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

February 8, 2013

EA-13-015

Mr. Michael Annacone
Vice President
Carolina Power and Light Company
Brunswick Steam Electric Plant
P.O. Box 10429
Southport, NC 28461

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT - NRC INTEGRATED INSPECTION REPORT NOS.: 05000325/2012005 AND 05000324/2012005; AND EXERCISE OF ENFORCEMENT DISCRETION

Dear Mr. Annacone:

On December 31, 2012, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Brunswick Unit 1 and 2 facilities. The enclosed integrated inspection report documents the inspection findings, which were discussed on January 17, 2012 and January 29, 2012 with you and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Two NRC-identified and one self-revealing finding of very low safety significance (Green) were identified during this inspection. Two of these findings were determined to involve a violation of NRC requirements. The NRC is treating these findings as non-cited violations (NCVs) consistent with Section 2.3.2 of the Enforcement Policy.

If you contest the violations or the 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 20555-0001; 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 Brunswick Steam Electric Plant.

If you disagree with the 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 Brunswick Steam Electric Plant.

The enclosed report also documents a noncompliance for which the NRC is exercising enforcement discretion in accordance with Section 9.1 of the NRC Enforcement Policy, "Enforcement Discretion for Certain Fire Protection Issues (10 CFR 50.48)." The noncompliance is associated with your implementation of the requirements and standards of 10 CFR 50, Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979." The noncompliance was identified by your staff, and is a violation of NRC requirements. The inspectors have screened the violation and determined that it warrants enforcement discretion per the Interim Enforcement Policy Regarding Enforcement Discretion for Certain Fire Protection Issues, and Section 11.05(b) of IMC 0305.

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 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/

Richard P. Croteau, Director
Division of Reactor Projects

Docket Nos.: 50-325, 50-324
License Nos.: DPR-71, DPR-62

Enclosure: Inspection Report 05000325, 324/2012005
w/Attachment: Supplemental Information

cc w/encl: (See page 3)

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Richard P. Croteau, Director
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Letter to Michael J. Annacone from Richard P. Croteau dated February 8, 2013.

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-325, 50-324
License Nos.:	DPR-71, DPR-62
Report Nos.:	05000325/2012005, 05000324/2012005
Licensee:	Carolina Power and Light Company
Facility:	Brunswick Steam Electric Plant, Units 1 & 2
Location:	8470 River Road, SE Southport, NC 28461
Dates:	October 1, 2012 through December 31, 2012
Inspectors:	M. Catts, Senior Resident Inspector P. Niebaum, Acting Senior Resident Inspector M. Schwieg, Resident Inspector R. Baldwin, Sr. Operations Engineer (Section 1R11) D. Lanyi, Operators Examiner (Section 1R11) M. Bates, Sr. Operations Engineer (Section 1R11) J. Laughlin, Emergency Preparedness Inspector (Section 1EP4) J. Montgomery, Reactor Inspector (Section 4OA3) R. Rodriguez, Senior Project Engineer (Section 4OA3) A. Nielsen, Sr. Health Physicist (Section 4OA6)
Approved by:	Randall A. Musser, Chief Reactor Projects Branch 4 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000325/2012005, 05000324/2012005; 10/01/12 – 12/31/12; Brunswick Steam Electric Plant, Units 1 & 2; Post Maintenance Testing, Follow-up of Events, and Other Activities.

This report covers a three-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Three Green findings were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, Significance Determination Process (SDP). The cross-cutting aspects were determined using IMC 0310, Components Within the Cross-Cutting Areas. Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. The inspectors identified a Green finding for the licensee not having an adequate procedure for maintenance on fluorescent lights over safety-related equipment. Specifically, between plant startup and August 29, 2012, the licensee did not have instructions for closing S-hooks on fluorescent lights over safety-related equipment during maintenance on the fluorescent lights. This resulted in over 40 S-hooks open in safety-related buildings which could result in fluorescent lights falling and impacting safety-related equipment during a seismic event. The licensee's corrective actions included closing the open S-hooks and adding instructions for closing S-hooks to work order (WO) 431558. The licensee entered this issue into the CAP as NCR 551646.

  The performance deficiency associated with this finding was the failure of the licensee to have an adequate procedure for maintenance on fluorescent lights over safety-related equipment. The finding was more than minor because if left uncorrected, the deficiencies could lead to a more significant safety concern. If left uncorrected, the failure to provide procedural guidance to close the S-hooks on fluorescent lights over safety-related equipment could lead to fluorescent lights falling on safety-related instruments during a seismic event resulting in a reactor trip. This finding is also associated with the design control attribute of the Initiating Events Systems Cornerstone. Using IMC 0609, Appendix A, issued June 19, 2012, The Significance Determination Process (SDP) for Findings At-Power, the inspectors determined the finding was of very low safety significance because the finding did not cause a reactor trip and the loss of mitigation equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition. The finding has a cross-cutting aspect in the area of problem identification and resolution associated with the CAP attribute because the licensee did not identify the open S-hook issue completely, accurately, and in a timely manner commensurate with their safety significance during the Fukushima walkdowns. [P.1(a)] (Section 4OA5)

Cornerstone: Mitigating Systems

- Green. A self-revealing Green NCV of Technical Specification (TS) 5.4.1a, Procedures, was identified because the licensee did not have an adequate maintenance procedure to perform work on the emergency diesel generator (EDG) 3 engine-driven jacket water pump (JWP). Specifically, between July 25, 1992 and November 15, 2012, Procedure 0CM-ENG528, Gould Engine Driven Jacket Water Pump Model 3736, did not provide the correct tolerances for the EDG JWP wear rings, resulting in the JWP seizure. The licensee's corrective actions included replacing the casing wear rings with wear rings with the correct tolerance and revising Procedure 0CM-ENG528. The licensee entered this issue into the corrective action program (CAP) as nuclear condition report (NCR) 572546.

  The performance deficiency associated with this finding was the failure of the licensee to have an adequate procedure for maintenance on the EDG 3 engine-driven JWP. The finding was more than minor because it was associated with the procedure quality attribute of the Mitigating Systems Cornerstone and affects the cornerstone objective of ensuring the availability and reliability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the inadequate procedure resulted in reduced availability of EDG 3 to repair the engine-driven JWP and reduced reliability of the jacket water system during operation. Using IMC 0609, Appendix A, issued June 19, 2012, The Significance Determination Process (SDP) for Findings At-Power, the inspectors determined the finding was of very low safety significance because the finding did not affect the design or qualification of a mitigating structure, system and component (SSC), the finding did not represent a loss of system and/or function, the finding did not represent an actual loss of a function of a single train for greater than the TS allowed outage time, the finding did not represent an actual loss of a function of one or more non-TS trains of equipment, and did not screen as potentially risk-significant due to a seismic, flooding, or severe weather initiating event. The finding does not have a cross-cutting aspect since the performance deficiency is not indicative of current plant performance. Procedure 0CM-ENG528 included the incorrect tolerances since July 25, 1992. (Section 1R19)

- Green. The inspectors identified a Green NCV of 10 CFR 50 Appendix B, Criterion III, Design Control, for failure to assure that the design basis for EDG 2 Alternate Safe Shutdown (ASSD) Switch A1 was correctly translated into specifications and drawings. Specifically, between original EDG 2 installation and September 1, 2012, a wiring discrepancy existed associated with EDG 2 ASSD Switch A1 which resulted in an induced fault that could have impacted the ability to locally control EDG 2 during certain fire scenarios. The licensee's corrective actions included correcting the EDG 2 control circuit wiring to ensure it was in accordance with the existing approved design and returning EDG 2 to operable status. The licensee entered this issue into the CAP as NCR 557897.

  The performance deficiency associated with this finding was the failure to assure that the design basis for EDG 2 ASSD Switch A1 was correctly translated into specifications and drawings. The finding was more than minor because it was associated with the protection against external factors (i.e. fire) attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, an induced fault could have impacted the ability to locally control EDG 2 during certain fire scenarios. Using IMC 0609, Attachment 4, issued June 19, 2012, Initial Characterization of Findings, and IMC 0609, Appendix F, Attachment 1, Part 1: Application of Fire Protection SDP Phase 1 Worksheet, the results of this evaluation required further significance evaluation. A phase 3 analysis was performed by a regional SRA in accordance with NRC IMC 0609 Appendix F. The finding affected the capability to achieve alternate safe shutdown for Unit 1. The result of the analysis was an increase in core damage frequency of <1E-6/year a GREEN finding of very low safety significance. The finding does not have a cross-cutting aspect since the performance deficiency is not indicative of current plant performance. The EDG 2 ASSD Switch A1 wiring discrepancy has existed since original EDG installation. (Section 4OA2)

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status

Unit 1 began the inspection period at 100 percent rated thermal power (RTP). On November 4, 2012, power was reduced to 70 percent for a rod sequence exchange. Power was returned to RTP on November 5, 2012. On November 6, 2012, power was reduced to 72 percent for a rod improvement. Power was returned to RTP on November 7, 2012 and stayed for the remainder of the inspection period.

Unit 2 began the inspection period at 100 percent RTP. On October 8, 2012, power was reduced to approximately 62 percent due to a feedwater heater level control valve failure. Following the repair, power ascension commenced to 100 percent RTP on October 10, 2012. On November 17, 2012, the licensee commenced a power reduction to 70 percent for a rod sequence exchange, turbine valve testing, a planned 230kV Corning Transmission Line outage, and corrective maintenance on the Generator Exciter rectifier bank assemblies. On November 19, 2012, power was reduced to approximately 60 percent due to elevated temperatures associated with the generator No Load Disconnect Switch (NLDS). On November 22, 2012, power was further reduced to approximately 22 percent to take the turbine off-line to make repairs to the NLDS. On November 24, 2012, the turbine was placed back on-line following repairs to the NLDS, but was subsequently taken off-line again due to a steam leak in the 'A' Feedwater Heater room. On November 26, 2012, the turbine was placed on-line and the main generator synchronized with the grid and power ascension began to approximately 60 percent. Power ascension to 100 percent commenced on November 29, 2012, following repairs due to water intrusion in the 2A reactor feed pump control cabinet. On December 12, 2012, power was reduced to 63 percent to conduct power suppression testing. Power was returned to 94 percent on December 13, 2012. Power was reduced to 69 percent for a rod improvement and returned to RTP on December 17, 2012. On December 18, 2012, power was reduced to 79 percent for another rod improvement. On December 19, 2012, power was returned to RTP and stayed for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01 – 2 samples)

.1 Winter Seasonal Readiness Preparations

a. Inspection Scope

The inspectors conducted a review of the licensee's preparations for winter conditions to verify that the plant's design features and implementation of procedures were sufficient to protect mitigating systems from the effects of adverse weather. Documentation for selected risk-significant systems was reviewed to ensure that these systems would remain functional when challenged by inclement weather. During the inspection, the inspectors focused on plant-specific design features and the licensee's procedures used to mitigate or respond to adverse weather conditions. Additionally, the inspectors reviewed the Updated Final Safety Analysis Report (UFSAR) and performance requirements for systems selected for inspection, and verified that operator actions were appropriate as specified by plant-specific procedures. Cold weather protection, such as heat tracing and area heaters, was verified to be in operation where applicable. The inspectors also reviewed CAP items to verify that the licensee was identifying adverse weather issues at an appropriate threshold and entering them into the CAP in accordance with station corrective action procedures. The inspectors' reviews focused specifically on the following plant systems due to their risk significance or susceptibility to cold weather issues:

- Freeze protection for the diesel fire pump; and
- Freeze protection for service water pumps

b. Findings

No findings were identified.

