59 evaluation prior to implementing changes to the facility as described in the modification.

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 Inspection Manual Chapter (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. The finding was evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," and determined to be potentially risk significant due to a design deficiency confirmed not to result in a loss of operability. 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). 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 5059 screen for EC272057. (Section 1R17)

Executive Summary

The focus of this inspection was activities surrounding development actions related to the Alkali-Silica Reaction (ASR) problem occurring in safety related structures and other structures of regulatory importance (covered by the maintenance rule). In particular, NRC staff reviewed the Prompt Operability Determinations for certain structures based on best available information. At the beginning of the inspection report period, we noted some areas that still needed to be addressed based on available information and these issues were satisfactorily addressed with revisions to the documents.

Prior NRC review of this area was documented in the following Inspection Reports 05000443/2010004, 2010005, 2011002, 2011003, and 2011007 and the results were summarized in the Background section of this report.

One unresolved items was closed and sufficient information was obtained in order to determine the performance deficiency – 50.59 screening to accept as-is conditions for reduced modulus of elasticity for certain safety related structures. Another unresolved item was left open – Prompt Operability Determinations for certain safety related structure with the ASR problem. Another finding of very low safety significance was determined in the course of the operability determination review. Both findings were summarized above.

The inspectors observed that NextEra plans for the ASR problem were not addressing unwritten assumptions in the operability determinations. These operability determinations listed no assumptions in the applicable sections. The design basis code ACI 318-1971 was based on empirical data for determining certain parameters that were a part of the design bases. Poisson's ratio on concrete cores were not being tested or evaluated and this ratio was in the UFSAR. The assumption of this empirical data was that the relationships were for ASR free concrete. Specific areas for which the plans do not address unwritten assumptions being made in the prompt operability determinations were list in section 4OA5.2

To address the above and in conjunction with the NRR technical reviewers, the inspectors noted that the current plans were not finalized and what was existing did not specifically address the unwritten assumptions in the following areas: 1) condition assessment (extent and characterization); 2) cause of the ASR as it impacts current degradation and operability; 3) estimate of expansion to date and current expansion rate; 4) interim structural assessment as it impacts current operability vs. longer term structural assessment; and, longer term monitoring ensure operability in the near future vs. longer term of the duration of the license (1-2 years vs. longer); and, 5) short term mitigation or needed remedial actions.

The NRC staff summarized why there is no immediate safety concerns for the existing conditions: walkdowns confirm no significant degradation, no visual evidence of distortion nor visible evidence of rebar corrosion; overall evidence of sufficient stiffness remaining; No appreciable evidence of cracking where found, in isolated sections of the wall; degradation appears to be localized; We know from best available research that the ASR rate slowly progresses and there is some evidence that it has progressed to a plateau but it needs to be confirmed by testing; parameters obtained for compressive strength and modulus of elasticity indicate robust design to strength of the concrete poured (4K psi concrete used in buildings only needing 3K psi concrete and no significant negative shift on seismic analysis.

On January 20, 2011 a final exit meeting was conducted. During the meeting, NRC staff summarized the change in status of the new findings and plans to issue a Task Interface Agreement between Region I and the Office of Nuclear Reactor Regulation simultaneously with this report. The TIA was placed in the public document room **(ADAMS Accession No. MLXXXXXXX)**. The purpose of the Task Interface Agreement was for the NRR staff to identify the review criteria in evaluating the operability determination for the "B" Electrical Tunnel affected by ASTR (part of the Control Building) in assistance to the Region I staff by addressing questions related to the problem.

REPORT DETAILS

Background

In June 2009, NextEra conducted walkdowns 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 License Renewal Audit results noted the alkali-silica reaction (ASR) problem and pointed to need for a good number of requests for additional information in this area since the issue was newly discovered for the site (noted as a area to address in the GALL, Generic Aging Lessons Learned, Revision 1). In the Fall of 2010, NextEra performed an Immediate and Prompt Operability Review (POD) based on core samples taken in Control Building in August 2010. In November 2010, Inspection Report 05000443/2010004 and, in February 2011, Inspection Report 2010005, followed developments in this area from an operability viewpoint. In these reports, no findings were noted since laboratory results determined compressive strength and modulus of elasticity met UFSAR values (degradation into reserve strength not design margin). In May 2011, Inspection Report 05000443/2011002 identified two noncited violations of very low safety significance in the structures monitoring area with respect to the maintenance rule (10 CFR 50.65 a(1) and b(2)). Also in May 2011, License Renewal Inspection (IP71002) Report 05000443/2011007 had an overall result: "Except for Structures Monitoring Program, results support a reasonable assurance determination for license renewal."

