UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
245 PEACHTREE CENTER AVENUE NE, SUITE 1200
ATLANTA, GEORGIA 30303-1257

October 10, 2012

Mr. Dennis R. Madison
Vice President
Southern Nuclear Operating Company, Inc.
Edwin I. Hatch Nuclear Plant
11028 Hatch Parkway North
Baxley, GA 31513

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT- NRC COMPONENT DESIGN BASES INSPECTION - INSPECTION REPORT 05000321/2012008 AND 05000366/2012008

Dear Mr. Madison:

On September 6, 2012, U. S. Nuclear Regulatory Commission (NRC) completed an inspection at your Edwin I. Hatch Nuclear Plant, Units 1 and 2. The enclosed inspection report documents the inspection results, which were discussed on September 6, 2012, with Mr. C. Lane 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 licenses. The team reviewed selected procedures and records, observed activities, and interviewed personnel.

Four NRC identified findings of very low safety significance (Green) were identified during this inspection. Three of these findings were determined to involve violations of NRC requirements. Additionally, the NRC has determined that a traditional enforcement Severity Level IV violation occurred. This traditional enforcement violation was identified with an associated finding. The NRC is treating these violations as non-cited violations consistent with Section 2.3.2 of the Enforcement Policy.

If you contest the violations, or significance of these NCVs, 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 20555-001; with copies to the Regional Administrator Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Hatch Nuclear Plant.

If you disagree with a cross-cutting aspect assignment in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region II, and the NRC Resident Inspector at the Hatch Nuclear Plant.

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 (PARS) component of NRC's Agencywide Document Access and Management 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/

Rebecca L. Nease, Chief
Engineering Branch 1
Division of Reactor Safety

Docket Nos.: 05000321, 05000366
License Nos.: DPR-57, NPF-5

Enclosure:
Inspection Report 05000321/2012008 and 05000366/2012008
w/Attachment: Supplemental Information

cc: (See page 3)

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Letter to Dennis R. Madison from Rebecca L. Nease dated October 10, 2012.

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT- NRC COMPONENT DESIGN BASES INSPECTION - INSPECTION REPORT 05000321/2012008 AND 05000366/2012008

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos:	05000321 and 05000366
License Nos:	DPR-57 and NPF-5
Report Nos:	05000321/2012008 and 05000366/2012008
Licensee:	Southern Nuclear Operating Company, Inc.
Facility:	Edwin I. Hatch, Units 1 and 2
Location:	Baxley, GA 31513
Dates:	June 4 – September 6, 2012
Inspectors:	G. Ottenberg, Resident Inspector, Oconee (Lead) S. Sandal, Senior Reactor Inspector J. Rivera-Ortiz, Senior Reactor Inspector N. Childs, Resident Inspector, Crystal River J. Heath, Resident Inspector, McGuire M. Riley, Reactor Inspector P. Wagner, Contractor (Electrical) M. Yeminy, Contractor (Mechanical)
Approved by:	Rebecca Nease, Chief Engineering Branch 1 Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000321/2012008, 05000366/2012008; 6/4/2012 – 9/6/2012; Edwin I. Hatch Nuclear Plant, Unit 1 and 2; Component Design Bases Inspection.

This inspection was conducted by a team of six Nuclear Regulatory Commission (NRC) inspectors from Region II, and two NRC contract personnel. Three Green non-cited violations (NCV), one Severity Level IV NCV, and one Green finding were identified. The significance of inspection findings is indicated by their color (Green, White, Yellow, Red) using the NRC Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," dated June 2, 2011. Cross cutting aspects are determined using IMC 0310, "Components Within the Cross Cutting Areas," dated October 28, 2011. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy, dated June 7, 2012. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," (ROP) Revision 4, dated December 2006.

<u>NRC identified and Self-Revealing Findings</u>

Cornerstone: Mitigating Systems

- <u>Green</u>. The team identified a non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to verify or check the adequacy of design of the plant service water system including the pump discharge check valves allowable back-leakage. As a result, the licensee entered the issue into their corrective action program as condition report 481741, performed an immediate determination of operability, and placed administrative control over the river level at which the pumps are declared inoperable to a level higher than the one specified in the plant's technical specifications until more detailed analyses could be performed. The limit was reduced back to the original technical specification level following the results of the analysis.

  The failure to verify the adequacy of the plant service water system design through calculational methods or through a suitable test program as required by 10 CFR 50, Appendix B, Criterion III, was a performance deficiency. The performance deficiency was more than minor because it affected the Mitigating Systems Cornerstone attribute of design control and adversely affected the cornerstone objective of ensuring the reliability, availability and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee did not implement a suitable test program to verify design inputs and ensure the capability of the system. The inspectors used Inspection Manual Chapter 0609, Att. 4, "Initial Characterization of Findings," for mitigating systems and Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," and determined the finding to be of very low safety significance (Green) because the finding was a design control deficiency issue that did not result in a loss of operability or functionality of the PSW system. The performance deficiency was indicative of current licensee performance since the system hydraulic model was verified in 2011, and was directly related to the complete documentation and labeling cross-cutting aspect of the resources component in the area of human performance because the licensee did not have accurate design documentation for the potential pump discharge check valve leakage that could cause reverse rotation of the pumps [H.2(c)]. (Section 1R21.2.3)

- Green. The team identified a non-cited violation of 10 CFR 50, Appendix B, Criterion III, “Design Control,” in that the licensee failed to verify or check the adequacy of the design of the intake structure ventilation support function for the plant service water and residual heat removal service water systems. Following the team’s discovery, the licensee performed a bounding analysis and verified that the safety related components in the intake structure would not fail under the worst case high temperature conditions. The licensee entered the issue into their corrective action program as condition report 477809 to address the issue.

  The failure to verify the adequacy of intake structure ventilation design through calculational methods or through a suitable test program as required by 10 CFR 50, Appendix B, Criterion III, was a performance deficiency. The performance deficiency was more than minor because it affected the Mitigating Systems Cornerstone attribute of design control and adversely affected the cornerstone objective of ensuring the reliability, availability and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee did not have adequate measures in place to ensure negative effects due to heat loading did not affect the reliability, availability, and capability of intake structure equipment. The inspectors used Inspection Manual Chapter 0609, Att. 4, “Initial Characterization of Findings,” for mitigating systems and Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” and determined the finding to be of very low safety significance (Green) because the finding was a design control deficiency issue that did not result in a loss of operability or functionality of the plant service water and residual heat removal service water systems. During the inspection, it was determined that there was adequate margin to preclude component failures when conservative heat loading and single failure criteria were assumed. No cross-cutting aspect was assigned to this finding because the failure to provide an adequate calculation or test is not indicative of current licensee performance due to the age of the heat load analysis. (Section 1R21.2.4)

- Green. The team identified a non-cited violation of 10 CFR 50, Appendix B, Criterion XI, “Test Control,” for the licensee’s failure to incorporate adequate acceptance limits in surveillance test procedures used to verify acceptable steady state output voltage of the emergency diesel generators. The licensee performed an immediate determination of operability to verify that the emergency diesel generators would reach and maintain a steady state voltage greater than the minimum 3,860 volts determined by the calculation and issued interim administrative limits for acceptable output voltage until technical specifications can be revised. The licensee entered this issue into their corrective action program as condition report 482310 to address the issue.