.2 Readiness for Impending Adverse Weather Condition

a. Inspection Scope

On October 25, 2012, a Tropical Storm Advisory was issued for the plant area due to the potential path of Hurricane Sandy. The inspectors reviewed the licensee's overall preparations/protection for impending adverse weather conditions. The inspectors performed walkdowns of plant areas susceptible to high winds, including the licensee's emergency alternating current (AC) power systems. The inspectors evaluated the licensee staff's preparations against the site's procedures. During the inspection, the inspectors focused on plant-specific design features and the licensee's procedures used to respond to specified adverse weather conditions. The inspectors also toured the plant grounds to look for any loose debris that could become missiles during the periods of high winds anticipated on site. The inspectors also reviewed a sample of CAP items to verify that the licensee identified adverse weather issues at an appropriate threshold and dispositioned them through the CAP in accordance with station corrective action procedures.

The inspectors reviewed the inspection guidance in Operating Experience Smart Sample (OpESS) 2012/01, High Wind Generated Missile Hazards, to verify adequate protection of equipment and outside structures from high winds and high wind generated missiles.

b. Findings

No findings of were identified.

1R04 Equipment Alignment (71111.04Q – 3 samples)

Quarterly Partial System Walkdowns

a. Inspection Scope

The inspectors performed three partial system walkdowns of the following risk-significant systems:

- EDG 4 while EDG 3 was out of service for corrective maintenance on October 16, 2012;
- Unit 2 Reactor Building Closed Cooling Water System on November 8, 2012; and
- Unit 2 High Pressure Coolant Injection (HPCI) on December 4, 2012.

The inspectors selected these systems based on their risk-significance relative to the reactor safety cornerstones at the time they were inspected. The inspectors attempted to identify any discrepancies that could impact the function of the system, and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, the UFSAR, TS requirements, outstanding work orders, condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also walked down accessible portions of the systems to verify that system components and support equipment were aligned correctly and were operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the CAP with the appropriate significance characterization.

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05Q – 5 samples)

.1 Quarterly Resident Inspector Tours

a. Inspection Scope

The inspectors conducted five fire protection walkdowns which were focused on availability, accessibility, and the condition of firefighting equipment in the following risk-significant plant areas:

- E1-E4 Switchgear Room 50' Elevation 1PFP-DG-11/14;
- Unit 2 Reactor Building North-South 20' Elevation 2PFP-RB2-1g N/S;
- Unit 1 Reactor Building West 80’ Elevation 1PFP-RB1-1j W;

- E5-6 Switchgear Room 23' Elevation 1PFP-DG-6/7; and
- E7-8 Switchgear Room 23' Elevation 2PFP-DG-8/9.

The inspectors reviewed areas to assess if the licensee had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant, effectively maintained fire detection and suppression capability, maintained passive fire protection features in good material condition, and had implemented adequate compensatory measures for out-of-service, degraded or inoperable fire protection equipment, systems, or features in accordance with the licensee's fire plan. The inspectors selected fire areas based on their overall contribution to internal fire risk as documented in the plant's Individual Plant Examination of External Events, including their potential to impact equipment which could initiate or mitigate a plant transient, or their impact on the plant's ability to respond to a security event. Using the documents listed in the attachment, the inspectors verified that fire hoses and extinguishers were in their designated locations and available for immediate use; that fire detectors and sprinklers were unobstructed; that transient material loading was within the analyzed limits; and fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors also verified that minor issues identified during the inspection were entered into the licensee's CAP.

b. Findings

No findings were identified.

1R07 Heat Sink Performance (71111.07A – 1 sample)

a. Inspection Scope

The inspectors reviewed the licensee's testing of the EDG 1 Jacket Water Heat Exchanger to verify that potential deficiencies did not mask the licensee's ability to detect degraded performance, to identify any common cause issues that had the potential to increase risk, and to ensure that the licensee was adequately addressing problems that could result in initiating events that would cause an increase in risk. The inspectors reviewed the licensee's observations as compared against acceptance criteria, the correlation of scheduled testing and the frequency of testing, and the impact of instrument inaccuracies on test results. Inspectors also visually inspected the service water side of the heat exchange to ensure that the heat exchanger was free of debris and biological growth.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program

.1 Quarterly Review of Licensed Operator Requalification Testing and Training (71111.11Q – 1 sample)

a. Inspection Scope

On November 6, 2012, the inspectors observed a crew of licensed operators in the plant's simulator during an Emergency Preparedness (EP) drill to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems, and to ensure that training, where appropriate, was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

- licensed operator performance;
- crew's clarity and formality of communications;
- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms;
- correct use and implementation of abnormal and emergency procedures;
- control board manipulations;
- oversight and direction from supervisors; and
- ability to identify and implement appropriate TS actions and EP actions and notifications.

The crew's performance in these areas was compared to pre-established operator action expectations and successful critical task completion requirements.

b. Findings

No findings were identified.

.2 Quarterly Review of Licensed Operator Performance in the Main Control Room (71111.11Q – 1 sample)

a. Inspection Scope

Inspectors observed and assessed licensed operator performance in the plant and main control room, particularly during periods of heightened activity or risk and where the activities could affect plant safety. Specifically, on November 23, 2012, the inspectors observed Unit 2 evolutions following the main turbine trip to repair the main generator no load disconnect switch (NLDS). The inspectors reviewed various licensee policies and procedures listed in the Attachment. The inspectors evaluated the following areas:

- operator compliance and use of procedures;
- control board manipulations;
- communication between crew members;
- use and interpretation of plant instruments, indications and alarms;

- use of human error prevention techniques;
- documentation of activities, including initials and sign-offs in procedures;
- supervision of activities, including risk and reactivity management; and
- pre-job briefs and crew briefs.

b. Findings

No findings were identified.

.3 Annual Review of Licensee Requalification Examination Results (71111.11A – 1 sample)

a. Inspection Scope

On December 19, 2012, the licensee completed the annual requalification operating examinations required to be administered to all licensed operators in accordance with 10 CFR 55.59(a)(2). The inspectors performed an in-office review of the overall pass/fail results of the individual operating examinations and the crew simulator operating examinations in accordance with Inspection Procedure (IP) 71111.11, Licensed Operator Requalification Program. These results were compared to the thresholds established in IMC 0609, Significance Determination Process, Appendix I, Operator Requalification Human Performance Significance Determination Process.

b. Findings

No findings were identified.

.4 Licensed Operator Requalification (71111.11B – 1 sample)

a. Inspection Scope

The inspectors reviewed the facility operating history and associated documents in preparation for this inspection. During the week of November 26, 2012 the inspectors reviewed documentation, interviewed licensee personnel, and observed the administration of operating tests associated with the licensee's operator requalification program. Each of the activities performed by the inspectors was done to assess the effectiveness of the facility licensee in implementing requalification requirements identified in 10 CFR Part 55, "Operators' Licenses." The evaluations were also performed to determine if the licensee effectively implemented operator requalification guidelines established in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," and Inspection Procedure 71111.11, "Licensed Operator Requalification Program." The inspectors also evaluated the licensee's simulation facility for adequacy for use in operator licensing examinations using ANSI/ANS-3.5-1998, "American National Standard for Nuclear Power Plant Simulators for use in Operator Training and Examination." The inspectors observed two crews during the performance of the operating tests. Documentation reviewed included written examinations, Job Performance Measures (JPMs), simulator scenarios, licensee procedures, on-shift records, simulator modification request records, simulator performance test records, operator feedback records, licensed operator qualification records, remediation plans, watchstanding records, and medical records. The records were inspected using the criteria listed in Inspection Procedure 71111.11. Documents reviewed during the inspection are documented in the List of Documents Reviewed.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12Q – 2 samples)

a. Inspection Scope

The inspectors evaluated two degraded performance issues involving the following risk-significant systems:

- Unit 2 2B Screen Water Pump failure due a strainer leak on October 23, 2012; and
- Units 1 and 2 potential clogging of floor drain sock filters in safety-related pump rooms on November 24, 2012.

The inspectors reviewed events where ineffective equipment maintenance may have resulted in equipment failure or invalid automatic actuations of Engineered Safeguards Systems, and independently verified the licensee's actions to address system performance or condition problems in terms of the following:

- implementing appropriate work practices;
- identifying and addressing common cause failures;
- scoping of systems in accordance with 10 CFR 50.65(b) of the maintenance rule;
- characterizing system reliability issues for performance;
- charging unavailability for performance;
- trending key parameters for condition monitoring;
- ensuring 10 CFR 50.65(a)(1) or (a)(2) classification or re-classification; and
- verifying appropriate performance criteria for SSCs/functions classified as (a)(2) or appropriate and adequate goals and corrective actions for systems classified as (a)(1).

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the system. In addition, the inspectors verified maintenance effectiveness issues were entered into the CAP with the appropriate significance characterization.

b. Findings

No findings were identified.

(Opened) Unresolved Item (URI) 05000325/2012005-01 and 05000324/2012005-01, Floor Drains Not Functioning Due to Plugging

Introduction. The inspectors are opening an URI to review the licensee's evaluation of the potential for adverse impact due to floor drain sock filter plugging in safety-related pump rooms and determine if there is a performance deficiency.

Description. On November 24, 2012, during a steam leak in the 2A Feedwater Heater Room, water did not adequately drain from the room through the floor drains due to plugging in the floor drain sock filters. The licensee's immediate corrective actions included removing the sock filters so that the water could drain. The sock filters are also installed in safety-related pump rooms in the reactor building. The inspectors are opening an URI to review the licensee's evaluation of the potential for adverse impact due to drain plugging in safety-related pump rooms and determine if there is a performance deficiency. The licensee entered this issue in the CAP as NCR 574261. This issue is being tracked as a URI: **URI 05000325/2012005-01 and 05000324/2012005-01, Floor Drains Not Functioning Due to Plugging.**

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13 – 4 samples)

a. Inspection Scope

The inspectors reviewed the licensee's evaluation and management of plant risk for the four maintenance and emergent work activities affecting risk-significant equipment listed below to verify that the appropriate risk assessments were performed prior to removing equipment for work:

- Unit 1 elevated Yellow risk during Residual Heat Removal and Core Spray low pressure trip calibration test on October 10, 2012;
- Unit 2 elevated risk during scheduled maintenance on EDG 3 for a system outage and scheduled work on the 2A Conventional Service Water pump to replace discharge piping on November 14, 2012;
- Unit 2 elevated risk during a power reduction to approximately 22 percent to take the turbine off-line to make repairs to the NLDS on November 22, 2012; and
- Unit 2 elevated risk during power maneuvers following repairs to the steam leak in the 'A' Feedwater Heater room, and the 2A reactor feedwater pump (RFP) control cabinet on November 26, 2012.

These activities were selected based on their potential risk-significance relative to the reactor safety cornerstones. As applicable for each activity, the inspectors verified that risk assessments were performed as required by 10 CFR 50.65(a)(4) and were accurate and complete. When emergent work was performed, the inspectors verified that the plant risk was promptly reassessed and managed. The inspectors reviewed the scope of maintenance work, discussed the results of the assessment with the licensee's probabilistic risk analyst or shift technical advisor, and verified plant conditions were consistent with the risk assessment. The inspectors also reviewed TS requirements and walked down portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met.

b. Findings

No findings were identified.