As the year progressed, NextEra continued to identify and characterize the below-grade structures at Seabrook having experienced groundwater infiltration and a resultant reduction in concrete material properties. NextEra determined the degraded concrete condition was most likely due to distress from ASR in the concrete. ASR is a chemical reaction in concrete over time between the alkaline cement paste and reactive non-crystalline silica which is found in common coarse aggregates. The reaction only occurs in the presence of water and forms a gel that expands, forming micro-cracks that change the strength of concrete system. NextEra is in the discovery phase of condition assessment along with extent of condition reviews in order to support other building operability determinations and a plan to conduct an engineering evaluation which would appear to constitute a long term operability review (40 year license term) along with mitigation and monitoring measures.

The appearance of ASR degradation in safety related concrete structures was the first noted in the nuclear industry in the United States and could be significant related to preservation of reserve capacity with building design loads as reflected in the current licensing basis during normal operations and aging management over the period of extended plant operation. While the problem was noted as an aging effect in a license renewal topical report, the appearance of ASR reflected a newly discovered aging effect at Seabrook that needs to be managed for license renewal.

In August 2011, Inspection Report 05000443/2011003 addressed a noncited violation of very low safety significance related to the untimely Initial and Prompt ODs for results on extent of condition review for other buildings affected by ASR. Two unresolved items was also opened, one dealing with a potential inadequate screen in accordance with 10 CFR 50.59 for accepting the reduce parameter found on compressive strength and modulus of elasticity for the "B" Electrical Tunnel and the Containment Enclosure Building. The other unresolved dealt with the open prompt operability determinations associated with the "B" Electrical Tunnel and the

building subjected to an extent of conditions review. In August of 2011 a Task Interface Agreement (TIA) was issued between Region I and the Office of Nuclear Reactor Regulation on the ASR issue. The response to the TIA is publicly available (ADAMS Accession No. MLxxxxxxxxx). The purpose of this inspection was to followup on the unresolved items and to work with the NRR technical reviewers in developing the answers to the questions posed in the TIA.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R15 Operability Determinations and Functionality Assessments (71111.15 – 0 samples)

a. Inspection Scope

As a part of the review of an unresolved item (see section 4OA5.2), the inspectors, in conjunction with technical reviewers from the Office of Nuclear Reactor Regulation, reviewed the adequacy of operability determinations for the below grade concrete walls of seismic Category I buildings affected by alkali-silica reaction. The review focused on the specific design for the buildings affected by an alkali silica reaction and operability was addressed by two prompt operability determinations, one, to address the control building and the other to address the following extent of conditions with five other building/structures: Containment Enclosure Building, Emergency Diesel Generator Buildings (fuel oil rooms), Residual Heat Removal Equipment Vaults, Emergency Feedwater Pump House, and the Radiological Control Access (RCA) tunnel.

b. Findings

Inadequate Operability Determinations

Introduction. In April 2011, NextEra identified a degraded and nonconforming condition related to reduced modulus of elasticity for buildings housing safety related equipment, but did not thoroughly evaluate potential impacts in accordance with the requirements in NextEra procedure EN-AA-203. Specifically, the evaluation did not consider the affect of the reduced modulus on: overall stiffness and resulting seismic response, concrete material property changes affecting natural frequency and therefore the amplitude of the seismic response, and the performance of systems and components attached to the affected structures during seismic events.

Description. In April 2011 NextEra determined that certain below grade concrete walls were affected by alkali-silica reaction (ASR). The analysis of concrete cores showed a reduced concrete modulus of elasticity in the control building / electric tunnel (AR581434), containment enclosure building (AR1644074) and three other seismic Category I buildings (AR1664399). The lowest measured modulus was about 40% less than the design value of 3.62E+03 ksi.