  The licensee’s failure to include the correct minimum steady state output voltage as surveillance test acceptance criteria for the emergency diesel generators was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the procedure quality attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the finding challenged the assurance that the acceptance criteria used during surveillance testing would ensure the emergency diesel generators could perform their intended safety function and remain operable. In accordance with IMC 0609.04, “Initial Characterization of Findings,” the team used the mitigating systems column, which resulted in screening the finding through Inspection Manual Chapter 0609 Appendix A, “The Significance Determination Process (SDP) for Findings at Power.” The finding was determined to be of very low safety significance (Green) because it was not a design deficiency resulting in the loss of functionality or operability, did not represent an actual loss of system safety function, did not result in exceeding a technical specification allowed outage time, and did not affect external event mitigation. A cross-cutting aspect was not identified because this issue has existed since the implementation of Improved Technical Specifications on March 3, 1995, and is not indicative of current licensee performance. (Section 1R21.2.12)

- Green. The team identified a finding for the licensee’s failure to follow Regulatory Guide (RG) 1.155, “Station Blackout,” guidance for testing and test control for the emergency diesel generator (EDG) air start system check valves. The testing deficiency was entered into the licensee’s corrective action program as condition reports 490288 and 490210.

  The failure to implement the guidance in RG 1.155, to which the licensee was committed in the station’s Final Safety Analysis Report, was a performance deficiency. The performance deficiency was more than minor because it affected the procedure quality attribute of the Mitigating Systems Cornerstone objective of ensuring the capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the capability of the EDGs to start following a station blackout coping period was not ensured by the licensee’s test acceptance criteria for the air start check valves. The team used Inspection Manual Chapter 0609, Att. 4, “Initial Characterization of Findings,” for mitigating systems and Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” and determined a detailed risk evaluation was required, because the finding represented an actual loss of function of a non-Technical Specification train of equipment designated as high safety significant in accordance with the licensee’s maintenance rule program for greater than twenty-four hours. A regional senior reactor analyst performed an analysis to determine the risk associated with the finding. An actual loss of EDG function following a station blackout would require all of the Unit 1 EDGs to fail to start, because if any Unit 1 EDG ran and was connected to either emergency bus, even for a relatively short time, an air compressor would partially or fully recharge the 1A EDG’s air start tank. The calculation showed that the portion of plant risk that came from common cause fail to start of the Unit 1 EDGs, and of the site’s EDGs was less than the threshold for greater than green for conditional core damage frequency or large early release frequency in the SDP. Therefore, the finding is Green. There was no cross-cutting aspect associated with this finding because the performance deficiency is not indicative of current licensee performance due to the age of the established test acceptance criteria for the check valve leakage. (Section 1R21.2.15)

Cornerstone: Barrier Integrity

- SL IV. The team identified a non-cited violation of 10 CFR 50.72(b)(3)(ii)(A), for the licensee’s failure to provide an 8-hour event notification to the NRC for the plant being in a condition that caused a principal safety barrier to be seriously degraded. The licensee generated condition report 489079 to document the failure to provide the required 8-hour notification.

The team determined that the failure to report a seriously degraded principal safety barrier as required by 10 CFR 50.72(b)(3)(ii)(A) was a performance deficiency. Using the guidance of Inspection Manual Chapter 0612, Appendix B, "Issue Screening," the team determined the performance deficiency involved a violation that could have impacted the regulatory process, therefore, it was dispositioned using the traditional enforcement process. In accordance with Section 6.9.d.9 of the NRC Enforcement Policy, a failure to make a report required by 10 CFR 50.72 is a Severity Level IV violation. Cross-cutting aspects are not assigned to traditional enforcement violations. (Section 1R21.2.2)

Licensee-Identified Violations

None

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R21 Component Design Bases Inspection (71111.21)

.1 Inspection Sample Selection Process

The team selected risk significant components and related operator actions for review using information contained in the licensee's probabilistic risk assessment. In general, this included components and operator actions that had a risk achievement worth factor greater than 1.3 or Birnbaum value greater than 1 $X10^{-6}$. The sample included fourteen components, one component associated with containment large early release frequency , and five operating experience items.

The team performed a margin assessment and a detailed review of the selected risk-significant components and operator actions to verify that the design bases had been correctly implemented and maintained. Where possible, this margin was determined by the review of the design basis and Final Safety Analysis Report (FSAR) response times associated with operator actions. This margin assessment also considered original design issues, margin reductions due to modifications, or margin reductions identified as a result of material condition issues. Equipment reliability issues were also considered in the selection of components for a detailed review. These reliability issues included items related to failed performance test results, significant corrective action, repeated maintenance, maintenance rule status, Regulatory Issue Summary 05-020 (formerly Generic Letter 91-18) conditions, NRC resident inspector input regarding problem equipment, system health reports, industry operating experience, and licensee problem equipment lists. Consideration was also given to the uniqueness and complexity of the design, operating experience, and the available defense-in-depth margins. An overall summary of the reviews performed and the specific inspection findings identified is included in the following sections of the report.

.2 Component Reviews

.2.1 Unit 2 Core Spray Pumps [2E21C001A and 2E21C001B]

a. Inspection Scope

The team reviewed the FSAR, technical specifications (TS), applicable plant calculations, and drawings to identify the design and licensing bases requirements of the Unit 2 core spray (CS) pumps. The team examined system health reports, records of surveillance testing and maintenance activities, and applicable corrective action documents to determine if potential degradation was being monitored and prevented or corrected. The team reviewed the CS pump net positive suction head (NPSH) and vortexing design calculations to determine if adequate NPSH would be available to supply the CS pumps under accident conditions. The team also investigated an issue regarding pressurization of the Unit 2 CS pump discharge piping to determine if the condition adversely affected NSPH available for the CS pumps. The team reviewed surveillance test procedures to determine if acceptance criteria were appropriately correlated to the design and licensing basis requirements. The team also interviewed plant personnel to discuss component issues and performed a walkdown of the core spray system components to assess visible material condition and to check that installation was consistent with design documentation.

b. Findings

No findings were identified.