1R15 Operability Evaluations (71111.15 – 6 samples)

a. Inspection Scope

The inspectors reviewed the following six issues:

- Failure of the Unit 1 1A Conventional Service Water Pump 4kV breaker on October 11, 2012;
- High humidity conditions on the Unit 1 ‘B’ Standby Gas Treatment on October 15, 2012;
- Calibration problems with the EDG 1 recirculation and exhaust dampers on October 31, 2012;
- 2A Standby Liquid Control Pump accumulator low pressure on November 20, 2012;
- Diesel Fuel Oil Testing Change to TS Bases 3.8.3 and 3.8.3.2 on December 17, 2012; and
- Missed Surveillance Requirement 3.6.1.3.1 on May 18, 2012, 0PT-02.2.4A, Primary Containment Integrity Verification – Containment External, reviewed on December 18, 2012.

The inspectors selected these potential operability issues based on the risk-significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure that TS operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the TS and the UFSAR to the licensee’s evaluations, to determine whether the components or systems were operable. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations. Additionally, the inspectors also reviewed a sampling of corrective action documents to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations.

The inspectors reviewed the inspection guidance in Operating Experience Smart Sample 2012/02, Revision 1, Technical Specification Interpretation and Operability Determination, to verify conservative decision-making and proper application of TS Limiting Condition of Operation and Action requirements.

b. Findings

No findings were identified.

1R18 Plant Modifications (71111.18 – 1 sample)

a. Inspection Scope

The inspectors reviewed the modifications listed below to determine whether the modifications affected the safety functions of systems that are important to safety. The inspectors reviewed 10 CFR 50.59 documentation and post-modification testing results, and conducted field walkdowns of the modifications to verify that the modifications did not degrade the design bases, licensing bases, and performance capability of the affected systems.

- EDG 4 reliability upgrade - starting air cross-tie

b. Findings

No findings were identified.

1R19 Post Maintenance Testing (71111.19 – 6 samples)

a. Inspection Scope

The inspectors reviewed the following six post-maintenance activities to verify that procedures and test activities were adequate to ensure system operability and functional capability:

- 0PT-07.2.4a, Core Spray System Operability Test – Loop A after planned maintenance on October 6, 2012;
- 0PT-12.2C, No. 3 Diesel Generator Monthly Load Test after repairs to starting air on October 17, 2012;
- 0PT-12.2A, No. 1 Diesel Generator Monthly Load Test after a planned maintenance outage on November 2, 2012;
- 1OP-08, Control Drive Rod Hydraulic System Operating Procedure after replacement of HCU 06-23 accumulator on November 5, 2012;
- 0PT-12.2C, No. 3 Diesel Generator Monthly Load Test after replacement of engine driven JWP on November 16, 2012; and
- Unit 2 NLDS thermal scans after repairs on November 23, 2012.

These activities were selected based upon the SSC’s ability to impact risk. The inspectors evaluated these activities for the following: the effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed; acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate; tests were performed as written in accordance with properly reviewed and approved procedures; equipment was returned to its operational status following testing; and test documentation was properly evaluated. The inspectors evaluated the activities against TSs and the UFSAR to verify that the test results adequately ensured that the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed corrective action documents associated with post-maintenance tests to determine whether the licensee was identifying problems and entering them in the CAP, and that the problems were being corrected commensurate with their importance to safety.

b. Findings

.1 (Opened) Unresolved Item (URI) 05000325/2012005-02 and 05000324/2012005-02, Emergency Diesel Generator 3 Slow Start

Introduction. The inspectors are opening an URI to review the licensee's evaluation of the cause of the EDG failure and determine if there is a performance deficiency.

Description. On October 14, 2012, the licensee was running EDG 3 for a zero oil pressure start test in accordance with Procedure 0PT-12.2.c, No. 3 Diesel Generator Monthly Load Test. The EDG reached rated speed at approximately 38 seconds after the EDG was started and then tripped. Surveillance Requirement 3.8.1.7 requires the EDG reach rated conditions within 10 seconds. Several seconds after reaching rated speed, the EDG began to coast down due to receiving a lockout signal since full rated conditions were not achieved within the nominal time delay of 45 seconds. The licensee replaced the overspeed start emergency boost cylinder and declared the EDG operable on October 17, 2012. The inspectors are opening an URI to review the licensee's evaluation of the cause of the EDG failure and determine if there is a performance deficiency. The licensee entered this issue in the CAP as NCR 567016. This issue is being tracked as a URI: **URI 05000325/2012005-02 and 05000324/2012005-02, Emergency Diesel Generator 3 Slow Start.**

.2 Inadequate Maintenance Procedure for the EDG Jacket Water Pump Wear Ring Tolerances

Introduction. A self-revealing Green NCV of TS 5.4.1a, Procedures, was identified because the licensee did not have an adequate maintenance procedure to perform work on the EDG 3 engine-driven JWP. Specifically, between July 25, 1992 and November 15, 2012, Procedure 0CM-ENG528, Gould Engine Driven Jacket Water Pump Model 3736, did not provide the correct tolerances for the EDG JWP wear rings, resulting in the JWP seizure.

Description. On August 12, 2012, operations personnel reported a mechanical seal leak from the engine-driven JWP on EDG 3. Per Procedure 0OP-39, Diesel Operating Procedure, EDG jacket water leakage is limited to 150 drops per minute, and if leakage is greater than 150 drops per minute, operability should be verified by confirming that the leak is less than 15 ml per minute. The licensee evaluated the leakage, determined the leakage exceeded the operability limit, and declared EDG 3 inoperable. Work order (WO) 02124254-01 was issued to disassemble and repair the EDG 3 engine-driven

JWP. The WO referenced Procedure 0CM-ENG528, Gould Engine Driven Jacket Water Pump Model 3736, for disassembly, inspection and repair of the JWP. During the pump disassembly, it was found the casing wear rings were out of tolerance and required replacement. The replacement wear rings were machined to within tolerance and installed in accordance with Procedure 0CM-ENG528. The repair and testing was completed on August 16, 2012 and EDG 3 was returned to service.

On October 15, 2012, during a maintenance run, the motor driven JWP did not turn off as expected due to the engine-driven JWP not providing sufficient discharge pressure to reach the shutoff permissive for the motor driven pump. The jacket water system has two 100 percent capacity pumps, a motor driven and an engine-driven pump. Operations personnel declared the EDG operable but degraded because although the engine-driven JWP was not functioning as designed, the motor driven JWP was fully capable of supplying sufficient jacket water to the EDG. On November 12, 2012, during the EDG 3 outage, it was discovered that the engine-driven JWP had failed due to the pump impeller seizing and the impeller bolt sheering.

The inspectors reviewed the engineering evaluation (EC89436) which determined the most probable cause was insufficient clearance in the engine-driven JWP casing wear rings. The manufacturer, Gould, recommends a tolerance based on the material type (i.e. cast iron or stainless steel). The replacement wear rings were stainless steel which required a tolerance of 0.023" to 0.027" per the manufacturer recommendation. However, Procedure 0CM-ENG528 did not specify a material type and used the tolerance for cast iron wear rings, a tolerance of 0.018" to 0.022". As a result, the replacement wear rings were machined to a tighter tolerance than the manufacturer recommendation. This resulted in excessive galling on wear ring surfaces and eventual failure of the engine-driven JWP. The licensee replaced the casing wear rings with wear rings with the correct tolerance and revised Procedure 0CM-ENG528. The licensee entered this issue into the CAP as NCR 572546. The other diesels (EDG 1, 2, 4) were not affected by this procedure because the last (and only) JWP replacement was in 1990 (EDG3).

Analysis. The performance deficiency associated with this finding was the failure of the licensee to have an adequate procedure for maintenance on the EDG 3 engine-driven JWP. The finding was more than minor because it was associated with the procedure quality attribute of the Mitigating Systems Cornerstone and affects the cornerstone objective of ensuring the availability and reliability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the inadequate procedure resulted in reduced availability of EDG 3 to repair the engine-driven JWP and reduced reliability of the jacket water system during operation. Using IMC 0609, Appendix A, issued June 19, 2012, The Significance Determination Process (SDP) for Findings At-Power, the inspectors determined the finding was of very low safety significance because the finding did not affect the design or qualification of a mitigating SSC, the finding did not represent a loss of system and/or function, the finding did not represent an actual loss of a function of a single train for greater than the TS allowed outage time, the finding did not represent an actual loss of a function of one or more non-TS trains of equipment, and did not screen as potentially risk-significant due to a seismic, flooding, or severe weather initiating event. The finding does not have a cross-cutting aspect since the performance deficiency is not indicative of current plant performance. Procedure 0CM-ENG528 included the incorrect tolerances since July 25, 1992.

Enforcement. Technical Specification 5.4.1a, Procedures, states that "written procedures shall be established, implemented, and maintained covering the following activities: a) the applicable procedures recommended in Regulatory Guide 1.33, Appendix A, November 1972 (Safety Guide 33, November 1972). Regulatory Guide 1.33, Appendix A, November 1972 (Safety Guide 33, November 1972), Section I, states that "maintenance which can affect the performance of safety-related equipment should be properly preplanned and performed in accordance with written procedures, documented instructions, or drawings appropriate to the circumstance." Procedure 0CM-ENG528, Gould Engine Driven Jacket Water Pump Model 3736, Step 7.2.35.7, provides the acceptance criteria for new JWP impeller wear ring clearances. Contrary to the above, between July 25, 1992 and November 15, 2012, the licensee failed to properly establish and maintain Procedure 0CM-ENG528. Specifically, Procedure 0CM-ENG528, Step 7.2.35.7 did not provide the correct tolerances for EDG 3 JWP wear rings. This resulted in the JWP seizure and EDG 3 unavailability to replace the pump. The licensee's corrective actions included replacing the EDG 3 engine-driven JWP engine casing wear rings and revising the procedure to include the correct wear ring tolerances. Because this finding is of very low safety significance and was entered into the licensee's CAP as NCR 572546, consistent with Section 2.3.2 of the NRC's Enforcement Policy, this violation is being treated as a NCV: **NCV 05000325/2012005-03 and 05000324/2012005-03, Inadequate Maintenance Procedure for the EDG Jacket Water Pump Wear Ring Tolerances**.

1R22 Surveillance Testing

.1 Routine Surveillance Testing (71111.22 – 3 surveillance test samples)

a. Inspection Scope

The inspectors either observed surveillance tests or reviewed the test results for the following activities to verify the tests met TS surveillance requirements, UFSAR commitments, in-service testing requirements, and licensee procedural requirements. The inspectors assessed the effectiveness of the tests in demonstrating that the SSCs were operationally capable of performing their intended safety functions.

- 0GP-02, Source Range Monitor / Intermediate Range Monitor overlap on October 1, 2012;
- 0MST-RHR26Q, Residual Heat Removal / Core Spray Low Reactor Pressure Permissive on October 11, 2012; and
- 2MST-DG22R, DG-4 Trip Bypass Logic Test on October 22, 2012.

b. Findings

No findings were identified.

