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

August 28, 2009

EA-09-145

Mr. Gene St. Pierre
Site Vice President
NextEra Energy Seabrook, LLC
Seabrook Station
c/o Mr. Michael O'Keefe
P.O. Box 300
Seabrook, NH 03874

SUBJECT: SEABROOK STATION, UNIT NO. 1 - NRC INSPECTION REPORT 05000443/2009007; PRELIMINARY WHITE FINDING

On July 16, 2009, the NRC completed an inspection at the Seabrook Station, Unit No. 1. The enclosed report documents the inspection findings discussed during an exit meeting on July 16, 2009, with Mr. Paul Freeman and other members of your staff.

This 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.

This report documents one self-revealing finding that has preliminarily been determined to be White, a finding with low to moderate increased importance to safety that may require additional NRC inspections. As described in Section 1R18 of the attached report the finding is associated with the failure to establish adequate design control measures to modify a cooling water flange on the B emergency diesel generator (EDG), which led to the failure of the diesel during a test on February 25, 2009. This finding was assessed based on the best available information, using the applicable Significance Determination Process (SDP). The final resolution will be conveyed in separate correspondence.

Following the B EDG failure on February 25, 2009, NextEra investigated the event, evaluated the condition of the EDG and its support systems, and restored the EDG and its cooling system to an operable status. Following completion of repairs, NextEra performed extensive maintenance operability and reliability runs on the B EDG, and declared it operable on March 2, 2009. This finding does not represent an immediate safety concern because of the corrective actions you have taken.

The finding is an apparent violation of NRC requirements and is being considered for escalated enforcement action in accordance with the Enforcement Policy, which can be found on the NRC's Web site at http://www.nrc.gov/reading-rm/doc-collections/enforcement

In accordance with the NRC Inspection Manual Chapter (IMC) 0609, we intend to complete our evaluation using the best available information and issue our final determination of safety significance within 90 days of the date of this letter. The significance determination process encourages an open dialogue between the NRC staff and the licensee; however, the dialogue should not impact the timeliness of the staff's final determination. We understand that you continue to evaluate the results of your risk determination for the B EDG failure. We encourage you to provide the results of your evaluation to us when it is finalized using the process as described below.

Before we make a final decision on this matter, we are providing you with an opportunity (1) to attend a Regulatory Conference where you can present to the NRC your perspective on the facts and assumptions the NRC used to arrive at the finding and assess its significance, or (2) submit your position on the finding to the NRC in writing. If you request a Regulatory Conference, it should be held within 30 days of the date of this letter and we encourage you to submit supporting documentation at least one week prior to the conference in an effort to make the conference more efficient and effective. If a Regulatory Conference is held, it will be open for public observation. If you decide to submit only a written response, such submittal should be sent to the NRC within 30 days of the date of this letter. If you decline to request a Regulatory Conference or submit a written response, you relinquish your right to appeal the final SDP determination, in that by not doing either you fail to meet the appeal process outlined in the Prerequisite and Limitation Sections of Attachment 2 of IMC 0609.

Please contact Art Burritt at 610-337-5069, and in writing, within 10 days from the issue date of this letter to notify the NRC of your intentions. If we have not heard from you within 10 days, we will continue with our significance determination and enforcement decision and you will be advised of the results of our deliberations on this matter.

Because the NRC has not made a final determination in this matter, no Notice of Violation is being issued for this inspection finding at this time. In addition, please be advised that the number and characterization of the apparent violation may change as a result of further NRC review. The final resolution of this finding will be conveyed in separate correspondence.

The attached report also documents one licensee-identified finding of very low safety significance (Green) that involved a violation of NRC requirements (Section 4OA7). If you contest this violation, 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 the Seabrook Station. The information you provide will be considered in accordance with Inspection Manual Chapter 0305.