.2.2 Unit 2 Torus Purge Inlet Containment Isolation Valves [2T48-F309 and 2T48-F324]

a. Inspection Scope

The team selected the Unit 2 torus purge inlet containment isolation valves (CIVs) for review due to their contribution to large early release frequency. The team reviewed the FSAR, TS, applicable plant calculations, and drawings to identify the design bases requirements of the Unit 2 torus purge inlet CIVs. The team examined system health reports, records of surveillance testing and maintenance activities, and applicable corrective action documents to verify that potential degradation was being monitored and prevented or corrected. The team reviewed licensee event reports (LERs) and associated cause evaluations that had been generated due to local leak rate testing (LLRT) failures that occurred in April 2011. The team reviewed the licensee's testing methodology for the Unit 2 torus purge CIVs to determine if they were being tested consistent with 10 CFR 50, Appendix J requirements. The team also interviewed plant personnel to discuss component issues and performed a walkdown of the Unit 2 torus purge CIVs to assess visible material condition and to check that installation was consistent with design documentation.

b. Findings

Introduction: The team identified a Severity Level IV, non-cited violation (NCV) of 10 CFR 50.72(b)(3)(ii)(A), for the licensee's failure to provide an 8-hour event notification to the NRC for the plant being in a condition that caused a principal safety barrier to be seriously degraded.

Description: On April 16, 2011, while performing LLRT on Unit 2 primary containment penetration 2T23-X205, the penetration's primary containment isolation valves (PCIVs) 2T48F309 and 2T48F324 failed the LLRT acceptance criteria, resulting in the plant exceeding the overall containment leakage rate allowed by plant technical specifications.

The licensee recognized that a 60-day LER was required by 10 CFR 50.73 (Reference LER 05000366/2010-001-00 and LER 05000366/2010-001-01), but failed to recognize that an 8-hour 10 CFR 50.72 report was also required. The licensee initially screened this event as not reportable under 10 CFR 50.72 because, at the time of discovery, the plant was in mode 5 and containment was not required to be operable. However, 10 CFR 50.72(a)(1)(ii) states, in part, that a notification is required for those non-emergency events specified in paragraph (b) of this section that occurred within three years of the date of discovery. The team reviewed the event reporting guidelines contained in NUREG-1022, "Event Reporting Guidelines 10 CFR 50.72 and 50.73," Revision 2.

Section 3.2.4 states, in part, that an LER is required for a seriously degraded principal safety barrier and if not reported under 10 CFR 50.72(a), (b)(1), or (b)(2), an ENS notification is required under 10 CFR 50.72(b)(3) [an 8-hour report]. Section 3.2.4(A)(5) of the NUREG also provides an example regarding loss of containment integrity, including as-found containment leak rate testing, which closely resembles this event. Therefore, the team determined that the condition was subject to the reporting requirements of 10 CFR 50.73 and 10 CFR 50.72(b)(3)(ii)(A) [8 hour report].

The underlying technical issue, exceeding the technical specification leakage rate criterion, was previously evaluated using the significance determination process (SDP) in Section 4OA7 of integrated inspection report 05000366/2012002 (ADAMS ML12122A377) and determined to be a licensee-identified violation of very low safety significance (Green). The licensee documented the equipment issue in condition report (CR) 201105213 and completed an enhanced apparent cause determination of the equipment condition. Corrective actions were completed that addressed the causes of the condition, which were inadequate procedural guidance in adjusting valve travel as well as inadequate instructions to repair or replace worn parts identified during periodic maintenance activities. The licensee generated condition report (CR) 489079 to document the failure to provide the required 8-hour report.

<u>Analysis</u>: The team determined that the failure to report a seriously degraded principal safety barrier as required by 10 CFR 50.72(b)(3)(ii)(A) was a performance deficiency. Using the guidance of IMC 0612, Appendix B, “Issue Screening,” the performance deficiency involved a violation that could have impacted the regulatory process, therefore, it was dispositioned using the traditional enforcement process. In accordance with Section 6.9.d.9 of the NRC Enforcement Policy, a failure to make a report required by 10 CFR 50.72 is a Severity Level IV violation. Cross-cutting aspects are not assigned to traditional enforcement violations.

<u>Enforcement</u>: 10 CFR 50.72(a)(1)(ii) states, in part, that a notification is required for those non-emergency events specified in paragraph (b) of this section that occurred within 3 years of the date of discovery. 10 CFR 50.72(b)(3)(ii)(A) requires, in part, that operating reactor licensees shall notify the NRC within 8 hours of the occurrence of any event or condition that results in the nuclear power plant being in any event or condition that results in its principal safety barriers being seriously degraded. Contrary to the above, on April 16, 2011, the licensee failed to recognize that the aforementioned event met the reporting requirements of 10 CFR 50.72(b)(3)(ii)(A) and did not make the required 8-hour event notification. The violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy because it was Severity Level IV and was entered into the licensee’s corrective action program as CR 489079 to address recurrence. (NCV 05000366/2012008-01, Failure to Report a Degraded Primary Safety Barrier per 10 CFR 50.72(b)(3)(ii)(A)).

.2.3 <u>Plant Service Water (PSW) Pumps [1/2P41C001A/B/C/D]</u>

a. <u>Inspection Scope</u>

The team reviewed the plant’s FSAR, TS, design bases documents, and piping and instrumentation diagrams to identify the design bases of the PSW pumps. Design calculations and site procedures were reviewed to verify that the design bases and design assumptions were appropriately translated into these documents. Special attention was given to the pumps' required NPSH and required submergence to prevent air vortices. The effect of the pressure drop across the system's strainers was evaluated with respect to the required system pressure. Component walkdowns were conducted to verify that the installed configurations would support their design bases functions and had been maintained to be consistent with design limits and assumptions. Test procedures and recent test results were reviewed against design basis documents to verify that acceptance criteria for tested parameters were supported by calculations or other engineering documents and that tests and analyses served to validate component operation under accident and transient conditions. Vendor documentation, preventive and corrective maintenance history including pump refurbishment, and corrective action system documents were reviewed to verify that potential degradation was being monitored.

The team reviewed operating procedures and operator training material to verify that risk significant operator actions could be accomplished as relied upon in design basis calculations. The team conducted a walkdown of the PSW system to assess if the operator actions required to rotate the service water pumps discharge strainer could be successfully accomplished. Selected operator actions associated with the following risk significant basic event were reviewed:

- Operator Fails to Clean Strainer by Backwash [OPHE1P41D103]

b. Findings

Introduction: The team identified a Green NCV of 10 CFR 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to verify or check the adequacy of design of the PSW system including the pump discharge check valves allowable back-leakage.

Description: The team identified through the review of analyses and testing for the PSW system, that the licensee failed to adequately verify the PSW system would be able to perform its design function under design basis conditions. Specifically, the licensee failed to analyze the effects of potential back leakage through the PSW pump discharge check valves. The team discovered that system hydraulic analyses SMNH-02-012, "Benchmarking the Unit 1 PSW Model," and SMNH-03-04, "Benchmarking the Unit 2 PSW Model," did not account for back leakage through the valves, nor did the licensee perform a suitable test program to determine the amount of back leakage. Testing of the PSW pump discharge check valves consisted of ASME Inservice Testing which verified the check valves were closed by observing that the idle pump in the same train was not reverse rotating due to leakage past its discharge check valve. As a result of the team's inquiry, the licensee requested information from the pump manufacturer and determined that the PSW pumps may require 3,040 gpm backflow to reverse rotate the pumps. This large loss of flow rate was not accounted for in system modeling and hydraulic analysis. As a result, the licensee generated CR 481741, performed an immediate determination of operability, and placed administrative control over the river level at which the pumps are declared inoperable to a higher level than the one specified in the plant's TS until more detailed analyses could be performed. The limit was reduced back to the original TS level following the results of the analysis.