.2 In-Service Testing (IST) Surveillance (71111.22 – 1 IST sample)

a. Inspection Scope

The inspectors reviewed the performance of the Unit 2 Reactor Core Isolation Cooling system (RCIC) operability test on November 25, 2012 to evaluate the effectiveness of the licensee's American Society of Mechanical Engineers Section XI testing program for determining equipment availability and reliability. The inspectors evaluated selected portions of the following areas: 1) testing procedures; 2) acceptance criteria; 3) testing methods; 4) compliance with the licensee's IST program, TS, selected licensee commitments, and code requirements; 5) range and accuracy of test instruments; and 6) required corrective actions.

b. Findings

No findings were identified.

Cornerstone: Emergency Preparedness

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04 – 1 sample)

a. Inspection Scope

The Office of Nuclear Security and Incident Response (NSIR) headquarters staff performed an in-office review of the latest revisions of various Emergency Plan Implementing Procedures and the Emergency Plan located under ADAMS accession numbers ML 12096A029, ML 12177A270, ML 12180A514 and ML 12194A109 as listed in the Attachment.

The licensee determined that in accordance with 10 CFR 50.54(q), the changes made in the revisions resulted in no reduction in the effectiveness of the Plan, and that the revised Plan continued to meet the requirements of 10 CFR 50.47(b) and Appendix E to 10 CFR Part 50. The NRC review was not documented in a safety evaluation report and did not constitute approval of licensee-generated changes; therefore, these revisions are subject to future inspection. The specific documents reviewed during this inspection are listed in the Attachment. This inspection activity satisfied one inspection sample for the emergency action level and emergency plan changes on an annual basis.

b. Findings

No findings were identified.

1EP6 Emergency Planning Drill Evaluation (71114.06 – 1 sample)

a. Inspection Scope

The inspectors observed a site emergency preparedness drill conducted on November 6, 2012. The inspectors reviewed the drill scenario narrative to identify the timing and location of classifications, notifications, and protective action recommendations. During the drill, the inspectors assessed the adequacy of event classification and notification activities. The inspectors observed portions of the licensee's post-drill critiques and evaluations. The inspectors verified that the licensee properly evaluated the drill performance with respect to performance indicators and assessed drill performance with respect to drill objectives.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification (71151 – 4 samples)

Mitigating Systems Cornerstone

a. Inspection Scope

- Unit 1 Mitigating Systems Performance Index, Heat Removal System, RCIC
- Unit 2 Mitigating Systems Performance Index, Heat Removal System, RCIC
- Unit 1 Mitigating Systems Performance Index, High Pressure Injection Systems, HPCI
- Unit 2 Mitigating Systems Performance Index, High Pressure Injection Systems, HPCI

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index (MSPI) performance indicators listed above for the period from the fourth quarter of 2011 through the third quarter of 2012. The inspectors reviewed the licensee's operator narrative logs, issue reports, MSPI derivation reports, event reports and NRC Integrated Inspection reports for the period to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator.

b. Findings

No findings were identified.

4OA2 Identification and Resolution of Problems

.1 Routine Review of Items Entered Into the Corrective Action Program

a. Inspection Scope

To aid in the identification of repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed frequent screenings of items entered into the licensee's CAP. The review was accomplished by reviewing daily action request reports.

b. Findings

No findings were identified.

.2 Selected Issue Follow-up Inspection: Degraded / Nonconforming Condition Tracking (71152 – 1 sample)

a. Inspection Scope

The inspectors selected the list of degraded / nonconforming conditions for detailed review. This list tracks the resolution of issues classified as degraded / nonconforming in accordance with NRC Regulatory Information Summary 2005-20, Revision to Guidance Formerly Contained in NRC Generic Letter 91-18, Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability. The inspectors reviewed a sample of the NCRs included in the list to ensure the equipment was appropriately classified as degraded / nonconforming and that the corrective actions were scheduled for the first available opportunity. The inspectors evaluated the list against the requirements of the licensee's CAP as delineated in corporate procedure CAP-NGGC-0200, Corrective Action Program.

b. Findings

No findings were identified.

.3 Semi-Annual Trend Review (71152 – 1 sample)

a. Inspection Scope

The inspectors performed a review of the licensee's corrective action program and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment issues, but also considered the results of daily inspector CAP item screening discussed in Section 4OA2.1 above, licensee trending efforts, and licensee human performance results. The inspectors' review nominally considered the six-month period of

July 1, 2012, through December 31, 2012, although some examples expanded beyond those dates where the scope of the trend warranted.

Inspectors also reviewed major equipment problem lists, repetitive and rework maintenance lists, departmental problem/challenges lists, system health reports, quality assurance audit/surveillance reports, self-assessment reports, and Maintenance Rule assessments. The inspectors compared and contrasted their results with the results contained in the licensee's CAP trending reports. Corrective actions associated with a sample of the issues identified in the licensee's trending reports were reviewed for adequacy.

b. Findings and Observations

No findings were identified.

The inspectors evaluated a sample of departments that are required to provide input into the quarterly trend reports, which included self-assessment and system engineering departments. This review included a sample of issues and events that occurred over the course of the past two quarters to determine whether issues were appropriately considered or ruled as emerging or adverse trends, and in some cases, verified the appropriate disposition of resolved trends. The inspectors verified that these issues were addressed within the scope of the CAP, or through department review and documentation in the quarterly trend report for overall assessment. For example, the inspectors noted that consistent with the onset of an adverse trend in NCR generation, the licensee appropriately identified this trend as an open adverse trend with ongoing corrective actions to address this issue.

The inspectors identified an adverse trend in procedural adequacy. Examples of procedural adequacy include:

- Procedure 0OI-01.01, BNP Conduct of Operations Supplement, incorrectly stated leakage through mechanical joints (for example valve or flange packing leaks, seat leakage through an isolation valve, flange leakage, etc) is not considered an operation with potential for draining the reactor vessel (OPDRV). This resulted in the inoperability of secondary containment during an OPDRV as documented in Inspection Report 05000325/2012004 and 05000324/2012004 Section 71111.20 (NCR 562188);
- Procedure 0CM-ENG528, Gould Engine Driven Jacket Water Pump Model 3736, did not provide the correct tolerances for the EDG JWP wear rings as documented in Section 1R19 (NCR 572546);
- Work Order 431558 did not include instructions for closing S-hooks on fluorescent lights over safety-related equipment during maintenance on the fluorescent lights as documented in Section 4OA5 (NCR 551646).

The licensee entered this trend into the CAP as NCR 581194.

.4 Annual Sample: Review of Operator Workarounds (OWAs) (71152 – 1 sample)

a. Inspection Scope

The inspectors evaluated the licensee's implementation of their process used to identify, document, track, and resolve operational challenges. Inspection activities included, but were not limited to, a review of the cumulative effects of the OWAs on system availability and the potential for improper operation of the system, for potential impacts on multiple systems, and on the ability of operators to respond to plant transients or accidents. The inspectors performed a review of the cumulative effects of OWAs. The documents listed in the attachment were reviewed to accomplish the objectives of the inspection procedure. The inspectors reviewed both current and historical operational challenge records to determine whether the licensee was identifying operator challenges at an appropriate threshold, had entered them into the CAP, and proposed or implemented appropriate and timely corrective actions which addressed each issue. Reviews were conducted to determine if any operator challenge could increase the possibility of an Initiating Event, if the challenge was contrary to training, required a change from long-standing operational practices, or created the potential for inappropriate compensatory actions. Daily plant and equipment status logs, degraded instrument logs, and operator aids or tools being used to compensate for material deficiencies were also assessed to identify any potential sources of unidentified operator workaround.

The inspectors reviewed the inspection guidance in Operating Experience Smart Sample 2012/02, Revision 1, Technical Specification Interpretation and Operability Determination, to verify that compensatory measures, which substitute manual operator actions for automatic actions, were adequate to ensure plant safety.

b. Findings

No findings were identified.

.5 (Closed) Unresolved Item (URI) 5000325, 324/2012004-03, EDG2 Wiring on Alternate Safe Shutdown Switch

a. Inspection Scope

The inspectors completed an evaluation of URI 05000325; 324/2012004 for a wiring discrepancy on EDG 2 ASSD Switch A1. A contact in the circuit was determined to be bypassed that would have the potential to prevent proper isolation of the EDG 2 control circuits from the Main Control Room during certain fire scenarios. A wire was found installed in the EDG control panel, where the ASSD switches are located, that was not found on the plant drawings. The issue was discovered during a planned system outage for EDG 2 on August 28, 2012. This URI was opened to determine if a performance deficiency existed.

The inspectors performed a review of the licensee's quick cause evaluation documented in NCR 557897 and Licensee Event Report (LER) 05000325, 324/2012-005-00, Local

Control Capability of Emergency Diesel Generator No. 2 Not Available. The inspectors also reviewed the licensee's reportability evaluation and subsequent 8-hour report made to the NRC in accordance with 10 CFR 50.72(b)(3)(ii)(B). The inspectors discussed this issue with licensee personnel to understand the potential impacts and fire scenarios of concern.

b. Findings

An inspector identified violation was identified. This URI is closed.

Introduction. The inspectors identified a Green NCV of 10 CFR 50 Appendix B, Criterion III, Design Control, for failure to assure that the design basis for EDG 2 Alternate Safe Shutdown (ASSD) Switch A1 was correctly translated into specifications and drawings. Specifically, between original EDG 2 installation and September 1, 2012, a wiring discrepancy existed associated with EDG 2 ASSD Switch A1 which resulted in an induced fault that could have impacted the ability to locally control EDG 2 during certain fire scenarios.

Description. On August 28, 2012, a wiring discrepancy was identified that could affect the ability to locally control EDG 2 during a fire event. A post-maintenance continuity test on ASSD Switch A1 revealed unexpected results when the switch was taken to the Local position. The post maintenance test consisted of continuity checks across remote terminal points in the EDG 2 control panel with the A1 Switch in the Normal and Local positions. When continuity checks were made across the 2-2C contact, with the switch in Local position (contacts open), the switch appeared to be closed. After subsequent troubleshooting, the licensee found a wire incorrectly installed between terminal points F-6 and B-64 that effectively jumpered around the switch contacts. The licensee determined that this wire, not identified in wiring diagram F-09346, Diesel Generator No. 2 Circuits Control Wiring Diagram, created a short across ASSD Switch A1 contact 2-2C, which prevented isolation of the control room circuit for control of EDG 2. The function of ASSD Switch A1 is to flash the field on a diesel start signal.

The licensee originally concluded that there was no plausible hot short in the control building that could impact the EDG 2 ASSD function. The inspectors challenged the licensee on this determination. Due to the inspectors' questions, the licensee had a contractor assist in performing an evaluation to determine if any hot shorts existed that could impact EDG 2. From this evaluation, the licensee concluded that this wiring discrepancy may impact the ability of EDG 2 to perform its intended ASSD function. Because local control of EDG 2 is credited in the safe shutdown analysis, the licensee made an 8-hour notification (EN 48253) in accordance with 10 CFR 50.72(b)(3)(ii)(B) as an unanalyzed condition that significantly degraded plant safety.

The licensee's corrective actions included making the appropriate wiring changes to the EDG 2 control circuit to ensure it was in accordance with the existing approved design, and returning EDG 2 to operable status on September 1, 2012. The licensee inspected the other six ASSD switches on EDG 2 and all of the ASSD switches on EDGs 1, 3 and

4 and verified that a similar wiring error did not exist. The licensee entered this issue into the CAP as NCR 557897.