NextEra completed prompt operability determinations for the affected Category I concrete structures (reference ARs 581434, 1644074 and 1664399) as required by NextEra Procedure EN-AA-203-1001, "Operability Determinations/Functional

Assessments." In accordance with the procedure EN-AA-203-1001, a POD 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 SSC design basis safety functions; and evaluation of the effects of the degraded condition on the ability of the SSC to meet its specified function and performance requirements.

During the week of September 28, 2011, the inspectors in conjunction with technical reviewers from the Office of Nuclear Reactor Regulation reviewed NextEra's competed PODs for the ASR-affected Category I concrete structures. It was determined that the evaluations were not complete with respect to available information since NextEra did not evaluate the degraded condition with respect to key aspects of the structure design as described in UFSAR. Specifically, the initial PODs did not adequately address the effects of the reduced modulus of elasticity in the following areas:

1. As a result of further review of the design bases, the NRC staff determined that the walls below grade in the Control Building 'B' Electrical Tunnel do not contain shear reinforcement to resist dynamic lateral forces acting on the wall during a design basis earthquake. The design of the wall intentionally depends on the strength of the concrete alone to resist these dynamic forces and this information was not evaluated for the degraded conditions. The same was true for the Diesel Generator Building. The modulus of elasticity of concrete was a function of concrete compressive strength which is generally higher in the as-cast condition than assumed in the design. The concrete used in construction of Seabrook structures was formulated to have a design strength greater than 3000 psi. However, as stated in the Seabrook Updated Final Safety Analysis Report, Revision 12, Section 3.8, "While variability in concrete modulus has no significant effect on structural design, it influences structural stiffness and natural frequency, and, subsequently, the amplified response spectra of the seismic analysis."

2. Changes in the modulus of elasticity affect the material concrete properties and, therefore, the natural frequency of the structure, which affects how the buildings are analyzed in the seismic analysis. NRC reviews determined that the prompt operability determinations addressed the dynamic response of the structures in a qualitative manner noting that the ASR impacted walls are below grade and the structural loadings would be governed by the ground response spectra assumed in the original design. The initial evaluations did not sufficiently address the impact of the reduced modulus on the structure natural frequencies in a quantitative manner to validate that the structure response would remain rigid or that there would be a amplification of the response. Specifically, the initial evaluation did not verify there would be no amplification of the motions beyond those in a ground response spectra as assumed in the seismic analysis per UFSAR 3.7(B).2.

3. The initial POD addressed the effects of the reduced modulus on components housed within the structures, such as pipe supports, cable trays and component support anchors. NRC review determined that the initial evaluations addressed the impacts on the internal components in a qualitative manner, but did not verify the equipment performance would remain bounded by the analysis in the original design as described in UFSAR 3.7 and 3.8. Specifically, the initial evaluation did not verify

there would be no amplification of the motions beyond those in a ground response spectra as assumed in the seismic analysis per UFSAR 3.7.(B).2. Further, the initial evaluation did not evaluate the potential impacts on anchor or wall shear capacities caused by ASR induced changes in material properties beyond that allowed for as described in UFSAR 3.8.

In response to the NRC-identified issues, NextEra completed additional evaluations that determined the structures and other affected systems and components remained functional for design basis conditions. On October 14, 2011, NextEra completed Calculation C-S-1-10163, and revisions to the PODs for AR581434 (CB/ ET) and AR1664399 (CEB and other Category I Structures). The NRC determined that NextEra's additional analysis and revisions to the PODs adequately addressed the concerns discussed above. Specifically, the analysis confirmed a minor impact on the overall response of the structure during a seismic event, a small effect on the structure's natural frequencies that results in no appreciable amplification of the ground response during a seismic event, and no impact on the ability of the equipment anchors to perform their function due to the quality of the concrete and construction methods used.

