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Sakai, Stacie

From:	Conte, Richard
Sent:	Friday, January 06, 2012 6:42 PM
To:	Burritt, Arthur; Chernoff, Harold; Galloway, Melanie; Miller, Chris; Wilson, Peter; Murphy, Martin
Cc:	Chaudhary, Suresh; Cline, Leonard; Lehman, Bryce; Modes, Michael; Morey, Dennis; Plasse, Richard; Raymond, William; Sakai, Stacie; Sheikh, Abdul; Thomas, George; Auluck, Rajender
Subject:	SBK ASR Standalone Report
Attachments:	20120106 050442_201110 Seabrook ASR Standalone.docx

Here is the ASR Standalone report with strong message and request.

Need big picture comments by COB Monday 1/9, for a go/no go on re-exit for Thursday Jan. 12. Accordingly to Chris Miller, Melanie wanted to participate in the re-exit.

Keep your pen in your pocket and hand off the key board – consider this very draft – big picture comments only.

Rich Conte, EB-1 Branch Chief, Region I
(610) 337-5183 (Office)
(b)(6) (NRC cell)

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 December 2, 2011, 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 December 2, 2011 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.

On January 12, 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, our staffs discussed NRC staff concerns related to the lack of a final correction action plan to address the alkali-silica reaction in safety related structures – a significant condition adverse to quality. The details discussed are addressed in section 4OA6. For a significant condition adverse to quality, a corrective action program is needed to address the diagnosis, prognosis, appraisal and long term effects for the duration of the license for important-to-safety concrete structures subject to alkali-silica reaction (ASR) degradation. The problem was noted by NRC staff on or about October 2010 and, to date, plans have not been finalized. The current plans continue to be underdevelopment and appear to fall short of comprehensively addressing the key issues for confirming the related design basis. Further much of the prompt operability determinations are qualitative and, where calculations are made, they are dependent on ASR not affecting those functional relationships. These same operability determinations have evolved from operable, to operable but degraded with noted findings in implementing you process. While you have agreed that these issues need to be addressed, the details are still unclear and the final plans have languished. Notwithstanding the operability determinations as currently being acceptable, they do reflect a certain amount of concrete material property uncertainty and the staff expressed concern with the delay in finalizing a corrective action plan that will assure a viable engineering evaluation and final operability determination on or about March 2012.

Accordingly we request that you provide your corrective action plan along with root cause of the ASR problem in writing in response to this inspection report. We also asked that you provide in that submittal a basis for continued operations in light of the uncertainty in data associate with your prompt operability determinations. We request that you provided the submittal within 30 days of the issue of this inspection report. If an NRC management meeting on any of these issues is needed, we request that it occur within 30 days of the issue of this inspection report with the final submittal no later than 45 from the issue date of this report.

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,

Richard J. Conte, Chief
Engineering Branch 1
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

cc w/encl: Distribution via ListServ

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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 12, 2012 (Conference Call)
Inspectors:	M. Modes, Senior Reactor Inspector, Region I S. Chaudhary, Reactor Inspector, Region I W. Raymond, Senior Resident Inspector, Seabrook
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;

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". 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.

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

NEEDS TO BE UPDATED TO LATEST

Green. The inspectors identified a Green non-cited violation of Technical Specification 6.7.1.a that requires written procedures be established and implemented, including administrative procedures that define authorities and responsibilities for safe operation. 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 as required by NextEra procedure EN-AA-203-1001 prior to an NRC inspection. Additional evaluations were completed to assure the structures remained functional for design basis conditions.

Green. The inspectors identified a Green non-cited violation of Technical Specification 6.7.1.a, Procedures and Programs, which requires that procedures as described in Regulatory Guide 1.33 be implemented that define administrative requirements for safe operation, including 50.59 screens to be performed prior to implementation of an activity to determine whether a 10 CFR 50.59 evaluation should be performed. 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. EC272057 incorporated the degraded concrete modulus into the Seabrook design basis. Contrary to TS 6.7.1.a, NextEra did not complete a 10 CFR 50.59 evaluation prior to implementing changes to the facility as described in EC272057.