Analysis. The performance deficiency associated with this finding was the failure to assure that the design basis for EDG 2 ASSD Switch A1 was correctly translated into specifications and drawings. The finding was more than minor because it was associated with the protection against external factors (i.e. fire) attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, an induced fault could have impacted the ability to locally control EDG 2 during certain fire scenarios. Using IMC 0609, Attachment 4, issued June 19, 2012, Initial Characterization of Findings, and IMC 0609, Appendix F, Attachment 1, Part 1: Application of Fire Protection SDP Phase 1 Worksheet, the results of this evaluation required further significance evaluation. A phase 3 analysis was performed by a regional SRA in accordance with NRC IMC 0609 Appendix F. The finding affected the capability to achieve alternate safe shutdown for Unit 1. A bounding analysis was performed with the following major assumptions: a one year exposure period, only the non-load shed circuits from the 1B DC battery located in the cable spreading room were potential hot short source cables to the target jumper cable, two proper polarity hot shorts were required to impact the EDG2 control circuit, only severe fire growth or damaging hot gas layer scenarios were capable of creating simultaneous damage to the source and target cables. No recovery was assumed in the analysis for fuse replacement or re-alignment of the electrical distribution system. The dominant sequence was a challenging fire in the Unit 1 cable spreading room which remained unsuppressed long enough for fire damage to require main control room evacuation and loss of EDG 2 due to the performance deficiency resulting in core damage caused by failure of the ASSD procedure to maintain core heat removal. The risk was mitigated by the small number of source cables, the distance between source and target cables and the magnitude of fire growth required to sustain the damage to the source and target cables. The result of the analysis was an increase in core damage frequency of <1E-6/year a GREEN finding of very low safety significance. The finding does not have a cross-cutting aspect since the performance deficiency is not indicative of current plant performance. The EDG 2 ASSD Switch A1 wiring discrepancy has existed since original EDG installation.

Enforcement. Title 10 of the Code of Federal Regulations Part 50, Appendix B, Criterion III, Design Control, states in part, measures shall be established to assure that applicable regulatory requirements and the design basis, as defined in § 50.2 and as specified in the license application, for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions. Licensee wiring diagram F-09346, Diesel Generator No. 2 Circuits Control Wiring Diagram, does not specify a wire connection across ASSD Switch A1 contact 2-2C. Contrary to the above, between original EDG 2 installation and September 1, 2012, the licensee failed to assure that the design basis for EDG 2 ASSD Switch A1 was correctly translated into specifications and drawings. Specifically, a wiring discrepancy existed associated with EDG 2 ASSD Switch A1 which resulted in an induced fault that could have impacted the ability to locally control EDG 2 during certain fire scenarios. The licensee's corrective actions included correcting the EDG 2 control circuit wiring to ensure it was in accordance with the existing approved design and returned EDG 2 to operable status. Because this finding is of very low safety significance and was entered into the licensee's CAP as NCR 557897, consistent with Section 2.3.2 of the NRC Enforcement Policy, the violation is being treated as an NCV: **NCV 05000325/2012005-04 and 05000324/2012005-04, Inadequate Design of EDG 2 ASSD Switch A1.**

4OA3 Follow-up of Events and Notices of Enforcement Discretion (71153 – 3 samples)

.1 (Closed) LER 05000325, 324/2012-005-00, Local Control Capability of Emergency Diesel Generator No. 2 Not Available

a. Inspection Scope

On August 28, 2012, during planned maintenance on EDG 2, a post-maintenance continuity test of ASSD Switch A1 revealed a closed circuit, when the circuit was expected to be open, when the switch was taken to the Local position. It was determined that a wire, not identified on EDG wiring diagrams, created a short between two contacts associated with ASSD Switch A1. The licensee's cause evaluation determined that this was a historical issue and was likely introduced during original installation of these ASSD switches and no root cause was determined. The wiring error was corrected and the remaining EDG ASSD switches were inspected with no further discrepancies identified. This issue was entered into the licensee's CAP as NCR 557897.

b. Findings

An inspector identified violation was identified as discussed in Section 4OA3.1. This LER is closed.

.2 (Closed) LER 05000325/2012-006-00, Operation Prohibited by Technical Specifications Due to Operation with a Potential for Draining the Reactor Vessel (OPDRV)

a. Inspection Scope

On September 19, 2012 with Unit 1 in Mode 4, Cold Shutdown, maintenance activities were in progress to replace the 1B Recirculation Pump seal assembly with secondary containment inoperable. The 1B recirculation loop isolation valves were closed to support this maintenance and the approximate ten gallons per minute leakage past the isolation valves was routed to the drywell floor drain system. The secondary containment airlock doors for the Unit 1 reactor building were opened to provide additional ventilation flow to the reactor building in an effort to improve working conditions in the Unit 1 drywell. The licensee restored secondary containment operability by closing the reactor building airlock doors and entered this issue in the CAP as NCR 562188.

b. Findings

An inspector identified Green NCV of TS 3.6.4.1, Secondary Containment was identified in NRC Inspection Report 05000325/2012004 and 05000324/2012004 as NCV 05000325/2012004-01, Failure to Maintain Secondary Containment Operable During an OPDRV Activity. No additional violations were identified associated with this LER. This LER is closed.

.3 (Closed) LER 05000325/2011-002-00, Fire Related Unanalyzed Condition that could Impact Equipment Credited in Safe Shutdown Analysis

a. Inspection Scope

On December 8, 2011, the licensee submitted LER 2011-002-00 that documented the discovery of an unanalyzed condition related to their fire protection program (FPP). Inspectors reviewed the facts of the subject LER to determine if a more than minor performance deficiency existed; as well as the corrective actions taken by the licensee to determine if they were adequate. Inspectors also reviewed this finding against NRC enforcement guidance documents to determine if enforcement discretion was applicable.

b. Findings

Introduction. The licensee identified a noncompliance with 10 CFR 50, Appendix R, Section III.G.3, for the failure to provide alternative shutdown capability for fires in certain areas whose protection of SSCs do not satisfy the guidelines of 10 CFR 50, Appendix R, Section III.G.2. Specifically, the licensee had not considered the possibility of certain fire-induced spurious actuations that could adversely impact the ability to achieve and maintain safe shutdown (SSD).

Description. On December 8, 2011, the licensee submitted LER 2011-002-00, describing conditions that may not ensure a protected train of equipment remains available under certain postulated fire scenarios. The licensee discovered that a postulated fire in specific fire areas (FAs) could cause spurious actuation of critical components, potentially resulting in loss of equipment required for safe shutdown. These FAs included RB1-1 (Unit 1 Reactor Building General Areas), RB2-1 (Unit 2 Reactor Building General Areas), TB-1 (Turbine Building Units 1 & 2), CB-2 (Unit 2 Cable Access Ways), and CB-23E (Control Building). For fires in these areas, the licensee's strategy for safe shutdown relies on alternative shutdown, as required by 10 CFR 50, Appendix R, Section III.G.3. The postulated damaged electrical cables affect four general functions. These are:

1) Containment pressure to maintain residual heat removal (RHR) pump net positive suction head (NPSH)
2) Process monitoring
3) Essential electrical support
4) Reactor coolant system (RCS) inventory control

The licensee's analysis discovered that a postulated fire in FA RB2-1 could damage cables associated with containment atmospheric control valves 2-CAC-V9, 2-CAC-V15, and 2-CAC-V49. Fire damage to these cables could result in valves opening, which could lead to a decrease in primary containment pressure. Additionally, a postulated fire in FAs RB1-1 or RB2-1 could damage cables associated with reactor building closed cooling water pumps 1-RCC-1B-PMP and 2-RCC-2B-PMP. Fire damage to these cables could result in the inability to stop the pumps, which could also lead to a decrease in primary containment pressure. The inability to maintain containment pressure could result in elevated suppression pool temperatures, which could reduce the available NPSH for the RHR and core spray pumps.

The licensee's analysis also discovered that a postulated fire in FA RB2-1 could damage control cables for components associated with suppression pool level instrument 2-CAC-LT-2602. Damage to these cables could cause solenoid valves 2-CAC-SV-1219C or 2-CAC-SV-4344 to close, resulting in the loss of suppression pool level indication in the Main Control Room. The loss of this indication could affect the process monitoring function of SSD.

The licensee's analysis also discovered that a postulated fire in FA TB-1 could damage power cables associated with the 1B, 2A and 2C conventional service water pumps. Damage to these cables could result in the EDG's output breaker not being able to close and energize emergency bus E-1 or E-3. Additionally, a postulated fire in FA RB2-1 could damage cables associated with 2A Control Rod Drive pump. Damage to these cables could result in the pumps' feeder breaker remaining closed during an automatic load shed of Emergency Bus E-1 or E-3. These postulated fire scenarios could affect the essential electrical support function of SSD.

The licensee's analysis also discovered that a postulated fire in FAs CB-23E or RB2-1 could damage cables associated with the Unit 2 reactor core isolation coolant (RCIC) barometric condenser vacuum pump, 2-E51-C002-VAC-PMP-M. The pump is listed as an Appendix R safe shutdown component in calculation BNP-E-9.004, Brunswick Safe Shutdown Analysis Revision 8. The RCIC barometric condenser vacuum pump provides a support function for the RCIC system. The RCIC system provides RCS core cooling following a reactor shutdown. A postulated fire scenario involving the RCIC system could adversely affect the RCS inventory control function of SSD.

The licensee entered these conditions into the CAP as NCR 493784 and implemented roving fire watches for the affected fire areas. The licensee also revised alternative safe shutdown procedures to include compensatory actions to address fire damaged cables and equipment.

<u>Analysis</u>. The performance deficiency associated with this finding involved the failure to provide alternative shutdown capability in accordance with 10 CFR 50, Appendix R, Section III.G.3. The finding was more than minor because it is associated with the protection against external events attribute of the Mitigating System Cornerstone and adversely affects the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable

consequences. Specifically, the performance deficiency adversely impacted the licensee's capability to achieve and maintain SSD in the event of a fire in certain areas. Because this issue relates to fire protection and was identified during the licensee's transition to NFPA 805, this issue is being dispositioned in accordance with Section 9.1, "Enforcement Discretion for Certain Fire Protection Issues (10 CFR 50.48)" of the NRC Enforcement Policy. In order to verify that this noncompliance was not associated with a finding of high safety significance (Red), inspectors reviewed qualitative and quantitative risk analyses performed by the licensee. These risk evaluations took ignition source and target information from the Brunswick fire probabilistic risk analysis (PRA) to demonstrate that the significance of the non-compliance was less than Red (i.e. ΔCDF less than 1E-4/yr.). The inspectors performed walkdowns to verify key assumptions were applicable. The inspectors also performed a bounding risk calculation and independently determined that the risk of this issue, based solely on frequency, was less than Red. This calculation conservatively assumed no credit for any mitigation actions (i.e., detection, suppression, operator actions, etc.). The finding does not have a cross-cutting aspect since the performance deficiency is not indicative of current plant performance.