Analysis. The inspectors determined that not following a self imposed standard, not completely analyzing the effects of the reduced modulus of elasticity on Category I structures based on available information, per procedure EN-AA-203-1001 was a performance deficiency. Specifically, because the reduced modulus affected the dynamic response of Category I structures to seismic events relative to global response, changes in natural frequency and the effects on attached systems and components, procedure EN-AA-203-1001 required that these impacts be evaluated as part of the prompt operability determination. This performance deficiency was associated with the Mitigating Systems cornerstone and was determined to be more than minor because, based on a comparison with Examples 3.i of Appendix E of IMC 0612, it adversely affected the cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events in order to prevent core damage. Specifically, the effects of the reduced modulus of elasticity on the dynamic response of Category I structures to seismic loading required further evaluation to demonstrate the structures and enclosed systems remained functional as described in the licensing and design bases. The issue was evaluated using IMC 0609, "Significance Determination Process" (SDP), and was determined to be of very low safety significance (Green). Specifically, when evaluated under IMC 0609, Attachment 4, the performance deficiency was not a design or qualification deficiency resulting in an actual loss of safety function, was not a loss of a barrier function, and was not potentially risk significant for external events. 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 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 05000443/2011-10-01, Incomplete Operability Determination for Degraded Concrete Structures Housing Safety-Related Equipment based on available information.

1R17 Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications (71111.17 – 0 sample)

a. Inspection Scope

As a part of the review of an unresolved item (see section 4OA5.1), the inspector reviewed EC272057, dated April 25, 2011, for adequacy in which the EC was a design change to address reduced concrete modulus of elasticity in the control building electric tunnel and the containment enclosure building. The review was to determine if only a 50.59 screening was acceptable to accept "as-is" conditions for this concrete material property which was degraded from the design bases as reflected in the UFSAR apparently due to the ASR problem.

b. Finding

Inadequate 50.59 Screen Evaluation for EC272057

Introduction: A Severity Level IV NCV of 10 CFR 50.59(d)(1), "Changes, Tests, and Experiments," was identified because NextEra did not provide an evaluation that adequately documented why implementing a design change to address an identified reduction in the concrete modulus of elasticity for several Category I concrete structures, 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). Specifically, NextEra issued EC272057 on April 25, 2011, to address reduced concrete modulus of elasticity in the control building electric tunnel and the containment enclosure building, but did not complete a 10 CFR 50.59 evaluation prior to implementing changes to the facility as described in the modification.

Description: NextEra determined that certain below grade concrete walls were affected by alkali-silica reaction (ASR). The analysis of concrete cores taken from ASR affected areas, showed a reduced concrete modulus of elasticity in the control building / electric tunnel (AR581434), containment enclosure building (AR1644074) and four other seismic Category I buildings (AR1664399). The lowest measured modulus was about 40% less than the design value of 3.62E+03 ksi.

On April 25, 2011, NextEra issued EC272057, "Concrete Modulus of Elasticity Evaluation," to address the reduced modulus. 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* 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 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 Building Code for the applicable plant structures. The reduced modulus of elasticity caused by the ASR occurring in impacted concrete walls has the potential to affect the flexural capacity and dynamic response of the impacted structures. Therefore, the inspectors determined that there was sufficient evidence that the reduction in the modulus of elasticity was caused by the ASR and it was an "adverse affect" as described in Section 5.2.2 of the 5059 Resource Manual and thus required further evaluation per 50.59(c)(2). The additional evaluation required by 10 CFR 50.59 d(1) was needed to at least ensure that 10 CFR 50.59 c(2) (ii) and (iv) criteria were not met and, therefore, there would be a need for a license amendment per 10 CFR 50.90. 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 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.

On October 14, 2011, NextEra issued additional information to support of its engineering evaluation of the ASR impacted structures, including Calculation C-S-1-10163, the Prompt Operability Determination for AR581434 (CB/ ET) Revision 1, and the Prompt Operability Determination for AR1664399 (CEB and other Category I Structures) Revision 1. The reduced modulus caused the concrete to have increased flexure, but the results of NextEra's additional evaluations confirmed that the reduction in capacity was minimal and the resultant stresses on the steel and concrete caused by the ASR degradation remained below the design stress limits with margin. Similarly, the affect of the reduced modulus also reduced the natural frequency of the structures, but the additional evaluation again 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. Therefore, although the effect of the ASR on the impacted walls was to reduce the design modulus parameter, the structural integrity

remained fully intact under all design loads, and the buildings remained operable. NextEra actions continue to review the degraded concrete issue within the corrective action program, including the effects on the long term reliability of the structures. See Section 4OA5 of this report for further NRC reviews of the revised operability determinations for ASR impacted structures.