Analysis: The inspectors determined that the failure to verify the adequacy of PSW system design through calculational methods or through a suitable test program as required by 10 CFR 50, Appendix B, Criterion III, was a performance deficiency. The performance deficiency was more than minor because it affected the Mitigating Systems Cornerstone attribute of Design Control and adversely affected the cornerstone objective of ensuring the reliability, availability and capability of systems that respond to initiating events to prevent undesirable consequences Specifically, the licensee did not implement a suitable test program to verify design inputs and ensure the capability of the system. The inspectors used IMC 0609, Att. 4, “Initial Characterization of Findings,” issued 6/19/12, for mitigating systems and IMC 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” issued 6/19/12, and determined the finding to be of very low safety significance (Green) because the finding was a design control deficiency issue that did not result in a loss of operability or functionality of the PSW system. The inspectors determined that the cause of the finding was indicative of current licensee performance since the system hydraulic model was verified in 2011. It was directly related to the complete documentation and labeling cross-cutting aspect of the resources component in the area of human performance because the licensee did not have accurate design documentation for the potential pump discharge check valve leakage that could cause reverse rotation of the pumps [H.2(c)].

Enforcement: 10 CFR 50, Appendix B, Criterion III, “Design Control,” requires, in part, that design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. Contrary to the above, the licensee did not properly verify the adequacy of the PSW system flow rate to its safety related users through calculational methods or through a suitable testing program. This resulted in the potential to declare the PSW system and the PSW pump discharge check valves operable with unacceptable reverse leakage because the valves could meet test acceptance criteria of a non-reverse rotating pump. Because the finding is of very low safety significance and has been entered into the licensee's corrective action program as CR-481741, this violation is being treated as an NCV consistent with Section 2.3.2 of the NRC Enforcement Policy: (NCV 05000321/2012008-02 and 05000366/2012008-02, Failure to Adequately Account for Potential Pump Discharge Check Valve Back-leakage).

.2.4 Ultimate Heat Sink/Altamaha River

a. Inspection Scope

The team reviewed the capability of the PSW and the Residual Heat Removal Service Water (RHRSW) systems to perform their safety functions assuming worst-case river conditions. The team also reviewed the environmental conditions to which the PSW and RHRSW pumps were exposed and whether the safety related components of the PSW and RHRSW systems were capable of performing their safety functions under these conditions. Additionally, the team reviewed the adequacy of the plant’s traveling screens and the service water strainers, including the limiting conditions and operator actions associated with these components. Furthermore, the team reviewed the plant’s capability to prevent flooding that may damage equipment inside the Intake structure. The team also reviewed aspects of the licensee’s Generic Letter (GL) 89-13 program, specifically the flow rates, temperatures, and the limits imposed on the RHR heat exchangers which are cooled by river water. The team reviewed the regulatory evaluation performed to change the limiting design basis temperature of the ultimate heat sink (Altamaha River) from 95°F to 97°F in order to verify that the licensee adequately addressed the criteria in 10 CFR 50.59 and followed the guidance in NEI 96-07 for implementing changes to the facility as described in the FSAR. For this evaluation, the team reviewed the calculation approach to determine whether the licensee departed from a methodology described in the FSAR. Specifically, the inspectors evaluated a change in a calculation input associated with the heat transfer coefficient (K) of the RHR heat exchanger, where the licensee changed from a constant (K) to a temperature dependent (K).

The inspectors also reviewed operating procedures and operator training material to verify that risk significant operator actions could be accomplished as relied upon in design basis calculations. The inspectors conducted a walkdown of the plant service water system to specifically assess if the operator actions required to rotate the service water pump's discharge strainer could be successfully accomplished.

b. Findings

Introduction: The team identified a Green NCV of 10 CFR 50, Appendix B, Criterion III, "Design Control," in that the licensee failed to verify or check the adequacy of the design of the intake structure ventilation support function for the PSW and RHRSW systems.

Description: Through review of intake structure heat load analyses, dated 5/21/92, which supports operability of the PSW and RHRSW systems, the team identified that the licensee failed to adequately account for worst case design basis heat load in the analyses. Specifically, the licensee failed to analyze the additional heat load associated with the placement of additional pumps into operation in order to mitigate the consequences of design basis accidents and transients. The team discovered that the analyses of the capacity of the ventilation system did not account for the maximum possible number of PSW pumps and RHRSW pumps. Following the team's discovery, the licensee performed a bounding analysis and verified that the safety related components in the intake structure would not fail under the worst case high temperature conditions. The licensee generated CR 477809 to address the issue.

Analysis: The inspectors determined that the failure to verify the adequacy of intake structure ventilation design through calculational methods or through a suitable test program as required by 10 CFR 50, Appendix B, Criterion III, was a performance deficiency. The performance deficiency was more than minor because it affected the Mitigating Systems Cornerstone attribute of Design Control and adversely affected the cornerstone objective of ensuring the reliability, availability and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee did not have adequate measures in place to ensure negative effects due to heat loading did not affect the reliability, availability, and capability of intake structure equipment. The team used IMC 0609, Att. 4, "Initial Characterization of Findings," issued 6/19/12, for mitigating systems and IMC 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," issued 6/19/12, and determined the finding to be of very low safety significance (Green) because the finding was a design control deficiency issue that did not result in a loss of operability or functionality of the PSW and RHRSW systems. During the inspection, it was determined that there was adequate margin to preclude component failures when conservative heat loading and single failure criteria were assumed. No cross-cutting aspect was assigned to this finding because the failure to provide an adequate calculation or test is not indicative of current licensee performance due to the age of the heat load analysis.

Enforcement: 10 CFR 50, Appendix B, Criterion III, “Design Control,” requires, in part, that design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. Contrary to the above, the licensee did not verify or check the adequacy of the intake structure ventilation design through simplified calculational methods or through a suitable testing program. This resulted in the potential to affect the performance of the PSW and the RHRSW systems. Because the finding is of very low safety significance and because it has been entered into the licensee's corrective action program as CR 477809, this violation is being treated as an NCV consistent with Section 2.3.2 of the NRC Enforcement Policy: (NCV 05000321/2012008-03 and 05000366/2012008-03, Failure to Ensure Adequacy of Intake Structure Ventilation Design).