Enforcement. Title 10 of the Code of Federal Regulations, 50.48(b)(1) requires, in part, that all nuclear power plants licensed to operate prior to January 1, 1979, must satisfy the applicable requirements of Appendix R, Section III.G. Appendix R, Section III.G.3 states, in part, that alternative or dedicated shutdown capability be provided where the protection of systems whose function is required for hot shutdown does not satisfy the requirement of 10 CFR 50, Appendix R, Section III.G.2. Contrary to the above, from original plant startup to October 13, 2011, the licensee failed to provide an alternative or dedicated shutdown capability when the requirements of 10 CFR 50, Appendix R, Section III.G.2 were not met. Specifically, the licensee's alternative/dedicated post-fire SSD strategy for five FAs failed to ensure alternative shutdown capability because the licensee had not considered the possibility of certain fire-induced spurious actuations of critical components that would potentially result in the loss of equipment required for safe shutdown. Because the licensee committed to adopt NFPA 805 and change their fire protection licensing bases to comply with 10 CFR 50.48(c), and this commitment was documented prior to December 31, 2005, the NRC is exercising enforcement and reactor oversight process discretion for this issue in accordance with the NRC Enforcement Policy, Section 9.1, "Enforcement Discretion for Certain Fire Protection Issues (10 CFR 50.48)" and Inspection Manual Chapter 0305. This issue was identified and addressed during the licensee's transition to NFPA 805, it was entered into the licensee's CAP as NCR 493784, immediate corrective action and compensatory measures were taken, it was not likely to have been previously identified by routine licensee efforts, it was not willful, and it was not associated with a finding of high safety significance (Red).

This LER is closed.

4OA5 Other Activities

.1 Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

During the inspection period the inspectors conducted observations of security force personnel and activities to ensure that the activities were consistent with licensee security procedures and regulatory requirements relating to nuclear plant security. These observations took place during both normal and off-normal plant working hours. The inspectors reviewed the security logs to ensure that degraded conditions were adequately addressed and compensatory measures were correct for the application. These quarterly resident inspector observations of security force personnel and activities did not constitute any additional inspection samples. Rather, they were considered an integral part of the inspectors' normal plant status reviews and inspection activities.

b. Findings

No findings were identified.

.2 (Discussed) NRC Temporary Instruction (TI) 2515/187 – Inspection of Near-Term Task Force Recommendation 2.3 Flooding Walkdowns

a. Inspection Scope

Inspectors conducted independent walkdowns to verify that the licensee completed the actions associated with the flood protection feature specified in paragraph 03.02.a.2 of this TI. Inspectors are performing walkdowns at all sites in response to a letter from the NRC to licensees, entitled "Request for Information Pursuant to Title 10 of the *Code of Federal Regulations* 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident," dated March 12, 2012 (ADAMS Accession No. ML12053A340).

Enclosure 4 of the letter requested licensees to perform external flooding walkdowns using an NRC-endorsed walkdown methodology (ADAMS Accession No. ML12056A050). Nuclear Energy Industry (NEI) document 12-07 titled, "Guidelines for Performing Verification Walkdowns of Plant Protection Features," (ADAMS Accession No. ML12173A215) provided the NRC-endorsed methodology for assessing external flood protection and mitigation capabilities to verify that plant features, credited in the CLB for protection and mitigation from external flood events, are available, functional, and properly maintained.

b. Findings

Findings or violations associated with TI-187, if any, will be documented in a subsequent report.

.3 <u>(Closed) NRC Temporary Instruction 2515/188 – Inspection of Near-Term Task Force Recommendation 2.3 Seismic Walkdowns</u>

a. <u>Inspection Scope</u>

The inspectors accompanied the licensee on their seismic walkdowns of the following:

- Unit 1, seismic walkdown equipment list (SWEL) 1, 1-E11-C001C, Residual Heat Removal Service Water Booster Pump 1C, Reactor Building, 50' Elevation on July 26, 2012;
- Unit 2, SWEL 1, 2-H21-P006, Recirculation Pump A Instrument Rack, Reactor Building, -17' Elevation on July 26, 2012;
- Unit 1, SWEL 2, 1-G42-001 and 1-G42-002, Suction / Discharge Vortex Breakers, Reactor Building, 115' Elevation on August 9, 2012; and
- Unit 2, SWEL 2, 1-G41-V10 and 1-G41-V24, Fuel Storage Pool Recirculation Valve and Cleanup Return Check Valve, Reactor Building, 117' Elevation on August 9, 2012.

The inspectors accompanied the licensee and verified that the licensee confirmed that the following seismic features associated with these SWEL items were free of potential adverse seismic:

- Anchorage was free of bent, broken, missing or loose hardware;
- Anchorage was free of corrosion that is more than mild surface oxidation;
- Anchorage was free of visible cracks in the concrete near the anchors;
- Anchorage configuration was consistent with plant documentation for SWEL 1 items.
- SSCs will not be damaged from impact by nearby equipment or structures;
- Overhead equipment, distribution systems, ceiling tiles and lighting, and masonry block walls are secure and not likely to collapse onto the equipment;
- Attached lines have adequate flexibility to avoid damage;
- The area appears to be free of potentially adverse seismic interactions that could cause flooding or spray in the area;
- The area appears to be free of potentially adverse seismic interactions that could cause a fire in the area; and
- The area appears to be free of potentially adverse seismic interactions associated with housekeeping practices, storage of portable equipment, and temporary installations (e.g., scaffolding, lead shielding).

The inspectors independently performed their walkdowns of the following:

- Units 1 & 2, SWEL 1, 1-SW-V682 Emergency Diesel Generator 4 Engine Jacket Water Service Water Inlet Isolation Valve, Diesel Building, 24' Elevation on August 16, 2012; and
- Unit 1, SWEL 1, 1-CAC-TR-4426-1A, Drywell/Suppression Pool Temperature Recorder, Control Building, 49' Elevation on August 17, 2012.

Inspectors verified the following:

- Anchorage was free of bent, broken, missing or loose hardware;
- Anchorage was free of corrosion that is more than mild surface oxidation;
- Anchorage was free of visible cracks in the concrete near the anchors;
- SSCs will not be damaged from impact by nearby equipment or structures;
- Overhead equipment, distribution systems, ceiling tiles and lighting, and masonry block walls are secure and not likely to collapse onto the equipment;
- Attached lines have adequate flexibility to avoid damage;
- The area appears to be free of potentially adverse seismic interactions that could cause flooding or spray in the area;
- The area appears to be free of potentially adverse seismic interactions that could cause a fire in the area; and
- The area appears to be free of potentially adverse seismic interactions associated with housekeeping practices, storage of portable equipment, and temporary installations (e.g., scaffolding, lead shielding).

Observations made during the walkdown that could not be determined to be acceptable were entered into the licensee's CAP for evaluation.

Additionally, inspectors verified that items that could allow the spent fuel pool to drain down rapidly were added to the SWEL and these items were walked down by the licensee.

b. Findings

Introduction. The inspectors identified a Green finding for the licensee not having an adequate procedure for maintenance on fluorescent lights over safety-related equipment. Specifically, between plant startup and August 29, 2012, the licensee did not have instructions for closing S-hooks on fluorescent lights over safety-related equipment during maintenance on the fluorescent lights. This resulted in over 40 S-hooks open in safety-related buildings which could result in fluorescent lights falling and impacting safety-related equipment during a seismic event.

Description. During walkdowns associated with Temporary Instruction 188, Inspection of Near-Term Task Force Recommendation 2.3 Seismic Walkdowns, the inspectors identified open S-hooks on fluorescent lights over safety-related instruments and instrument racks 1-H21-P004 and 1H21-P005 (both reactor protection and nuclear steam supply instrumentation rack), and 1-E11-C001A (1C residual heat removal service water booster pump). The inspectors questioned if the S-hooks should be closed to prevent the fluorescent lights from falling on the safety-related instrument racks during a seismic event which could result in a reactor trip. The licensee determined that the S-hooks should be closed and closed the three S-hooks.

The inspectors reviewed Generic Letter 87-02, Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue A-46, that was issued on February 19, 1987, which concluded that the seismic adequacy of certain equipment in operating nuclear power plants should be reviewed against seismic criteria not in use when these plants were licensed. The NRC issued Supplemental Safety Evaluation Report No. 2 on February 14, 1992 that endorsed Generic Implementation Procedure (GIP), Revision 2 to be used to perform walkdowns and evaluations for seismic walkdowns. Updated Final Safety Analysis Report Section 3.10.1.6, Analytical Approaches for Equipment Supplied by Others, states, in part, that "Revision 3 of the SQUG GIP-03, as modified and supplemented by the NRC Supplemental Safety Evaluation Report No. 2 and No. 3 may be used as an alternative to existing methods for the seismic design and verification of modified, new and replaced equipment."

In the early 1990's, the licensee performed walkdowns and documented seismic qualifications and outliers in accordance with the GIP. The licensee sent their response to the NRC with a discussion of the outliers, and then later updated the NRC when the outliers were fixed. Some of the outliers included closing open S-hooks for lights over safety-related equipment. The GIP, Section D.3.3, Architectural Features, for light fixtures states, in part, that "Pendant-hung fluorescent fixtures and tubes pose the highest risk of failure and damage to sensitive equipment. The Seismic Capability Engineers should check for positive anchorage, such as closed hooks and properly twisted wires." The licensee did close the open S-hooks that were identified during the A-46 walkdowns; however, the inspectors identified that the licensee did not implement procedural guidance to ensure future closure of S-hooks over safety-related equipment. A procedure issue existed where maintenance performed on the lights fixtures over safety-related equipment did not have a step in the WO to ensure that the S-hooks were closed.

The licensee seismic walkdown implementing Procedure, URS Near Term Task Force Recommendation 2.3 Seismic, Section 4.3.1, states that "in preparation for the actual walkdowns of the equipment, the following information, but not limited to, should be obtained as needed for the walkdowns: available documents of prior walkdowns of those seismic walkdown equipment list (e.g. IPEEE, USI A-46)." The inspectors determined that the licensee's USI A-46 report showed open S-hooks as outliers requiring the licensee to close the open S-hooks. The inspectors noted that the licensee had the opportunity to but did not identify the open S-hook issues during their walkdowns post-Fukushima.

As part of extent of condition, the licensee inspected the overhead lights for open S-hooks in safety-related buildings and found over 40 S-hooks open in the reactor building. Licensee's corrective actions included closing the open S-hooks and adding instruction in generic WO 431558 for fluorescent lights to close the open S-hooks after maintenance. The licensee entered this issue into the CAP as NCR 551646.

Analysis. The performance deficiency associated with this finding was the failure of the licensee to have an adequate procedure for maintenance on fluorescent lights over safety-related equipment. The finding was more than minor because if left uncorrected, the deficiencies could lead to a more significant safety concern. If left uncorrected, the failure to provide procedural guidance to close the S-hooks on fluorescent lights over safety-related equipment could lead to fluorescent lights falling on safety-related instruments during a seismic event resulting in a reactor trip. This finding is also associated with the design control attribute of the Initiating Events Systems Cornerstone. Using IMC 0609, Appendix A, issued June 19, 2012, The Significance Determination Process (SDP) for Findings At-Power, the inspectors determined the finding was of very low safety significance because the finding did not cause a reactor trip and the loss of mitigation equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition. The finding has a cross-cutting aspect in the area of problem identification and resolution associated with the CAP attribute because the licensee did not identify the open S-hook issue completely, accurately, and in a timely manner commensurate with their safety significance during the Fukushima walkdowns. [P.1(a)]

Enforcement. This finding does not involve enforcement action because no regulatory requirement violation was identified since the florescent lights are not safety-related. The licensee entered this issue into the CAP as NCR 551646. Because this finding does not involve a violation and is of very low safety or security significance, it is identified as **FIN 05000325/2012005-05 and 05000324/2012005-05, Inadequate Maintenance Procedure for Fluorescent Lights over Safety-related Equipment.**

4OA6 Management Meetings

Exit Meeting Summary

On January 17, 2012 and January 29, 2012, the inspector presented the inspection results from the quarterly inspection to Mr. Michael Annacone, and other members of the licensee staff. The inspectors confirmed that proprietary information was retained by the inspectors.