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,

/RA by James W. Clifford Acting for/

David C. Lew, Director
Division of Reactor Projects

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

Enclosure: Inspection Report 05000443/2009007
w/Attachment: Supplemental Information

cc w/encl:
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M. Warner, Vice President, Nuclear Plant Support
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Board of Selectmen, Town of Amesbury
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R. Shadis, New England Coalition Staff
M. Metcalf, Seacoast Anti-Pollution League

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,

/RA by James W. Clifford Acting for/

David C. Lew, Director
Division of Reactor Projects

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SUNSI Review Complete: LC (Reviewer's Initials)

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U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	50-443
License No.:	NPF-86
Report No.:	05000443/2009007
Licensee:	NextEra Energy Seabrook, LLC
Facility:	Seabrook Station, Unit No.1
Location:	Seabrook, New Hampshire 03874
Dates:	February 25, 2009 through July 16, 2009
Inspectors:	W. Raymond, Senior Resident Inspector C. Cahill, Senior Reactor Analyst, DRS K. Mangan, Senior Reactor Inspector, DRS R. Moore, Reactor Inspector, DRP J. Heinly, Reactor Inspector, DRP J. Rady, Reactor Inspector, DRS E. Burket, Reactor Inspector, DRS
Approved by:	David C. Lew, Director Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000443/2009007; 02/25/2009-07/16/2009; Seabrook Station, Unit No. 1; Plant Modifications.

The report covered a four-month period of inspection by resident and regional inspectors. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP) and the cross-cutting aspect of a finding is determined using IMC 0305, “Operating Reactor Assessment Program.” One apparent violation was identified. Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

Cornerstone: Mitigating Systems

Preliminary White. A self-revealing apparent violation of 10 CFR 50, Appendix B, Criterion III, Design Control was identified following a review of the identified causes for the failure of the B EDG jacket water cooling system on February 25, 2009. Specifically, NextEra’s failure to adequately control design changes implemented on the B EDG jacket water cooling system in January 2009 led to the failure of the gasket on flange JTR005 in the B EDG jacket water cooling system on February 25.

The inspectors determined that this finding is more than minor because it is associated with the design control attribute of the Mitigating Systems Cornerstone and affects the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, design modification 08MSE11, intended to address flange JTR005 alignment and change the flange gasket design was inadequate and resulted in inoperability of the B EDG. In accordance with IMC 0609, “Significance Determination Process,” a Phase 3 risk analysis was performed and determined that the calculated delta CDF for the finding was 2.27E-6, which represents a low to moderate safety significance or White finding. The cause of the finding is related to the corrective action component of the cross-cutting area of problem identification and resolution because NextEra did not thoroughly evaluate problems in a timely manner such that resolutions address causes (P.1(c)). Specifically, NextEra did not adequately evaluate deficient conditions when addressing B EDG cooling water flange leaks, failed to adequately use readily available internal operating experience, and failed to adequately evaluate and correct the impact of engine vibrations on flange JTR005 integrity, contributing to a subsequent failure of the flange. (1R18)

Other Findings

Violations of very low safety significance (Severity Level IV) that were identified by NextEra, have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee’s corrective action program. These violations and the licensee’s corrective action tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Mitigating Systems

1R18 Plant Modifications (71111.18)

a. Inspection Scope

On February 25, 2009, the B EDG failed to complete a routine operability test when a leak occurred on the engine from a two bolt flange (joint JTR005) on the right bank (RB) turbocharger at the connection to the jacket water return line. NRC Inspection Report 2009002 documented NextEra's immediate response and the NRC's initial review of the event. As of the end of the inspection documented in that report, NextEra's evaluation of the causes for the failure were still ongoing and the inspectors had identified several issues of concern regarding the adequacy of the repairs and modifications completed during the January 2009 overhaul and the adequacy of corrective actions taken to assess and correct the potential effect of the RB turbo vibrations on EDG operability. The NRC opened URI 05000443/2009002-01 to track NextEra's completion of the root cause evaluation for the February 25 event and the NRC's subsequent review of NextEra's completed evaluation.

To close URI 05000443/2009002-01 the inspectors reviewed NextEra actions to monitor B EDG conditions and address identified deficiencies including work completed during the B EDG overhaul conducted between January 29 and February 2, 2009. The inspectors reviewed NextEra modifications to the B EDG jacket water cooling system piping and gaskets on flanged connections, including the design changes in 00MMOD531, 06MSE037, 08MSE211 and EC144905. In particular, the inspectors reviewed the flange gasket change completed per maintenance support evaluation (MSE) 08MSE211, and the repairs conducted per work order (WO) 0821400 to address alignment. The inspectors also reviewed NextEra actions to address vibrations in the RB turbo during engine operation and the results of the root cause investigation for the February 25, 2009 event, including the evaluations conducted for Action Request AR 191440. This inspection did not represent an inspection sample.

b. Findings

Introduction. A self-revealing apparent violation (AV) of 10 CFR 50, Appendix B, Criterion III, Design Control was identified following a review of the identified causes for the failure of the B EDG jacket water cooling system on February 25, 2009. Specifically, NextEra's failure to adequately control design changes implemented on the B EDG jacket water cooling system in January 2009 led to the failure of the gasket on flange JTR005 in the B EDG jacket water cooling system on February 25.