REPORT DETAILS

Summary of Plant Status

Seabrook Station operated at full power during the period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R15 Operability Determinations and Functionality Assessments (71111.15 – 3 samples)

a. Inspection Scope

The inspectors reviewed operability determinations for the below grade concrete walls of seismic Category I buildings affected by alkali-silica reaction. NextEra determined that sections of the below grade concrete walls of seismic Category I buildings were affected by alkali-silica reaction (ASR). NextEra addressed the degraded conditions in the corrective action program and completed operability determinations for affected structures. Prior NRC review of this area was documented in Inspection Reports 2010-04, 2010-05, 2011-02, 2011-03 and 2011-07.

b. Findings

Inadequate Operability Determination – AR 581434, 1644074 and 1664399

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 fully address the affect of the reduced modulus on: the global dynamic response of the affected Category I structures during seismic events, the stiffness and natural frequency of the affected structures and the performance of systems and components attached to the affected structures. In response to this issue NextEra completed additional evaluations that determined the structures and other affected systems and components remained functional for design basis conditions.

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.

The inspectors reviewed NextEra's competed PODs for the ASR-affected Category I concrete structures and determined that the evaluations were not adequate because they 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 related to the containment enclosure building and three other structures. The PODs evaluated the effect of the reduced modulus by conducting a static analysis of the locally impacted sections of the structure, but did not address the response of the entire structure to seismic loading and how the induced seismic stresses would shift between the concrete and the steel in adjoining sections of the structure.

In response to the NRC-identified deficiencies, on October 14, 2011, NextEra completed additional analysis to support the engineering evaluation of the ASR impacted structures that included 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 NextEra not adequately analyzing the effects of the reduced modulus of elasticity on Category I structures 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 based on available information. 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 based on available information.

Enforcement. Because this finding does not involve a violation and has very low safety significance, it is identified as **FIN 05000443/2011-10-01, Inadequate Operability Determination for Degraded Concrete Structures Housing Safety-Related Equipment.**

1R17 Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications (71111.15 – 1 samples)

a. Inspection Scope

The inspectors reviewed an unresolved item No. 05000442/2011003-02 to ensure the NextEra properly conducted a 50.59 screening and subsequent evaluation if needed related to EC272057 on April 25, 2011, to address reduced concrete modulus of elasticity in the control building electric tunnel and the containment enclosure building. EC272057 incorporated the degraded concrete modulus into the Seabrook design basis. Prior NRC review of this area was documented in Inspection Report 05000442/ 2011-03.

b. Findings

Introduction: A Severity Level IV NCV of 10 CFR 50.59(d)(1), "Changes, Tests, and Experiments," was identified by the inspector for the NextEra's failure to provide an evaluation that adequately documented why implementing a design change regarding the reduced concrete modulus of elasticity did not present a more than minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety previously evaluated in the updated safety analysis report (USAR). 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. EC272057 incorporated the degraded concrete modulus into the Seabrook design basis. Contrary to 10 CFR 50.59(d)(1), NextEra did not complete a 10 CFR 50.59 evaluation prior to implementing changes to the facility as described in EC272057.

Description: NextEra determined that certain below grade concrete walls are 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 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 prior NRC approval would be required for proposed changes to the facility. The Seabrook 5059 Resource Manual establishes the program at Seabrook for completing 5059 evaluations, including 5059 screens to be performed prior to implementation of an activity to determine whether a 10 CFR 5059 evaluation should be performed. 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 (page 55) regarding the manner in which code requirements are met to assure the design functions. The concrete modulus is a design value specified in the UFSAR and ACI 318 Building Code. The reduced modulus in the ASR impacted concrete walls has the potential to affect the flexural capacity and dynamic response of the impacted structures. The reduced modulus would be an adverse affect related to the ACI 318 code used to establish the acceptable structural design requirements. Thus, a reduction in the modulus design value affects the design function acceptance criteria which should be screened in as requiring further evaluation per 50.59(c)(2). The magnitude of the adverse effect (that is, is the minimal increase standard met) is the focus of the 50.59(c)(2) evaluation. Using the guidance of the Seabrook 10CFR5059 Resource Manual and NEI 96-07, Revision 1, the inspectors determined NextEra should have evaluated the change to the facility per 10 CFR 50.59(c)(2). 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.