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 Inspection Manual Chapter (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, 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," Tables 3b and 4a, for the Mitigating Systems Cornerstone. The inspector answered "Yes" to Question 5 under the Mitigating Systems Cornerstone column of the Phase 1 worksheet because the inspector concluded that the finding screened as potentially risk significant due to a design or qualification deficiency confirmed not to result in a loss of operability or functionality. 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). Further evaluation determined that the structures remained operable despite the degraded modulus condition. 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).

NextEra personnel did not complete the 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). 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 50.59 Resource Manual when preparing the 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 05000443/2011010-02, Failure to Properly Complete a 5059 Screen for EC272057).**

4. OTHER ACTIVITIES

4OA5 Other Activities

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

a. Inspection Scope

The inspector reviewed EC272057, dated April 25, 2011, for adequacy in which the EC addressed reduced concrete modulus of elasticity in the control building electric tunnel and the containment enclosure building. The review was to determine if only a 50.59 screening was acceptable to accept "as-is" conditions for this concrete material property which was degraded from the design bases as reflected in the UFSAR.

b. Observations/Findings

This issue was closed since the inspectors identified one Severity Level IV NCV, as described in Section 1R17 of this report.

.2 (Open) Unresolved Item 05000443/2011003-03, Open Operability Determinations for Safety-Related Structures Affected by ASR

a. Inspection Scope

The scope of this review was to update NextEra actions to date. The inspectors reviewed the prompt operability determination for the control building, and for the extent of conditions review for other buildings affected by the alkali silica reaction. The inspectors utilized site records and interviews to develop the design basis for the safety related structures in greater detail than summarized in section 3.8 of the Updated Final Safety Analysis Report. Additionally, this review was assessed progress in the development of a plan and schedule to address inspection activities, in-situ and laboratory testing to address the alkali-silica-reaction degradation with specific focus on the Control Building as a test case for review.

With respect to laboratory conditions 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. Particular care was 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: 1) caliper (model 500-505-10, serial #0014816) calibration document and calibration sticker on the caliper; and, 2) compression machine (model CM5000-D, serial #11005) calibration document and calibration sticker on the machine. With respect to test technician qualifications, the inspector also verified qualification records (Level 2 qualification for concrete testing up to date). The inspector also reviewed the Altran Commercial Grade Dedication Plan 10-0076 Part 05 – In Field Check List - the check list was in hand during the preparation and testing.

During the week of November 28, 2011, the inspector continued to review historical documentation from the construction phase of the plant, correlation 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 was discussed and reviewed with cognizant engineering and management personnel. 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 IWL inspection of the containment, and the results of the licensee's efforts in inspection and mapping of 'crazed' cracking in containment and enclosure building wall and the adequacy and validity of the documented results.

b. Observations/Findings

The inspectors identified one finding which was addressed in section 1R15 of this report. In summary of the finding, the operability determinations did not fully evaluate design information that was available on the potential effects of the reduced modulus of elasticity in the following areas: overall stiffness and resulting seismic response, concrete material property changes affecting natural frequency and therefore the amplitude of the seismic response, and the performance of systems and components attached to the affected structures during seismic events. Procedure EN-AA-203-1001 required that these impacts be evaluated as part of the prompt operability determination based on available information.