Description. Operators shutdown the B EDG during a routine operability test on February 25, 2009, when a leak developed in the RB turbo jacket water cooling line at a 2-bolt flanged connection. The NextEra investigation of the failure found the bolts for flange JTR005 loose and the gasket material severely damaged and blown out along a part of its circumference. Portions of the flange gasket were compressed 60% versus the vendor recommended maximum of 16%. The flange faces had irregularities (bowing and cupped surfaces) and there was a misalignment (gap) between the RB turbo outlet flange and the jacket water coolant pipe flange. The gap ranged from 0.164 to 0.245 inches, and by comparison, the installed gasket material had a nominal thickness of 0.0625 inches. NextEra evaluated the apparent cause of the flange failure and repaired the flange under EC144905 and Work Order 1185637. The repairs included changes to address the flange misalignment, gasket material compression, and positive measures to prevent rotation of the bolts.

The NextEra Root Cause evaluation identified several factors that contributed to the failure of the B EDG jacket water cooling line at flange JTR005. In January 2009 NextEra had implemented design change 08MSE211 to change the flange JTR005 gasket design from a 1/8-inch thick full-face gasket to a 1/16-inch annular configuration. The design change was implemented per work order WO 0821400, which also conducted maintenance to address flange JTR005 alignment. The root cause was that the 1/16 inch annular gasket installed under 08MSE211 was an inadequate design for the flange specific conditions. The combination of thinner gasket annular design, cupped surfaces, flexed flange, flange gap and bolt loosening from vibration resulted in gasket compression well below the minimum required. The gasket vendor specified a bolt pre-load to achieve a 6000 psi compressive force, with a minimum of 3244 psi needed to make the flange connection leak tight. NextEra found that most of the gasket surface was at 1000 psi or less. This resulted in an essentially free floating gasket with no sealing pressure in the area where the gasket failed. Thus, even though flange JTR005 successfully passed a post work test as part of WO 0821400 on January 31, 2009, the as-built gasket design and flange conditions in combination with vibrations which loosened the bolts, left flange JTR005 in a condition to fail with continued B EDG operation.

The cause of the flange JTR005 leak on February 25 was the inadequate design and design control measures used to change the flange gasket from full face to annular configuration. Design Change 08MSE211 addressed leakage considerations by stipulating attributes in the gasket design that address compressive load. 08MSE211 did not address the suitability of the gasket design with adequate consideration of the flange performance history. The gasket design did not adequately consider flange specific conditions (bowing under pre-load, surface irregularities), misalignment (gap) or the effects of vibration. 08MSE211 and WO 0821400 stipulated that the flange condition was required to be “true and flat,” but provided inadequate instructions to the workers on how to achieve the required conditions. The work was assumed to be within the skill of the worker. The work order was also intended to correct flange JTR005 alignment issues. NextEra concluded the excessive gap found between the flange faces was likely caused by the welding completed during WO 0821400. Although 08MSE211 stated, “reweld as required ensuring piping is not pulled,” the design control measure was inadequate because no specific guidance was provided. Similarly, although WO 0821400 stated the repair “should eliminate any misalignment issues providing care is taken not to ‘pull’ the flange in final weld out,” the work order provided no guidance on how to verify or measure flange JTR005 alignment after welding. Design change 08MSE211 and WO 0821400 failed to adequately control the welding process relative to flange alignment; failed to address flange specific irregularities; and failed to address vibration that could impact bolt torque and gasket compressive load. As a consequence, the B EDG jacket water cooling line was left in a condition to fail at flange JTR005 with continued B EDG operation. The inspectors determined that this was a performance deficiency.