On October 14, 2011, NextEra issued additional information in 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 causes the concrete to have increased flexure, which was evaluated to show that the reduction in capacity was minimal and the stresses on the steel and concrete remain below the design stress limits with margin. Similarly, the affect of the reduced modulus reduced the natural frequency of the structures, which was evaluated to show the shift in natural frequency was minimal and remained well above the ground response peak frequency range such that the response of the structures remains rigid. Although the effect on 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 were 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 4OA5 of this report for further NRC reviews of the revised operability determinations for ASR impacted structures. The inspectors determined that the failure to evaluate changes to the facility as described in EC272057 in accordance with 10 CFR50.59(c)(2) and the 5059 Resource Manual was a performance deficiency.

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 5059 screen properly because they misunderstood the guidance in the 5059 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 5059 Resource Manual when preparing the 5059 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 VI.A 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 05000443/2011-003-02: 50.59 Evaluation for Accepting reduced concert modulus of elasticity in certain safety related structures affected by Alkali Silica Reaction (ASR).

The review of EC272057 on April 25, 2011, to address reduced concrete modulus of elasticity in the control building electric tunnel and the containment enclosure building was addressed in section 1R17 of this report. A finding was identified.

.2 (Open) Unresolved Item 05000443/2011-003-03: Open Operability Determinations for Safety Related Structures affected by Alkali Silica Reaction (ASR)

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.

One finding was identified and it was addressed in section 1R15 of this report. 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 needs to be 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.

As a result of the reported values from core samples, the reduction of the Control Building 'B' Electrical Tunnel Concrete compressive strength and modulus of elasticity was previously evaluated by NextEra as part of a broader immediate operability determination. This calculation demonstrated the continued integrity of the structure, and showed that, although degraded, the structure would perform its static design basis function. However, 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 adequate 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. The same is true for the Diesel Generator Building. The modulus of elasticity of concrete is 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." Thus, the NRC concluded the operability determination was incomplete because the supporting analysis did not take into account the effect of alkali silica degradation on the structural stiffness of the wall and therefore the wall's dynamic response. The incompleteness of the operability determination was addressed as a finding in section R15 of this report. In response NextEra agreed with the determination, current information was available to address the issue, and they indicated the need to address the issue in an updated prompt operability determination.

A secondary issue of anchor integrity was identified by the NRC reviewers. The prompt operability had not addressed the ability of the concrete and anchor system to resist pullout in light of the degradation due to alkali-silica-reaction. Again, NextEra agreed with the determination, current information was available to address the issue, and they indicated the need to address the issue in an updated prompt operability determination.

At the time of the inspection, the NRC staff determined that there was no immediate safety issue for the following reasons:

1. the tunnel should still perform its dynamic function because the degradation is localized, and the other parameters did not reflect a significant negative shift;

2. the wall should remain, as a whole, stiff enough for the intended purpose;

3. because the anchors, used to attach critical systems and components to the degraded walls, is inserted below the first layer of reinforcing bar, and concrete will resist removal by breaking along a direction 45 degrees to the pulling force, an anchor generally cannot be pulled from beneath the rebar; and

4. the alkali silica degradation increases the volume of the concrete, and therefore, the force on the anchored device, in effect binding the anchor further.

Because NextEra did not address these issues in their operability determination, they were to document their justification for anchor operability, and tunnel wall stiffness, in the upcoming revision to their operability determination. The revised prompt operability determinations, AR 581434, Revision 1, for the Control Building "B" electrical tunnel and AR 01664399, Revision 1, for the Containment Enclosure and other building designated for the extent of conditions review were completed on October 14, 2011 and addressed the above noted issues.