.2.5 Residual Heat Removal Service Water (RHRSW) Pumps [1/2E11C001A/B/C/D]

a. Inspection Scope

The team reviewed the plant’s FSAR, TS, design basis documents, and piping and instrumentation diagrams to identify the design bases of the RHRSW pumps. Design calculations and site procedures were reviewed to verify that the design bases and design assumptions were appropriately translated into these documents. Special attention was given to the pumps’ capability to provide the required flow rate in order to remove the required heat load at the RHR heat exchangers. The team reviewed the adequacy of assumptions and newly imposed limit on the number of plugged tubes in the RHR heat exchanger. Component walkdowns were conducted to verify that the installed configurations would support their design bases functions and had been maintained to be consistent with design limits and assumptions. Test procedures and recent test results were reviewed against design basis documents to verify that acceptance criteria for tested parameters were supported by calculations or other engineering documents and that tests and analyses served to validate component operation under accident and transients. The team also reviewed the new inservice testing baseline of the refurbished RHRSW pumps to validate its adequacy. Vendor documentation, preventive and corrective maintenance history, and corrective action system documents were reviewed to verify that potential degradation was being monitored. The team also reviewed the modification that installed a cutter impeller in the RHRSW pumps and the issues associated with small debris that could enter the pump and be collected in the system’s strainers.

The team reviewed a risk significant operator action for aligning RHRSW system for injection into the reactor vessel in an emergency scenario where normal emergency core cooling is not available. The team reviewed emergency operating procedures and operator job performance measures to verify that this operator action could be accomplished. The team also observed a control room simulator scenario where the operators had to align the RHRSW system for reactor vessel injection, in order to assess operator knowledge and confirm if the instructions in the emergency operating procedures could be successfully accomplished within the expected time frame. The team also discussed with operations training staff the past results of the job performance measure for this evolution to identify any past operator failures or challenges to accomplish this activity. Selected operator actions associated with the following risk significant basic event were reviewed:

- Failure of Operator to Align for Injection of RHRSW into Reactor Vessel [OPHERSWINJ]

b. Findings

No findings were identified.

.2.6 Unit 2 High Pressure Coolant Injection Minimum Flow Valve [2E41-F012]

a. Inspection Scope

The team reviewed the Unit 2 high pressure coolant injection minimum flow valve, 2E41-F012, to verify it was capable of performing its design bases functions. The team reviewed the licensee's calculations of operational margin and verified important inputs into the calculations were sufficiently conservative. The team also verified that the in-field setup of torque and limit switch settings for the valve actuator were within the setup window assumed in design margin calculations, and verified that test equipment accuracies were considered. The maintenance history of the valves and actuators and system health reports were also reviewed to examine mechanical condition of the components. The team reviewed calculations for reduced voltage at the motor terminals to ensure that worst-case voltage was used in calculating available motor output torque when determining margin. The team verified that maintenance was performed in accordance with vendor instructions.

b. Findings

No findings were identified.

.2.7 125V/250V Station Battery System [1/2R42S001A/B]

a. Inspection Scope

The team reviewed the design, testing, and operation of the Unit 1 station batteries to ensure that the batteries were capable of performing their design function of providing an uninterruptable source of power to connected normal and emergency 125 VDC and 250 VDC power loads under a design basis accident and all operating and transient conditions. The team reviewed design calculations to assess the adequacy of the batteries' sizing to ensure they could power the required loads under accident conditions for a sufficient duration, and at a voltage above the minimum required for equipment operation. The team reviewed battery test results to ensure the testing was in accordance with design calculations, plant TS, vendor recommendations, and industry standards; and that the results confirmed acceptable performance of the batteries. The team reviewed design calculations for ventilation sizing requirements to control hydrogen accumulation during normal and postulated accident conditions. Design and system engineers were interviewed regarding the design, operation, testing, and maintenance of the batteries. The team performed a walkdown of the 1A and 1B station battery and associated distribution panels to assess the material condition of the equipment. A sample of condition reports was reviewed to ensure the licensee was identifying and properly correcting issues associated with the station battery system.

The team selected a risk significant operator action for limiting the station battery loads during a station blackout scenario in order to support the operation of RCIC for a period of approximately five hours. The team reviewed abnormal operating procedures to verify that the procedures contained instructions to strip non-essential loads from the station batteries to ensure the mission time of the batteries as relied upon in the design basis calculations. Selected operator actions associated with the following risk significant basic event were reviewed:

- Depress Recovery when RCIC is Available for 5 hours on Battery during SBO [DEPRESSREC]

b. Findings

No findings were identified.

.2.8 Unit 1 "A" Diesel Generator Battery [1R42S002A]

a. Inspection Scope

The inspectors reviewed the 125 VDC Diesel Battery 1A battery sizing calculations, TS surveillance requirements, and completed surveillances to confirm that sufficient capacity existed for the battery to perform its safety function during a postulated station blackout event. In addition, design and system engineers were interviewed regarding the design, operation, testing, and maintenance of the battery. The inspectors also performed a walkdown of the battery and associated charger to assess the material condition of the equipment.

b. Findings

No findings were identified.

.2.9 Automatic Depressurization System (ADS) Initiation Circuitry

a. Inspection Scope

The inspectors reviewed the Unit 1 ADS control logic system to verify that the system was capable of performing its design basis function. The team reviewed results from previous control logic tests and interviewed operations and systems engineers to ascertain that test procedures demonstrate that individual logic trains meet the functional requirements of TS. The team also reviewed the plant analysis used to determine the setting of the bypass timer for the ADS modification in order to verify the timer settings were conservative in limiting peak cladding temperature during a design basis event. The inspectors reviewed a drift study analysis which was applied to associated ADS time delay relays and used to support the 24-month Cycle Extension Project. The review included calculations related to setpoint determinations used to determine uncertainty for ADS time delay relays. The inspectors also reviewed qualification documents specific to ADS electrical equipment potentially exposed to harsh environments.

b. Findings

No findings were identified.

.2.10 Unit 1 “B” Instrument Bus [1R25-S065]

a. Inspection Scope

The team reviewed the plant TS, FSAR, system design criteria, vendor manuals, and selected drawings to identify the design bases for the 1B instrument bus. The team reviewed AC load flow calculations to verify that the 120/208V instrument bus had sufficient capacity to support its required loads under worst-case accident and degraded grid voltage conditions. The team reviewed the overcurrent protection scheme for the 120/208V instrument bus to verify that circuit breakers and fuses used in the bus system protected the loads from spurious tripping events and adverse operating conditions. The team reviewed testing procedures against design basis documents to verify that the acceptance criteria were supported by design calculations or other design basis documents and that the testing served to validate component operation under accident conditions. The team also reviewed the testing procedures to ensure that design assumptions included in calculations were properly verified during testing. The team reviewed system health reports, maintenance records, and work orders to verify that potential degradation of the instrument bus is monitored and trended. The team performed a walkdown of the 120/208V instrument bus to assess operability and material conditions.