On October 25, 2012, the inspectors held a teleconference with licensee staff and a State of North Carolina radiation protection representative to discuss the status of Brunswick's groundwater monitoring program. The licensee provided an update on tritium concentrations in water collected from onsite and offsite groundwater and surface water sampling locations and discussed ongoing remediation efforts associated with the Storm Drain Stabilization Pond (SDSP) and areas near a U1 Condensate Storage Tank (CST) underground pipe leak. Although seasonal fluctuations can occur, the inspectors noted that onsite tritium concentrations in and near the SDSP have generally trended downward since 2007 when the contamination was discovered and corrective actions were initiated. The licensee has installed shallow and intermediate-depth wells in the vicinity of the CST piping leak in order to better characterize the tritium plume and to facilitate remediation of the groundwater. Some of these wells have detected low levels of tritium in the top of the Castle Hayne aquifer in the area immediately below the Brunswick site. Wells have also been constructed further away from the leak site to monitor any plume migration through Castle Hayne. Samples taken from these wells have not shown any detectable tritium. The inspectors noted that although very low concentrations of tritium have been identified periodically in the offsite environs, e.g., Nancy's Creek immediately adjacent to the SDSP, all reported values for offsite samples have remained significantly below established regulatory limits. The licensee is currently remediating the groundwater around the SDSP through a network of sub-surface pumping wells. Water pumped from this network is transferred to a new, double-lined, retention pond. Publicly available information regarding onsite groundwater monitoring and radionuclide concentrations in the environment near Brunswick can be found in the Annual Radiological Environmental Operating Report. The 2011 Annual Report is currently available through the Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html (accession number ML12145A348). A 30-day report regarding the U1 CST piping leak is also available through ADAMS (accession number ML110190210).

4OA7 Licensee-Identified Violations

None.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

M. Annacone, Site Vice President
D. Barker, Manager of Nuclear Oversight
S. Bostic, Supervisor – Nuclear Construction Management
A. Brittain, Manager – Security
J. Burke, Director – Engineering
C. Dunsmore, Manager – Shift Operations
P. Dubrouillet, Manager - Training
J. Krakuszeski, Plant General Manager
C. George, Manager – Nuclear Systems Engineering
K. Gerald, Superintendent – Mechanical Maintenances
S. Gordy, Manager – Maintenance
L. Grzeck, Supervisor – Licensing / Regulatory Programs
R. Ivey, Director – Audits and Programs
F. Jefferson, Manager – Systems Engineering
J. Johnson, Manager – Environmental and Radiological Controls
C. Martinec, Duke Corporate Sr. Scientist
J. McGee, Silar Services – Hydrologist
M. McGowan, Supervisor – Environmental
P. Mentel, Director – Nuclear Merger Integration
M. Millinor, Sr. Chemistry Specialist
M. Nemec, Supervisor LOR Training
D. Petrusic, Superintendent – Environmental and Chemistry
A. Pope, Manager– Support Services
J. Price, Manager– Design Engineering
E. Rau, Lead Operations Training Instructor
M. Regan, Project Manager
T. Sherrill, Senior Engineer - Technical Support Specialist
T. Silar, Silar Services
P. Smith, Superintendent – Electrical, Instrumentation, and Controls Maintenance
J. Spencer, URS Engineers (Modeling)
M. Turkal, Lead Engineer - Technical Support
J. Vincelli, Superintendent – Radiation Protection
E. Wills, Director – Site Operations

North Carolina Department of Environment and Natural Resources

P. Cox, Health Physicist

NRC Personnel

R. Musser, Chief, Reactor Projects Branch 4, Division of Reactor Projects Region II
B. Bonser, Chief, Plant Support Branch 1
R. Cady, Sr. Performance Assessment Analyst – Office of Nuclear Regulatory Research
R. Conatser, Health Physicist – Office of Nuclear Reactor Regulation
T. Nicholson, Sr. Technical Advisor for Radionuclide Transport - Office of Nuclear Regulatory Research

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000325; 324/2012005-03	NCV	Inadequate Maintenance Procedure for the EDG Jacket Water Pump Wear Ring Tolerances (Section 1R19)
05000325; 324/2012005-04	NCV	Inadequate Design of EDG 2 ASSD Switch A1 (Section 4OA2.5)
05000325; 324/2012005-05	FIN	Inadequate Maintenance Procedure for Fluorescent Lights over Safety-related Equipment (Section 4OA5.3)

Opened

05000325; 324/2012005-01	URI	Floor Drains Not Functioning Due to Plugging (Section 1R12)
05000325; 324/2012005-02	URI	Emergency Diesel Generator 3 Slow Start (Section 1R19)

Closed

05000325; 324/2012004-03	URI	EDG2 wiring on Alternate Safe Shutdown Switch (Section 4OA2.5)
05000325, 324/2012-005-00	LER	Local Control Capability of Emergency Diesel Generator No. 2 Not Available (Section 4OA3.1)
05000325/2012-006-00	LER	Operation Prohibited by Technical Specifications Due to Operation with a Potential for Draining the Reactor Vessel (OPDRV) (Section 4OA3.2)
05000325/2011-002-00	LER	Fire Related Unanalyzed Condition that could Impact Equipment Credited in Safe Shutdown Analysis (Section 4OA3.)
NRC Temporary Instruction 2515/188	TI	Inspection of Near-Term Task Force Recommendation 2.3 Seismic Walkdowns (Section 4OA5.3)

Discussed

NRC Temporary Instruction 2515/187	TI	Inspection of Near-Term Task Force Recommendation 2.3 Flooding Walkdowns (Section 4OA5.2)

LIST OF DOCUMENTS REVIEWED

Common Documents Reviewed
Updated Final Safety Analysis Report
Individual Plant Examination
Individual Plant Examination of External Events
Technical Specifications and Bases
Technical Requirements Manual
Control Room Narrative Logs
Plan of the Day

Section 1R01: Adverse Weather Protection
Procedures
0AOP-13.0, Operation during Hurricane, Flood Conditions, Tornado, or Earthquake
0AI-68, Brunswick Nuclear Plant Response to Severe Weather Warnings
0PEP-02.1, Initial Emergency Actions
0PEP-02.6, Severe Weather
0O1-01.03, Non-Routine Activities, Freeze Protection and Cold Weather Bill
0PM-HT001, Preventative Maintenance on Plant Freeze Protection and Heat Tracing System

Condition Reports
575093

Miscellaneous
Operating Experience Smart Sample (OpESS) 2012/01, High Wind Generated Missile Hazards

Section 1R04: Equipment Alignment
Procedures
0OP-39, Diesel Generator Operating Procedure
SD-39, System Description, Emergency Diesel Generators
OP-21, Reactor Building Closed Cooling Water System Operating Procedure
2OP-19, Unit 2 High Pressure Coolant Injection Operating Procedure

Drawings
D-02266, sheets 2A and 2B, Piping Diagram for Diesel Generators Starting Air System Units 1 and 2
D-02269, sheets 2A and 2B, Piping Diagram for Diesel Generators Fuel Oil System Units 1 and 2
D-2538, Reactor Building Closed Cooling Water Piping Diagram
D-02523, High Pressure Coolant Injection P&ID

Section 1R05: Fire Protection
Procedures
0PFP-DG, Diesel Generator Building Prefire Plans
1PFP-RB, Reactor Building Prefire Plans Unit 1
2PFP-RB, Reactor Building Prefire Plans Unit 2
0PFP-013, General Fire Plan
0-FPP-013, Transient Fire Load Evaluation0PT-34.11.2.0, Portable Fire Extinguisher Inspection

0PT-34.6.7.10, Fire Barrier Penetrations Seals Diesel Generator Building
0OP-41, Fire Protection and Well Water System

Condition Reports

572189	572185	572186	572190	572191	573084

Drawings
F-04093, Diesel Generator Building Ventilation System Plans at Elev. 23' and 50'
LL-FB-07200, Diesel Generator Bldg. Fire Barrier Penetrations General Layout El. 23'-0"

Miscellaneous
BNP-E-9.004, Safe Shutdown Analysis Report

Section 1R07: Heat Sink Performance

Procedures
0MST-DG500R, Emergency Diesel Generators 24 Month Inspection

Condition Reports

355194	353245	311477

Miscellaneous
EDG-1-JWC-2010, Anatec preliminary eddy current inspection report
EDG-3-JWC-2012, Anatec final eddy current inspection report
0ENP-2704, Administrative Control of NRC Generic Letter 89-13 Requirements
NLS-90-005, CP&L Response to NRC Generic Letter 89-13
Calculation 0SW-0096, Calculation for Tube Plugging and Fouling of Service Water Safety

Section 1R11: Licensed Operator Requalification

Procedures
0TPP-200, Licensed Operator Continuing Training
0PEP-04.3, Performance of Training, Exercises, and Drills
TAP-417, Licensed Operator Continuing Training Administrative Procedure
TRN-NGGC-0014, NRC Initial Licensed Operator Exam Development and Administration
1EOP-01-LPC, Level/Power Control
0PEP-2.1.1, Emergency Control – Notification of Unusual Event, Alert, Site Area Emergency, or General Emergency
0PEP-02.1, Initial Emergency Actions
0OI-01.01. BNP Conduct of Operations Supplement
OPS-NGGC-1000,. Fleet Conduct of Operations

Records:
License Reactivation Packages - 4 Reviewed]).
LORP Training Attendance records - 12 Reviewed.
Medical Files - 10 Reviewed.
Remedial Training Records - 12 Reviewed.
Remedial Training Examinations - 8 Reviewed.
Feedback Summaries - 15 Reviewed.