The inspectors also determined that the primary contributing cause for the performance deficiency was that NextEra did not adequately use internal operating experience or adequately evaluate deficient conditions when addressing the B EDG cooling water flange issues. The work control, corrective action and engineering records show a documented history of leakage from flange JTR005. While preparing and implementing the gasket design change per 08MSE211, NextEra did not adequately research the performance history for flange JTR005. Readily available plant operating experience showed that a flexible gasket material installed under 06MSE037 was a proven design providing leak free service for two years. The flexible gasket design could better tolerate flange surface imperfections, was better for a flange experiencing vibrations, and could better accommodate gaps between flange surfaces. Had the performance history been adequately considered, NextEra could have either retained the 06MSE037 proven design, or better prepared the 08MSE211 design change to address flange JTR005 conditions. Further, NextEra did not thoroughly evaluate problems such that resolutions addressed causes. Specifically, during the repairs to flange JTR005 per WO 0821400, on January 29, workers requested the use of a locking mechanism on the flange because the fasteners were found less than the required torque (CR200901470). In an evaluation dated February 5, 2009, NextEra concluded a locking feature would be evaluated if the fasteners were loose in the future. The flange failed during the next EDG run on February 25. The failure to adequately review the request for locking devices or evaluate why they were needed was a missed opportunity to prevent vibration induced loosening of the flange bolts. Locking wires were added to the flange as part of the subsequent design change and repair activity under EC144905.

<u>Analysis.</u> The performance deficiency associated with this finding was that inadequate design control measures used to correct flange alignment and change the gasket design on the B EDG right bank turbocharger jacket water cooling line resulted in the B EDG cooling water line failure on February 25, 2009. The Seabrook design control manual requires that the design measures for safety related systems consider the equipment performance history and whether materials are suitable for the application and conditions. Specifically, design change 08MSE211 was inadequate because it did not adequately consider the flange performance history and the suitability of gasket materials and thickness relative to flange specific conditions (cupping and bowing); it did not adequately consider welding stresses during repair and then failed to assure flange alignment was acceptable after welding; and, it did not address the impacts of known vibrations on flange performance and gasket compressive load.

The inspectors determined that this finding is more than minor because it is associated with the design control attribute of the Mitigating Systems Cornerstone and affects the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, design modification 08MSE11, intended to address flange JTR005 alignment and change the gasket design was inadequate and resulted in inoperability of the B EDG. In accordance with IMC 0609, “Significance Determination Process,” Phase 1 worksheets, a Phase 2 risk analysis was required because the finding represents an actual loss of safety function of a single train for greater than the TS allowed outage time of 14 days.

The Phase 2 risk evaluation was performed in accordance with IMC 0609, Appendix A, Attachment 1, “User Guidance for Significance Determination of Reactor Inspection Findings for At-Power Situations.” The total exposure period for the degraded condition was approximately 625 hours (26 days). Using Seabrook's Phase 2 SDP notebook, pre-solved worksheets, and an initiating event likelihood of 3-30 days, the inspector identified that this finding is of potentially substantial safety significance (Yellow). The finding affected sequences in the loss of offsite power (LOOP) and LOOP and Loss of Class 1E 4.16 kV AC Bus A (E5) (LEACA) worksheets. For the LOOP condition, sequences that resulted in a station blackout (SBO) were the dominant contributor to core damage. For the LEACA condition, sequences that involved a stuck open relief valve were the dominant contributor to core damage. The sum of the sequences in the LOOP and LEACA, for the identified exposure period resulted in a Yellow. In recognition that the Phase 2 notebook typically yields a conservative result, a NRC Region I Senior Reactor Analyst (SRA) performed a Phase 3 risk assessment of this finding.

The SRA used the Seabrook Standardized Plant Analysis Risk (SPAR) model, Revision 3.50, dated July 2, 2009, and Graphical Evaluation Module (GEM), in conjunction with the System Analysis Programs for Hands-On Integrated Reliability Evaluations (SAPHIRE), Version 7, software to estimate the internal risk contribution. In discussions with the licensee, it was discovered that the SPAR model for Seabrook did not credit instrument air accumulators. Information on these backup accumulators was included in the SDP notebook. Additionally, Seabrook questioned the modeling of switchgear ventilation and provided design information to support the modeling revision to reflect the design success criteria. The SRA worked with Idaho National Laboratory (INL) and modified the model to correct the instrument air dependencies and modified the ventilation success criteria. Specific changes included:

1. Basic event SWS-FAN-FC-RMCOOL (1E-3) was added to the SWS switchgear cooling fault trees (EPS-DGNA-SWS, EPS-DGNB-SWS, SWS-RMCLA, SWS-RMCLB). This event was ANDed with the existing SWS switchgear ventilation logic.
2. Basic event IAS-TNK-FC-ADV (4.8E-8) was added to the atmospheric steam dump valve (ADV) air supply fault tree (MSS-ADVS-AIR). This event allows operation of the ADVs based on the air supply of the accumulators. The information provided indicated that these accumulators would facilitate 10 cycles over a period of 10 hours.