The team selected the risk significant operator action for manually transferring the power supply for instrument bus 1B to an alternate source (bus 1A). The team reviewed abnormal operating procedures and operator job performance measures to verify that this operator action could be accomplished as relied upon in the design basis calculations. The team also conducted a walkdown of local operator actions in the plant and the simulator control room with a senior reactor operator in order to assess operator knowledge and confirm if the instructions in the emergency operating procedures could be successfully accomplished within the expected time frame. Selected operator actions associated with the following risk significant basic event were reviewed:

- Operator Action to Manually Transfer Instrument Bus Power Supply [OPHES064/S065FL]

b. Findings

No findings were identified.

.2.11 Unit 1 “A” Diesel Generator Building 600V Motor Control Center (MCC) [1R24-S025]

a. Inspection Scope

The team reviewed alternating current load flow calculations to verify that the 600V bus had sufficient capacity to supply its loads under design basis accident conditions and degraded grid voltage conditions. The team also reviewed the protective device coordination between the 600V/208V/120V buses to verify that the protection scheme would isolate faults associated with the motor control center and ensure availability of other safety related components needed to respond in a design basis accident. The team reviewed system health reports, corrective action documents, and maintenance records to determine whether there were any adverse operating trends. The team performed a walkdown of the 600V safety buses to assess operability and condition. In addition, the team performed a non-intrusive visual inspection of the motor control center to verify that the motor control center showed no signs of material degradation and vulnerability to hazards such as flooding, seismic interactions, and missiles.

b. Findings

No findings were identified.

.2.12 Unit 1 4160 Volt Bus "E" [1R22-S005]

a. Inspection Scope

The team reviewed the plant's FSAR, TS and design basis documents to identify the design bases of the electrical system with particular emphasis on the 4kilovolt emergency bus 1E and its circuit breakers. Electrical drawings and site procedures were reviewed to verify that the design bases and design assumptions had been appropriately translated into these documents. The team reviewed system modifications to verify that the subject modifications did not degrade the component's performance capability and were appropriately incorporated into relevant design documents, drawings and procedures.

The team also reviewed a number of CRs that were initiated over the past three years to evaluate problems and determine if there were reoccurrences of problems related to the power supply bus or its circuit breakers. Component walkdowns were conducted to verify that the installed configurations would support their design bases functions under accident conditions and had been maintained to be consistent with design assumptions. The team reviewed test procedures and results against design documentation and manufacturer's recommendations to verify that acceptance criteria for tested parameters were supported by calculations or other engineering documents and that individual tests and/or analyses served to validate component operation under accident conditions. The team reviewed vendor documentation, system health reports, preventive and corrective maintenance history, and corrective action system documents were reviewed in order to verify that potential degradation was monitored or prevented and that component replacement was consistent with equipment qualification life. The team also reviewed 10 CFR Part 21 evaluations related to circuit breaker failures to determine their adequacy.

b. Findings

Introduction: The team identified a Green non-cited violation of 10 CFR 50, Appendix B, Criterion XI, "Test Control," for the licensee's failure to incorporate adequate acceptance limits in surveillance test procedures used to verify acceptable steady state output voltage of the emergency diesel generators.

<u>Description</u>: During the review of calculation SENH-10-006, "Unit 1 Station Auxiliary System Study," the team observed that the calculation determined that the steady state voltage needed on the emergency 4.16 kilovolt buses to mitigate a loss of coolant accident was 3,860 volts. Surveillance test procedure 42SV-R43-027-1, "Diesel Generator 1C LOCA/LOSP LSFT," contained acceptance criteria for maintaining emergency diesel output voltage at greater than or equal to 3,740 volts and less than or equal to 4,243 volts after steady state conditions were met. Based on this discrepancy between the calculation and the surveillance test procedure, the minimum acceptance limit of 3,740 volts contained in the surveillance test procedure would allow the emergency diesel generator to be considered operable in technical specifications with an output voltage below the minimum 3,860 volts necessary for safety-related components to perform their safety function. This discrepancy also applies for emergency diesel generators 1A, 1B, 2A, and 2C. After review, the licensee concurred that the TS surveillance requirements in TS 3.8.1.2, TS 3.8.1.5, TS 3.8.1.7, TS 3.8.1.9, TS 3.8.1.10, TS 3.8.1.13, TS 3.8.1.17, and TS 3.8.1.18 were non-conservative and had existed since the implementation of Improved Technical Specifications on March 3, 1995. The licensee performed an immediate determination of operability to verify that the emergency diesel generators (EDGs) would reach and maintain a steady state voltage greater than the minimum 3,860 volts determined by the calculation and issued interim administrative limits for acceptable output voltage until the TS can be revised. The licensee entered this issue into their corrective action program, as condition report 482310.

<u>Analysis</u>: The licensee's failure to include the correct minimum steady state output voltage as surveillance test acceptance criteria for the emergency diesel generators was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the procedure quality attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the finding challenged the assurance that the acceptance criteria used during surveillance testing would ensure the emergency diesel generators could perform their intended safety function and remain operable. In accordance with IMC 0609.04, "Initial Characterization of Findings," issued 6/19/12, the team used the mitigating systems column which resulted in screening the finding through IMC 0609 Appendix A, "The Significance Determination Process (SDP) for Findings at Power," issued 6/19/12. The finding was determined to be of very low safety significance (Green) because it was not a design deficiency resulting in the loss of functionality or operability, did not represent an actual loss of system safety function, did not result in exceeding a TS allowed outage time, and did not affect external event mitigation. A cross-cutting aspect was not identified because this issue has existed since the implementation of Improved Technical Specifications on March 3, 1995, and is not indicative of current licensee performance.

<u>Enforcement</u>: 10 CFR 50, Appendix B, Criterion XI, "Test Control," states, in part, that a test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. Contrary to the above, since March 3, 1995, surveillance testing procedures for the emergency diesel generators did not incorporate the appropriate acceptance limits contained in calculation SENH-10-006, "Unit 1 Station Auxiliary System Study," for required steady

state output voltage. This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy because it was of very low safety significance and was entered into the licensee's corrective action program as condition report 482310 to address recurrence. (NCV 05000321/2012008-04 and 05000366/2012008-04, Failure to Incorporate Appropriate Test Acceptance Criteria to Assure Satisfactory Steady State EDG Performance)

.2.13 Unit 1 600 Volt Bus "D" [1R23-S004]

a. Inspection Scope

The team reviewed the plant's FSAR and TS to provide an understanding of the design bases of the 600 Volt Emergency Bus 1D and its circuit breakers. Electrical drawings and site procedures were reviewed to verify that the design bases and design assumptions had been appropriately translated into these documents. The team reviewed system modifications to verify that the subject modifications did not degrade the component's performance capability and were appropriately incorporated into relevant drawings and procedures. The team also reviewed a number of CRs that had been initiated over approximately the past three years to evaluate problems that had occurred and determine if there were reoccurrences of problems related to the power supply bus or its circuit breakers. Component walkdowns were conducted to verify that the installed configurations would support their design bases functions and had been maintained to be consistent with design assumptions. Test procedures and results were reviewed against design documentation to verify that acceptance criteria for tested parameters were supported by calculations or other engineering documents and that individual tests and/or analyses served to validate component operation. Vendor documentation, system health reports, preventive and corrective maintenance history, and corrective action system documents were reviewed in order to verify that potential degradation was monitored or prevented and that component replacement was consistent with equipment qualification life.