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.

The above analysis appeared to indicate that, although the effect of the alkali-silica-reaction on the impacted walls was to reduce the design modulus parameter, the structural integrity remained fully intact under all design loads, hence, the buildings were considered Operable, but degraded, and below 'Full Qualification'. However, 'Full Qualification' may be established after the testing and analysis plans developed to

address alkali-silica-reaction have been completed and incorporated into the updated final safety analysis report and design documents.

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 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. The completion of this review is not expected until early 2012.

As of December 2, 2011, NextEra continued to work on:

1. their plans and implementing schedule, building initial assessments along with evaluation results for additional core sampling;

2. 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;

3. 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,

4. remodeling efforts on the Containment Enclosure Building.

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 Meetings, including Exit

On September 30 and December 2, 2011, the inspectors presented the interim results of this inspection to Mr. P. Freeman 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 12, 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 concerns were communicated to NextEra Management as follows with respect to the ASR problem in safety related structures.

For a significant condition adverse to quality, a corrective action program is needed to address the diagnosis, prognosis, appraisal and long term effects for the duration of the license for important-to-safety concrete structures subject to alkali-silica reaction (ASR) degradation. This corrective action plan should, in general, consist of the following elements: (i) condition assessment (extent and characterization); (ii) root cause; (iii) testing to estimate "expansion to date" and "the current expansion rate"; (iv) testing to estimate "potential for further expansion"; (v) interim and long term structural appraisal; (vi) long term monitoring for the duration of the operating license; (vii) mitigation and remedial measures and (ix) potential for further deterioration due to other mechanisms. The problem was noted by NRC staff on or about October 2010 and, to date, plans have not been finalized. The current plans continue to be underdevelopment and appear to fall short of comprehensively addressing the broad issues above. More specifically, the current plans:

1. appear to be at the contractor input stage without a finalized project and test plan issued by NextEra;
2. do not appear to exist to test concrete cores for the following key design parameters from the design basis building code ACI 318-71: tensile strength, shear strength, Poisson's ratio.
3. do not appear to consider other nondestructive test to assess the current progression of the ASR expansion rate before destructive testing of the concrete cores; and,
4. lack a clear regulatory framework for core concrete sampling e.g. random sampling vs. worst case sampling and bounding calculations.

Further much of the prompt operability determinations were qualitative and, where calculations are made, they were dependent on ASR not affecting those functional relationships. These same operability determinations have evolved from operable, to operable but degraded with noted findings in implementing you process. While the license has agreed that these issues need to be addressed, the details are still unclear and the final plans have languished. Notwithstanding the operability determinations as current being acceptable, the staff expressed concern with the delay in finalizing a

corrective action plan that will assure a viable engineering evaluation and ultimately a viable final operability determination on or about March 2012.

NRC management asked the licensee to provide their corrective action plan along with root cause of the ASR problem in writing in response to this inspection report. NRC management asked that a basis for continued operation be provided in that submittal to address the data uncertainties in the prompt operability determination and that the submittal would be made within 30 days of the issue of the this inspection report. NextEra management agreed?????. NRC management agreed that if a management meeting was needed to discuss any regulatory uncertainties associated with the above, then the management meeting should occur in 30 days of the issue of this inspection report with the final submittal no later than 45 from the issue date of this report.

ATTACHMENT: SUPPLEMENTARY INFORMATION

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

Licensee
P. Freeman
R. Noble, Director of Engineering
M. Collins, Manager, Design Engineering
M. O'Keefe, Licensing Manager

NRC Staff
C. Miller, Director Division of Reactor Safety (DRS), Region I
M. Galloway, Acting Director, Division of License Renewal, Office of Nuclear Reactor Regulation
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 5059 Screen for EC272057

Updated

05000443/2011-003-02	URI	Review of 50.59 screening to accept-as-is reduced values for concrete properties in safety related structures.
05000443/2011-003-03	NCV	Prompt Operability Determination for Safety Related Structures affected by ASR.

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