The unresolved item was kept open because the operabilitiy determinations have not been finalized. Several other observations were made in order to update this item. The reviewers noted that NextEra had engaged knowledgeable vendors, appropriate consultants, and recognized experts for testing, analysis, and evaluation of the effects of alkali-silica-reaction, on the serviceability and safety of the affected structures. Although the NextEra plan has not been refined before the NRC staff may be able to determine that if it meets the rigor expected of an Appendix B Quality Assurance program, a preliminary schedule of actions generally consistent with the proposed plans from the contractors was underdevelopment. It was noted that some of the elements of an aging management program had been developed and a final one was under development putting for putting in place, including monitoring and trending. Additional core samples were planned, the operability determination was to be updated as new information, analysis, and assessments became available on an as needed basis. The extent of condition was being comprehensively determined along with building initial assessments through the work of its consultants and contractors. NextEra was also developing and scheduling actions to determine where the alkali-silica-reaction was on the alkali

depletion curve (relates to extent of potential future degradation), although the tests may take up to two years to complete, and provide reliable data.

On October 14, 2011, NextEra issued additional information in support of its engineering evaluation of the alkali-silica-reaction impacted structures, including Calculation C-S-1-10163, the Prompt Operability Determination for AR581434 (for the Control Building and Electrical Tunnel), Revision 1, and the Prompt Operability Determination for AR1664399 (other Category I Structures), Revision 1. In C-S-1-10163, the fundamental frequency was evaluated using the measured modulus of elasticity determined in concrete core samples taken from the building walls. The calculation evaluated the impact of the reduced modulus (compared to the design value) on the wall stiffness with respect to the ground response spectra for the Seabrook site. The building response frequency was calculated using the principles and equations of engineering mechanics for a uniformly loaded fixed-fixed beam model (a simple span fixed at both ends during a seismic event). The effect of the reduced modulus was similarly evaluated to assess the impact on the natural frequency of the structures. The seismic analysis for Seabrook described in Updated Final Safety Analysis Report Section 3.7(B).2, was used in the design of Category I Structures, systems and components at Seabrook.

The inspectors completed a detailed review of C-S-1-10163 and verified that the calculation inputs were supported by plant data and the design references cited in the calculation. No inadequacies were identified. The results of C-S-1-10163 supported NextEra's conclusions in the revised prompt operability determinations, AR 581434 and AR1664399.

Also, during the week of November 14, 2011, a Region III inspector reviewed laboratory testing for compressive strength on fifteen concrete core samples taken from the control building in the October 2011 time frame. The testing was conducted at a laboratory in Northbrook, Illinois. The scope of this review was as noted above. For the testing the week of November 14, 2011, 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 during the cutting phase to be used by Seabrook for further petrography in the near future. Sample preparation (capping) was done in accordance with ASTM C617. Compression testing was done in accordance with ASTM C39. No concerns were noted with respect to quality control during all aspects of compression testing.

Other observations were made during the week of November 14, 2011. Multiple laboratory engineers, licensee engineer, and Altran engineer were involved in making call on fracture patterns. All but one of the obtained compressive strengths were fairly consistent with previous lab's results (2010, 2011 data). Core sample L5-C exhibited highest compressive strength of 6610 PSI whereas the previous lab's data identifies strength at 3950 PSI. This core compressive strength value was the only apparent outlier amongst the data set. All 15 destroyed cores are to be shipped back to Seabrook later today including the cut samples to be used for petrography.

During the week of November 28, 2011, the IWL Examination Report for the Primary Containment in October 2011, recorded information related to concrete conditions to ensure no unacceptable surface conditions for cracking (greater than 40 mils) and report

on other conditions such spalling and discoloration conditions. Related to this review the inspector noted in AR 01641413 an evaluation of crazed cracking on the exterior surface of the primary containment at azimuth 315^0 and elevation (-)30 feet, 00 inches. While several factors were identified by NextEra in support of structural integrity of this structure, it was noted the continued evaluation would be done in accordance with the extent of condition review per AR 574120 which identified the loss of concrete strength due to alkali-silica reaction (ASR) in other buildings noted herein. The inspector noted that no specific cause for the crazed cracking was identified which could be due ASR or other mechanisms. Further the inspector questioned the reliance on the April 2008 10 CFR 50 Appendix J, Type A Test at 49.6 psig that showed no evidence of cracking greater than 40 mils without the use of a before and after crack mapping effort. No unacceptable conditions were found and the extent of condition review noted above is a part of this open unresolved item.