Written Examinations:
11-6 Exam 2 - SRO, Week 2 Biennial Written Exam - SRO, Revision 0
11-6 Exam 4 - SRO, Week 4 Biennial Written Exam - SRO, Revision 0

Procedures:
0TPP- 206, Simulator Program, Revision 5
SI-214.1, Simulator Documentation, Revision 3
SI-216.1, Regulatory Testing, Revision 20
TAP – 409, Miscellaneous Simulator Training Guidelines Revision 20.
TAP-412, Simulator Operation and Maintenance, Revision 4
TRN-NGGC-002, Performance Review and Remedial Training, Revision 4
TRN-NGGC-0300, Development Phase, Revision 5
TRN-NGGC-0420, Conduct of Simulator Training and Evaluation, Revision 3
TRN-NGGC-0425, Simulator Scenario Based Training, Revision 0
TRN-NGGC-0441, Licensed Operator Requal Annual/Biennial Exam Development, Revision 1
FORM CAP-NGGC-0201-6-17, Self-Assessment Debrief Notes, 493862

Simulator Steady State Tests:
STP-OL-001, Simulator Operating Limits Test, Revision 8
STP-SS-003, 75% Steady State, Revision 11

Simulator Normal Evolution Tests:
SMR 09-0009-0, B2C 19 Reload Core Design, 2/5/2009
STP-TN-007, Max Rate Power Ramp – Recirc Flow Controller in Manual, Revision 4

Simulator Transient Tests:
STP-TN-005, Single Recirculation Pump Trip, Revision 4
STP-TN-11, Inadvertent HPSI Initiation, Revision 4

Simulator Problem Reports & Design Change Requests:
PM-SSR 10-0070, Decommission of ESF System Status Board
SSR 10-0023, Full Core Display Problems
SSR 11-0063, Drywell High Range Radiation Monitor D22-RI-4198

Scenario Packages:
LORX-205, Examination, RB Vent Radiation Monitor Failure causing RB HVAC to isolate, Condenser Tube Leak, Emergency Depressurization due to Two Areas above Maximum Safe Operating Temperature, Revision 01, 11/21/2012
LORX-001, APRM Critical Fault, 2B CBP Pump Trip, Loss of 4140 V Bus 2D, Automatic Scram Failure, HPCI FIC Power Loss, Small Break LOCA, ADS Timers Fail, Emergency Depressurization at TAF, Revision 16, 11/21/2012

JPM Packages:
AOT-ADM-JP-051-05, DC Ground Calculation, Revision 2
AOT-OJT-JP-041-A03, Alternate Coolant Injection – Starting of the Diesel Fire Pump with Battery Failure, Revision 7
AOT-OJT-JP-300-J11, LEP-01; Alternate Coolant Injection - SLC Pumps Using Fire Protection Water, Revision 8

JPM – LOT-ADM-JP-301-A01, Estimate Release From Unit 1 Reactor Building and Turbine Building, Per PEP- 03.6.1
JPM – AOT- OJT – JP – 303 –A12, Station BLACKOUT: Cross-Tying 480 V Emergency Buses – Breaker Charging Spring Failure (AP)
JPM-AOT-JP-300-J11, LEP-01; Alternate Coolant Injection – SLC Pumps Using Fire Protection Water
JPM – LOT-SIM-JP-017-A08, Draining the Suppression Pool to Radwaste with a Thermal Overload (AP)
JPM-LOT-SIM-JP-024-A03, Primary Containment Ventilation During Personnel Entry.
AOT-OJT-JP-303-A12, Station Blackout: Cross-Tying 480 V Emergency Buses – Breaker Charging Spring Failure, Revision 0
LOT-ADM-JP-301-A01, Estimate Release From Unit One Reactor Building and Turbine Building Per PEP-03.6.1, Revision 3
LOT-OJT-JP-012-A01, Place a PCIS Channel In Tripped – Reactor Water Level, Revision 5
LOT-SIM-JP-007-02, Perform Rod Drift Alarm Test, Revision 1
LOT-SIM-JP-017-A08, Draining the Suppression Pool to Radwaste with a thermal overload, Revision 5
LOT-SIM-JP-024-A03, Primary Containment Ventilation During Personnel Entry, Revision 2
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ADM-NGGC-0203, Preventive Maintenance and Surveillance Testing Administration
EGR-NGGC-0351, Condition Monitoring of Structures
ADM-NGGC-0203, Preventive Maintenance and Surveillance test Administration
0AP-022, BNP Outage Risk Management

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574261 558584 521571 520545

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NUMARC 93-01, Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants

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0AI-144, Risk Management
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574252 574256 574259 574262

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1OP-10, Unit 1 Standby Gas Treatment Operating Procedure
0LP-AO010, Calibration and Functional Testing of Johnson Controls Model D-9502

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0-FP-20014, Engine Pneumatic Control Schematic
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DBD-05, Design Basis Document, Standby Liquid Control System
DBD-37.4, Design Basis Document, Diesel Generator Building Ventilating Air System

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EGR-NGGC-0005, Engineering Change
EGR-NGGC-0011, Engineering Product Quality
0PT-12.3.2B, No. 2 Diesel Generator Starting Air Valve Operability Test

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SK-70110-M-2200, Starting Air for Diesel Generators Piping Diagram
SK-70110-M-2203, Diesel Generator Building Steam Heating & Ventilation
SK-70110-M-2208, Diesel Generator Building Steam Heating & Ventilation
SK-70110-M-2209, Diesel Starting Air System No.1 to No. 3 Cross-tie
SK-70110-M-2210, Diesel Starting Air System No.2 to No. 4 Cross-tie
SK-70110-E-3000, Engine Pneumatic Control Schematic
F-02214, Diesel Generator Building Composite Piping Sheet 1
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D-02265, Starting Air for Diesel Generator Piping Diagram Sheet 1A
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DBD-30, Emergency Diesel Generator System
Engineering Change EC70110, EDG Reliability Upgrade - Starting Air Cross-Tie
0SP-EC70110-DG2, Emergency Diesel No. 2 Starting Air Modification Functional Acceptance Testing

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ADM-NGGC-0107, Equipment Reliability Process Guideline
ADM-NGGC-0203, Preventative Maintenance and Surveillance Test Administration
NGG-PMB-PMP-02, Equipment Reliability Template – Horizontal Pumps
0PT-12.2A No. 1 Diesel Generator Monthly Load Test
0PT-12.2C No. 3 Diesel Generator Monthly Load Test
0CM-ENG528, Gould Engine Driven Jacket Water Pump Model 3736
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0MST-RHR26Q, Residual Heat Removal / Core Spray Low Reactor Press Permissive
0GP-02, Approach to Criticality and Pressurization of the Reactor
0PT-10.1.1, Reactor Core Isolation Cooling System Operability Test

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574276	575276	575361	507685	504376

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OPEP-02.6.21, “Emergency Communicator,” Revision 62
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OPEP-02.6.30, “Activation and Operation of the Alternate Emergency Facility,” Revision 3
OPEP-02.6.26, “Activation and Operation of the Technical Support Center,” Revision 27
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0EOP-02-PCCP, Primary Containment Control Procedure

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BNP EP Team Drill Narrative Summary, November 6, 2012
Brunswick Event Notification Forms, November 6, 2012

Section 4OA1: Performance Indicator Verification

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REG-NGGC-0009, NRC Performance Indicators and Monthly Operating Report Data

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MSPI Reports for HPCI/RCIC October 2011 to December 2011
MSPI Reports for HPCI/RCIC January 2012 to March 2012
MSPI Reports for HPCI/RCIC March 2012 to May 2012
MSPI Reports for HPCI/RCIC June 2012 to August 2012

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0E&RC- 1010, Fuel Oil Testing Program
0PT-02.2.4A, Primary Containment Integrity Verification – Containment External
0AOP-36.1, Loss of Any 4160V Buses or 480V E-Buses
1-AOP-04.0, Low Core Flow
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355253	470151	466589	537734	567182	461690
474928	466834	504293	477994	678801	416714
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Maintenance Rule a(1) Issues, October 2012
Operating Experience Smart Sample (OpESS) 2012/02, Revision 1, Technical Specification Interpretation and Operability Determination
Operations Hit List, October 2012
Performance Assessment & Trending Report, Third Quarter 2012
Regulatory Guide 1.137, Fuel Oil Systems for Standby Diesel Generators
System Health Report, Emergency Diesel Generators & Auxiliaries, December 2012
Technical Specification Bases Change 2011-03, Technical Specification Bases Change for Diesel Fuel Oil Program

Section 4OA3: Follow-up of Events
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0OI-01.01, BNP Conduct of Operations Supplement
EGM 11-003, Enforcement Guidance Memorandum on Dispositioning Boiling Water Reactor Licensee Noncompliance with Technical Specification Containment Requirements During Operations with a Potential for Draining the Reactor Vessels

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BNP-0251, Documents Information to Support the SDP for LER BSEP 11-0104, dated August 8, 2012

Drawings
F-09346, Diesel Generator 2 Circuits Control Wiring Diagram
9527-F-36044, Reactor Building-Unit No. 1 Plan-Tray Installation Elevation 20’ 0” – West, Revision 7
9527-F-36045, Reactor Building-Unit No. 1 Plan-Tray Installation Elevation 20’ 0” – East, Revision 7
F-03495, Control Building-Unit No. 2 Plan-Trays and Conduits Elevation 23’ 0”, Revision 29
9527-F-3644, Reactor Building-Unit No. 2 Plan-Tray Installation Elevation 20’ 0” – West, Revision 11
F-03446, Turbine Building-Unit No. 2 Plan-Trays and Conduits Elevation 20’ 0”, Northwest Quadrant, Revision 42
F-03448, Turbine Building-Unit No. 2 Plan-Trays and Conduits Elevation 20’ 0”, Northeast Quadrant, Revision 73

F-03485, Turbine Building Plan-Electrical Cableways & Conduits Below Elevation 20’ 0”, Revision 48
9527-F-3645, Reactor Building-Unit No. 2 Plan-Tray Installation Elevation 20’ 0” – East, Revision 10
F-03487, Control & Radwaste Buildings-Unit No. 2 Plan-Electrical Cableways & Conduits-Elevation 9’ 0”, Revision 33
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9527-F-3646, Reactor Building-Unit No. 2 Plan-Tray Installation Elevation 50’ 0” – West, Revision 2
9527-F-3647, Reactor Building-Unit No. 2 Plan-Tray Installation Elevation 50’ 0” – East, Revision 2
F-035031, Reactor Building-Unit No. 1 Plan-Trays and Conduits Elevation 50’ 0”-East, Revision 44
9527-F-36047, Reactor Building-Unit No. 1 Plan-Tray Installation Elevation 50’ 0” – East, Revision 6
F-03497, Control Building-Units No. 1 & 2 Plan-Trays and Conduits Elevations 38’ 0” and 70’ 0”, Revision 25
F-03530, Reactor Building-Unit No. 2 Plan-Trays and Conduits Elevation 50’ 0”-West, Revision 63
F-34052, Turbine Building-Unit No. 1 Plan-Trays and Conduits Els. 38’ 0” & 45’ 0”-Southwest Quadrant, Revision 44
F-035056, Turbine Building-Unit No. 1 Plan-Trays and Conduits Elevations 38’ 0’ & 45’ 0”-Southeast Quadrant, Revision 29
F-34046, Turbine Building-Unit No. 1 Plan-Trays and Conduits Elevation 20’ 0”-Southwest Quadrant, Revision 33
F-03527, Reactor Building-Unit No. 2 Plan-Trays and Conduits Elevation 20’ 0”-East, Revision 83
F-03531, Reactor Building-Unit No. 2 Plan-Trays and Conduits Elevation 50’ 0”-East, Revision 48

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LER 05000325,324/2012-005-00, Local Control Capability of Emergency Diesel Generator No. 2 Not Available
LER 05000325/2012-006-00, Operation Prohibited by Technical Specifications Due to Operation with a Potential for Draining the Reactor Vessel (OPDRV)

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0AOP-13.0, Operation During Hurricane, Flood Conditions, Tornado or Earthquake
0AI-68, Brunswick Nuclear Plant Response to Severe Weather Warnings

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1-FP-09319, Reactor Building Railroad Doors, Sheet 1
1-FP-09319, Reactor Building Railroad Doors, Sheet 2

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Specification 024-001, Specification for Special Doors
Walkdown Record Form for 1-RB1-DR-EL020-209, Railroad Track Door/Equipment Access Air Lock, October 4, 2012
NEI 12-07, Guidelines for Performing Verification Walkdowns of Plant Flood Protection Features
IPEEE – Individual Plant Examination for External Events, June 1995

Section 4OA6: Management Meetings

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402755