The team selected a risk significant operator action for manually aligning 600-volt bus D to the backup 4,160-volt bus. The team reviewed system operating procedures and operator job performance measures to verify that this operator action could be accomplished as relied upon in the design basis calculations. The team also conducted a walkdown of local operator actions in the plant and the simulator control room with a senior reactor operator in order to assess operator knowledge and confirm if the instructions in the emergency operating procedures could be successfully accomplished within the expected time frame. Selected operator actions associated with the following risk significant basic event were reviewed:

- Operator Fails to Align 600V Bus to Backup 4160V Bus [OPHEEPANOLINK]

b. Findings

No findings were identified.

.2.14 Essential Bus “1B” transformer [1R11-S042]

a. Inspection Scope

The team reviewed the plant’s FSAR and TS to determine if there were any specific requirements and to identify the design bases for this transformer. Electrical drawings and site procedures were reviewed to verify that the design bases and design assumptions had been appropriately translated into these documents. The team reviewed system modifications to determine if any changes had been made to this power supply system. The team also reviewed the list of CRs that had been initiated over approximately the past three years to determine if there had been any problems related to the transformer. The installation was observed during a plant walkdown to verify that the installed configuration would function under accident/event conditions and had been maintained to be consistent with design assumptions. Test procedures and results were reviewed against design documentation to verify that acceptance criteria for tested parameters were supported by calculations or other engineering documents and that individual tests and/or analyses served to validate component operation under accident/event conditions. Vendor documentation, system health reports, preventive and corrective maintenance history, and corrective action system documents were reviewed.

b. Findings

No findings were identified.

.2.15 Emergency Diesel Generator Fuel Oil Supply and Delivery System

a. Inspection Scope

The team reviewed the FSAR, TS, design basis documents, and piping and instrumentation diagrams to identify the design bases of the EDG’s fuel oil supply and delivery system. Design calculations and site procedures were reviewed to verify that the design bases and design assumptions had been appropriately translated into these documents. The team found that no recent modifications were made to the Fuel Oil Transfer system. The team reviewed the potential effects of flooding on the system. Component walkdowns were conducted to verify that the installed configurations would support their design bases function under accident, flood, and loss of offsite power conditions and had been maintained to be consistent with design assumptions. Operating procedures were reviewed to verify that component operation and alignments were consistent with design and licensing bases assumptions. Test procedures and test results were reviewed against the design bases to verify that acceptance criteria for tested parameters were supported by calculations or other engineering documents and that tests and analyses served to validate component operation under accident and loss of offsite power conditions. Vendor documentation, preventive and corrective maintenance history, and corrective action system documents were reviewed to ascertain that potential degradation was monitored or prevented.

b. Findings

Introduction: An NRC identified Green finding was identified for the licensee's failure to follow Regulatory Guide (RG) 1.155, "Station Blackout," guidance for testing and test control for the EDG air start system check valves.

Description: Testing of the EDG Air Receiver check valves used an established acceptance criterion of a leak rate of 65 psig in four hours to assure that the air receiver maintains sufficient pressure to restart the EDG at the conclusion of the Station Blackout (SBO) coping period of four hours. This amount of leakage, which could have been considered a successfully passed test, would allow the EDG air start pressure to degrade following failed attempts to start after the onset of an SBO and during the four-hour coping period to the point where the required EDG starting air pressure following the coping period was not assured to be above the required 150 psig. At the onset of a loss of offsite power, the air start system pressure begins at 240 psig and all 3 EDGs on the affected unit will attempt to automatically start, resulting in a decrease of 17.5 psig in the air receiver. Based on procedures and operator training, a second start attempt is expected which would reduce receiver pressure by another 17.5 psig. Assuming the maximum leakage of 65 psig past the check valves allowed by the inservice testing procedure over the four-hour coping time, the available starting air pressure at the end of the four-hour coping period would be 140 psig, which is lower than the minimum pressure to assure a successful start of an EDG. The licensee is committed to RG 1.155 in the facility FSAR, Appendix A, and RG 1.155 states in part, "A test program should be established and implemented to ensure that testing is performed and verified by inspection and audit to demonstrate conformance with design and system readiness requirements." The licensee did not establish and implement a test program that demonstrated EDG system readiness to recover from an SBO due to the test acceptance criteria deficiency.

The inspectors discovered that on June 1, 2009, the licensee tested check valve 1R43-F3034A and found that it leaked at a rate of 64 psig in four hours. However, since it passed the inappropriate acceptance criterion of 65 psig in four hours, the valve was not declared non-functional, nor was any corrective maintenance performed on it. The valve was retested approximately two years later on August 28, 2011, and was found leaking at a rate of 104 psig in four hours. It was then declared non-functional and replaced. The testing deficiency was entered into the licensee's corrective action program as CRs 490288 and 490210.

Analysis: The failure to implement the guidance in RG 1.155, to which the licensee was committed in the station's FSAR, was a performance deficiency. The performance deficiency was more than minor because it affected the Procedure Quality attribute of the Mitigating Systems Cornerstone objective of ensuring the capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the capability of the EDGs to start following an SBO coping period was not ensured by the licensee's test acceptance criteria for the air start check valves. The team used IMC 0609, Att. 4, "Initial Characterization of Findings," issued 6/19/12, for mitigating systems and IMC 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," issued 6/19/12, and determined a detailed risk evaluation was required because the finding represented an actual loss of function of a non-TS train of equipment designated as high safety significant in accordance with the licensee's maintenance rule program for greater than 24 hours. The regional Senior Reactor

Analyst performed an analysis to determine the risk associated with the finding. An actual loss of EDG function following an SBO would require all of the Unit 1 EDGs to fail to start, because if any Unit 1 EDG ran and was connected to either emergency bus, even for a relatively short time, an air compressor would partially or fully recharge the 1A EDG's air start tank. The calculation showed that the portion of plant risk that came from common cause fail to start of the Unit 1 EDGs, and of the site's EDGs was less than the threshold for greater than green for core damage frequency or large early release frequency in the SDP. Therefore, the finding is Green. There was no cross-cutting aspect associated with this finding because the performance deficiency was not indicative of current licensee performance due to the age of the established test acceptance criteria for the check valve leakage.