The following assumptions were used for this assessment:

1. To closely approximate the type of failure exhibited by the B EDG, the SRA used the B EDG failure to run basic event <EPS-EDN-FR-1B > and changed its failure probability to 1.0, representing a 100 percent failure-to-run condition.
2. The exposure time for this condition was 625 hours (546.95 hours, plus 77.75 hours of unavailability during troubleshooting and repair).
3. Based upon the nature of the failure, no additional operator recovery credit was provided.
4. All remaining events were left at their nominal failure probabilities.
5. Cutset probability calculation truncation was set at 1E-13.

Based upon the above assumptions, the Seabrook SPAR model internal contribution to conditional core damage probability (CCDP) was calculated at 1.8E-6. This low E-6 delta CCDP value represents a low to moderate safety significance (White). The dominant internal event sequences involve a loss of offsite power event with subsequent failure of the A EDG and the supplemental emergency power system (SEPS) resulting in a Station

Black Out. Additionally, the site fails to recover a diesel generator within four hours and the failure to recover offsite power within four hours. These Phase 3 SPAR model results correlate well to the Phase 2 SDP Notebook dominant core damage sequences.

The Seabrook Probabilistic Safety Assessment (PSA) is a full scope model that includes events such as seismic events, internal fires and internal floods. The PSA summarizes the contribution mainly from a turbine building fire or flooding as representing approximately 31% of the total (internal and external) core damage frequency, or nearly one third of the annualized risk. For the given exposure period this equates to an external events delta CCDP of 4.7E-7. The NRC does not have an external risk model for Seabrook. Consequently, the SRA used the licensee's external risk assessment to quantify the external risk contribution for this condition.

The SRA used IMC 0609, Appendix H, "Containment Integrity Significance Determination Process," to determine if this finding was a significant contributor to a large early release. The Seabrook containment is classified as a pressurized water reactor large-dry containment design. Based upon the dominant sequences involving loss of offsite power and station blackout (SBO) initiating events, per Appendix H, Table 5.2, "Phase 2 Assessment Factors – Type A Findings at Full Power," the failure of the B EDG does not represent a significant challenge to containment integrity early in the postulated core damage sequences. Consequently, this finding does not screen as a significant large early release contributor because the close-in populations can be effectively evacuated far in advance of any postulated release due to core damage. Accordingly, the risk significance of this finding is associated with the delta CDF value, per IMC 0609, Appendix H, Figure 5.1, and not delta LERF.

The Seabrook model used to evaluate the condition was RISKMAN model DBGOOS which was based on SB2006NM. For the given assumptions, for a failure of the B EDG to run, over the given exposure period, the licensee calculated ΔCDF was 1.48E-6. The contribution from internal events was 1.01E-6, and external event contribution was 4.7E-7. Similar to the NRC internal risk contribution, Seabrook's model illustrates that the largest percentage of internal risk is derived from station blackout events.

For the given assumptions, the licensee and NRC results are in close agreement. As a result, the calculated total risk significance of this finding is based upon NRC analysis. The calculated risk is the summation of internal and external risk contributions (delta CCDP internal + delta CCDP external (fires and floods) = delta CCDP total) which equates to; 1.8E-6 + 4.7E-7 = 2.27E-6 delta CCDP. Annualized, this value of 2.27E-6 delta CDF represents a low to moderate safety significance or White finding.

The cause of the finding is related to the corrective action component of the cross-cutting area of problem identification and resolution in that the licensee failed to thoroughly evaluate problems in a timely manner such that resolutions address causes (P.1(c)). Specifically, NextEra failed to adequately evaluate deficient conditions when addressing B EDG cooling water flange leaks, failed to adequately use readily available internal operating experience, and failed to adequately evaluate and correct the impact of engine vibrations on flange JTR005 integrity.

Enforcement. 10 CFR 50, Appendix B, Criterion III, "Design Control," states, in part, that measures shall be established to assure that regulatory requirements and the design basis for systems and components are correctly translated into specifications and instructions. Measures shall also be established for the selection and review for suitability of application of materials and parts that are essential to the safety-related functions of the systems and components.