During this inspection, it was determined that NextEra and its contractor were conducting a remodeling effort on the Containment Enclosure Building (an extent of condition building reviewed in AR 01664399) using current data from core sample and in-situ reviews such as crack mapping etc. The purpose of this remodeling was to conduct a seismic reanalysis to demonstrate the effects of the reduced modulus on structural response because of the unique design of the building and because of the need to address a global response vs. a localized response. The completion of this review was not expected until early 2012. NRC review determined that the initial evaluation for the CEB did not address the response of the entire structure to seismic loading comparable to the methods described in UFSAR 3.8 and how the induced seismic stresses would shift between the concrete and the steel in adjoining sections of the structure. In response, NextEra pointed out that they began development of an analytical model to reanalyze the CEB using the as-measured elastic modulus (40% reduced) applied to that ASR-impacted sections. The results of this analysis will be further reviewed as a part of an Engineering Evaluation scheduled to be completed by March 2012.

Notwithstanding the acceptable revised operability determination based on available information, the inspectors noted that these determinations listed no assumptions in the applicable sections. The inspectors also noted that design basis code ACI 318-1971 was based on empirical data for determining certain parameters that are a part of the design bases. Also Poison's ratio on concrete cores are not being tested or evaluated and this ratio was in the UFSAR. The assumption of this empirical data was that the relationships were for ASR free concrete. Further, based on a review of the implementation schedule from contractor-submitted plans that do not have NextEra approval as of Nov. 29th, the inspectors noted that the plans do not address the following which are directly related to addressing the unwritten assumptions being made in the prompt operability determinations:

1. The plans do not appear to test concrete cores for the following key design parameters from the design basis code ACI 318-1971, as for tensile and shear strength, rebar bond strength and Poison's ratio.
2. The plans do not appear to address nondestructive testing to assess the current progression of the ASR expansion rate before the destructive tests of the concrete cores.

3. There was a apparent lack a clear framework for concrete core sampling in the buildings to ensure how representative the core sampling addressing the need for random core sampling in distinction to smart sampling on worst case conditions/doing a bounding calculation along addressing the impact of too much core boring and re-grouting on the building structural integrity .
4. The plans do not appear to address potential effects of other degradations from an aggressive groundwater environment along with the presence of ASR.

In summary at the close of the inspection, NextEra continued to work on: their plans and implementing schedule, building initial assessments along with evaluation results for additional core sampling; identifying the in-situ and out-of-situ testing (concrete core samples and nondestructive testing of concrete core samples) for the structure areas affected by alkali-silica-reaction; need to address key design parameters for the buildings, such as compressive strength, tensile strength, bond strength (between rebar and the concrete), modulus of elasticity and Poisson's Ratio in terms of how alkali-silica-reaction has affected the non-alkali-silica-reaction functional relationship between these parameters per the design code ACI 318-1971; and, remodeling efforts on the Containment Enclosure Building.

Overall, this area remains open pending further project and test plan development by NextEra and further NRC staff review of the final operability determinations on or about March 2012.

4OA6 <u>Meetings, Including Exit</u>

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, 2011 a final exit meeting was conducted and lead 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

Enclosure

ATTACHMENT

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 Manager Participants for Teleconference of January 12, 2012

NRC Staff

C. Miller, Director Division of Reactor Safety (DRS), Region I
R. Conte, Chief Engineering Branch No. 1, DRS, Region I
A. Burritt, Chief Reactor Projects Branch No. 3, Region I

LIST OF ITEMS OPENED, CLOSED, DISCUSSED, AND UPDATED

Opened:

05000443/2011-010-01	FIN	Inadequate Operability Determination for Degraded Concrete Structures Housing Safety-Related Equipment
05000443/2011-010-02	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	NCV	Prompt Operability Determination for Safety Related Structures affected by ASR.

Documents Reviewed:

LIST OF ACRONYMS

AR	????
IMC	Inspection Manual Chapter
NCV	Non-Cited Violation
NRC	U.S. Nuclear Regulatory Commission
NRR	Nuclear ????
TS	Technical Specification
URI	Unresolved Item