Enforcement: This finding does not involve enforcement action because no regulatory requirement violation was identified. Because this finding does not involve a violation and is of very low safety significance, it is identified as FIN 05000321/2012008-05 and 05000366/2012008-05, Failure to Provide Appropriate Acceptance Criteria for EDG Air-Start System Check Valves.

.3 Operating Experience

a. Inspection Scope

The team reviewed five operating experience issues for applicability at the Edwin I. Hatch Nuclear Plant. The team performed an independent review for these issues and where applicable, assessed the licensee's evaluation and dispositioning of each item. The issues that received a detailed review by the team included:

- NRC Information Notice 2010-03, "Failures of Motor Operated Valves Due to Degraded Stem Lubricant"
- NRC Information Notice 2011-01, "Commercial Grade Dedication Issues Identified During NRC Inspections"
- NRC Information Notice 2009-16, "Spurious Relay Actuations Result in Loss of Power to Safeguards Buses"
- NRC Information Notice 2011-12, "Reactor Trips Resulting From Water Intrusion Into Electrical Equipment"
- NRC Information Notice 2008-02, "Findings Identified During Component Design Basis Inspections"

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA6 Meetings, Including Exit

On September 6, 2012, the team presented the inspection results to Mr. Cory Lane and other members of the licensee's staff. The inspectors verified that no proprietary information was retained by the inspectors or documented in this report.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel:
C. Lane, Engineering Director
M. Ajluni, Licensing Director
S. Tipps, Principal Licensing Engineer
H. Barnes, Site Design Engineering Supervisor

NRC personnel
R. Nease, Chief, Engineering Branch Chief 1, Division of Reactor Safety, Region II
F. Ehrhardt, Chief, Projects Branch 2, Division of Reactor Projects, Region II
E. Morris, Senior Resident Inspector, Division of Reactor Projects, Hatch Resident Office
S. Sparks, Senior Enforcement Officer, Enforcement and Investigation Coordination Staff, Region II

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

05000366/2012008-01	NCV	Failure to Report a Degraded Primary Safety Barrier per 10 CFR 50.72(b)(3)(ii)(A) [Section 1R21.2.2]
05000321, 366/2012008-02	NCV	Failure to Adequately Account for Potential Pump Discharge Check Valve Back-leakage [Section 1R21.2.3]
05000321, 366/2012008-03	NCV	Failure to Ensure Adequacy of Intake Structure Ventilation Design [Section 1R21.2.4]
05000321, 366/2012008-04	NCV	Failure to Incorporate Appropriate Test Acceptance Criteria to Assure Satisfactory Steady State EDG Performance [Section 1R21.2.12]
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S56746, Instruction Manual, Westinghouse Type DHP-VR Replacement CBs, Rev. 2.0
S-2012-1, Standing Order to Check Circuit Breaker Charging Motors, 1/5/12
SCM-CGDP-005, Capacitor Commercial Grade Dedication, Rev. 11
SCM-CGDP-005.12, Capacitor Commercial Grade Dedication Appendix, Rev. 4
SCM-CGDP-121, Washer Commercial Grade Dedication, Rev. 7
TP-12, Nuclear Plant Offsite Power Steady-State Study, Approved 3/30/10
2Q12 System Health Report for Station Auxiliary DC Power Systems (R22,R24,R25,R27, R42)
1Q12 Unit 1 A70 – Analog Trip and Transmitter System
1989-02-17 Bechtel letter SBO Equipment List
B21-ADS-LP-03801, ADS Lesson Plan, Rev. 4
P285-4, Plant Hatch Intake structure Outside Structure Siltation Study, Rev. 0

<u>Condition Reports Generated as a Result of the Inspection</u>
CR 475870, Difference in acceptance criteria for EDG Voltage Regulator identified during NRC CDBI
CR 475876, Difference in flow observed for ‘A’ & ‘C’ PSW pumps during NRC CDBI inspection
CR 476465, Intake structure heat load limitations identified in NRC CDBI
CR 476774, Fire extinguishers seismically mounted
CR 476783, During CDBI walkdown with the NRC, a question was raised
CR 476794, fire extinguishers seismically mounted
CR 476795, Remote racking device seismically secured and stored
CR 476801, Remote racking device seismically secured and stored
CR 477102, CDBI Inspections of Intake Structure
CR 477110, CDBI Inspections of Intake Structure
CR 477113, Core Spray Penetrations
CR 477114, CDBI Inspections of Intake Structure
CR 477116, Cable Tray Kaowool
CR 477117, CDBI Inspections of Intake Structure
CR 477140, CDBI Inspections of Intake Structure
CR 477164, Unit 1 Core Spray Pump Serial Number Discrepancy
CR 477434, CDBI Quality Records Discrepancy
CR 477441, CDBI Quality Records Discrepancy
CR 477809, Potential for Intake Structure Heat Load to Exceed Design Calculations
CR 478554, Failure to Generate MOV PM Changes to Implement PM Maintenance Strategy
CR 480058, CDBI Broadness Walkdown Inspection
CR 480069, CDBI Broadness Walkdown Inspection

CR 480071, CDBI Broadness Walkdown Inspection CR 481741, NRC CDBI question involving potential reverse flow through PSW and RHRSW pump discharge chk vlv
CR 481404, Receipt of vendor letter for PSW and RHRSW pumps
CR 481670, discrepancy identified in FSAR section 8.5.3 associated with battery discharge test rates
CR 481862, Bolts lock wired together on 1A Diesel Turbocharger
CR 482072, Drawing H-23280 sh 2 typo error
CR 482310, NRC CDBI question regarding inaccurate Tech Spec acceptance criteria
CR 482361, NRC CDBI question regarding Hatch MOV setup procedures
CR 482389, Incorrect information in FSAR
CR 482392, Incorrect FSAR Information
CR 482747, Electrical calculation SENH 10-006 assumption in section 4.6.3 refers to outdated information related
CR 482790, Drawing H13361 error found during CDBI insp CR 482902, NRC CDBI – Discrepancy in stock code applicability to PSW 1P41-D103A strainer gearbox
CR 483145, NRC CDBI – Procedure Enhancement Opportunity
CR 483867, This CR is to investigate an editorial name change in Hatch calculation SENH 10-006 for the buses 20
CR 488720, FSAR discrepancy identified during NRC CDBI inspection
CR 489079, Failure to make required notification identified in NRC CDBI inspection
CR 489789, NRC CDBI question regarding establishing baseline pump flows for IST
CR 490067, This CR is written to review for consistency the micro ohms used in the TRM section T9.1-1 and calcu
CR 490210, NRC CDBI question regarding EDG air start solenoid valves and SBO commitment
CR 490288, Document review during NRC CDBI inspection involving retesting components following test failure
CR 490897, NRC CDBI Inspection identifies issue with SNC Part 21 procedure
CR 490948, Operator Action in PRA needs review
CR 491037, Provide Basis for Discharge Pressure Value Referenced in Unit 2 Surveillance Procedure