The Seabrook Station Design Control Manual (DCM) was developed pursuant to the above to establish design control measures for safety related components, including the emergency diesel generators. DCM Chapter 2, Section 8.0, describes the Maintenance Support Evaluation (MSE) as the design control measure to implement in support of maintenance. When preparing the MSE, the DCM requires that the design inputs and interdisciplinary review guidelines on Figures 4-1-1 through 4-1-14 shall be used to prepare and develop the design change and understand the areas impacted. DCM Figure 4-1-1, Design Inputs, and Figure 4-1-3, Independent Reviewer Guidelines, requires that the design shall consider mechanical requirements such as stresses and vibration; whether materials are suitable for the application; credible failure modes of connected equipment; and, account for equipment performance history.

Contrary to the above, design change 08MSE211, implemented by Work Order 0821400 on January 29 - 31, 2009, to modify and repair a two bolt flange (joint JTR005) on the B EDG right bank turbocharger, did not adequately consider: mechanical requirements such as stresses and vibration; whether materials were suitable for the application; credible failure modes of connected equipment; and, account for equipment performance history. Specifically, design change 08MSE211 and WO 0821400 did not adequately address the suitability of materials relative to flange specific conditions (cupping and bowing); did not adequately control welding stresses during repair and did not assure post weld flange alignment was acceptable; did not adequately consider the flange performance history and potential failures; and, did not address the impacts of known vibrations on flange performance and gasket compression. As a result, the B EDG turbocharger flange JTR005 was left in a condition to fail with continued B EDG operation, and the diesel was declared inoperable during a test on February 25, 2009, when the flange gasket blew out causing a rapid loss of jacket cooling water. This issue was entered into Seabrook's corrective action program as CR 191440. Pending final determination of significance, this finding is identified as an AV (AV 05000443/2009007-01, Inadequate B EDG Design Change). Therefore URI 05000443/2009002-01 was closed.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On July 16, 2009, the resident inspectors presented the inspection results to Mr. Paul Freeman and other members of his staff, who acknowledged the finding. NextEra acknowledged that none of the material examined by the inspectors during the inspection was considered proprietary in nature.

4OA7 Licensee-Identified Violation

The following violation of very low safety significance (Severity Level IV) was identified by NextEra. It was a violation of NRC requirements that met the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as a non-cited violation (NCV).

10 CFR 50, Appendix B, Criterion XVI, Corrective Action, requires that measures be established to assure that conditions adverse to quality are promptly identified and corrected. In the case of significant conditions adverse to quality, the measure shall assure that the cause of the condition is determined and corrective action is taken to preclude repetition.

The Florida Power and Light (FPL) Energy Quality Assurance Topical Report (QATR) was written pursuant to the above and states in Section A-6 that FPL implements a corrective action program to promptly identify and correct conditions adverse to quality. Procedure PI-AA-205 requires that significant conditions adverse to quality be resolved through corrective actions to prevent recurrence.

Contrary to the above, NextEra Nuclear Oversight issued a finding on April 9, 2009, (QR 090-017) after determining that past corrective actions for B EDG turbocharger vibration issues were inadequate and have not been effective based on a past and recent history of increased vibration, bolt failures, bolt loosening, turbocharger related coolant piping weld failures, coolant system leaks and a failure in some instances to document these conditions in the condition reporting system. The failure to resolve long standing and increasing vibration and related issues for the B EDG constituted ineffective corrective action.

The finding was more than minor because the ineffective action to resolve turbocharger vibrations impacted the availability and reliability of a mitigating system. Further, turbocharger vibration was causal to the B EDG failure on February 25, 2009 (reference Section 1R18 above). The finding had very low safety significance because it did not involve a loss of safety function or impact the safety function for a time greater than the allowed outage time in the technical specifications. While increased vibrations were causal to the February 25th B EDG failure, they were not the root cause since the cooling water system would have failed due the inadequate gasket design and irregular flange conditions. Further, the finding identified in QR 09-017 is separate from NRC Violation 20090701 since the inadequate design change resulting in the February 25 B EDG failure occurred during the discrete time period of January 29-31, 2009, whereas the corrective actions for the B EDG turbocharger vibrations have been ongoing for a longer period of time (reference 2001 CR 200107312). The inspectors determined that the Criterion XVI violation was licensee-identified. NextEra entered the issue into the corrective action program as CR 00194370.

SUPPLIMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

R. Arn, Engineering
K. Browne, Assistant Operations Manager
R. Campo, Plant Engineer
P. Freeman, Plant General Manager
G. Kim, Risk Analyst
K. Kiper, Risk Analyst
N. Levesque, Engineering Supervisor
E. Metcalf, Operations Manager
M. Ossing, Engineering Support Manager
M. Palumbo, Plant Engineer
R. Plante, Maintenance Supervisor
R. Samson, Maintenance Supervisor
G. St. Pierre, Site Vice President

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000443/2009007-01	AV	Inadequate B EDG Design Change

Closed

05000443/2009002-01	URI	B EDG Emergency Shutdown During Testing on 2/25/09

LIST OF DOCUMENTS REVIEWED

Miscellaneous
Operations Logs – Various
MRC Associates Report, Modal Analysis of the Turbocharger on the B EDG, June 1992
Fairbank Morse Engineering Report, Turbocharger Vibration, September 27, 1991
Risk Significance of DG-B Failure February 25, 2009, 5/7/09 and 5/27/2009
Engineering Evaluations EE-09-002, Revision 0, 6/24/09; Revision 1, 7/29/09
B EDG Vibration Monitoring Data
System Engineering Notes on B EDG turbocharger vibrations, July 1999
Fairbank Morse/Coltec Industries Engineering Report, Turbocharger Vibration, 9/27/91
ARC Associates Report, Modal Analysis of the Turbocharger Section of the B Diesel Generator, 6/9/92

Condition Reports
Root Cause Analysis for CR 191440, 194370
Action Requests 00191440, 00191586, 00191608
CR200901470, CR199917417, CR200901470, CR200107312, CR200304671, CR200509803, CR200210604, CR200412056, CR200505245, CR200509803, CR200800136, CR200801690, CR200809251, CR200809307, CR200901505

Design Changes
DCR 94-00012, EDG Safety Classification Review, DCN 01
06MSE037, EDG Cooling Water System Gasket Replacement (AFLAS)
00MMOD531, EDG Turbocharger Cooling Water Piping Upgrade, DCN 12
94-064, D.G. Cooling Water System Gasket Replacement, Rev 01
08MSE211, EDG Turbocharger CC Water Piping Optional Gasket Configuration and Bolting Type
EC144905, EDG Turbocharger CC Piping Outlet Cover Modification/Gasket Replacement

Drawings
Drawing B20466, DG Cooling Water System Detail
Drawing 1-NHY-310882, CWD for Pressurizer Pressure Control Valve PCV-456B
P&ID 1-NHY-506402, DB – DG B Lube Oil System Control Loop Diagram
P&ID 1-NHY-504120, DG – DG Temperature Scanner Logic Diagram
P&ID 1-NHY-310008, 4160 Bus E6 One Line Diagram
P&ID 1-DG-B20463, Diesel Generator Lube Oil System Train B Detail
1-NHY-310002, Unit Electrical Distribution One Line Diagram, Rev. 40
1-NHY-310010, D1A and DG-1B One Line Diagram Sh.1, Rev. 14
1-NHY-310010, DG-1A and DG-1B One Line Diagram Sh.2, Rev. 4

Work Orders
Work Orders (WO) 0821400, 0812472, 0442764, 05131067, 072419, 0805715, 01185637

Procedures
PI-AA-205, Condition Identification and Corrective Action
PI-AA-01, Corrective Action and Condition Reporting
ES0815.002, General Welding Procedure, Rev 00, Chg 21
ES0815.004, Welding of Carbon Steel Materials, Rev 00, Chg 08
ES1807.001, Visual Examination Procedure for Welding, Rev 07, Chg 02
MA-AA-203, Work Order Planning Process, Rev 5
MA-AA-202, Work Order Execution Process, Rev 2
MS0517.03, Flange Maintenance, Rev 9

Manuals
FPLE Quality Assurance Topical Report (QATR), Section A-6, "Corrective Action"
Design Change Manual (DCM), Revisions 37- 45
DCM Sections 1.0, 2.0 and 8.0
DCM Figures 4-1-1 through 4-1-14
DCM Figure 4-1-1, Design Inputs
DCM Figure 4-1-3, Independent Reviewer Guidelines

LIST OF ACRONYMS

AR	Action Request
CR	Condition Report
DCM	Design Control Manual
EDG	Emergency Diesel Generator
LERs	Licensee Event Reports
MSE	Maintenance Support Evaluation
NCV	Non-Cited Violation
NRC	U.S. Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
PARS	Publicly Available Records
RB	Right Bank
RV	Reactor Vessel
SDP	Significance Determination Process
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
WO	Work Order