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

April 30, 2010

Mr. Christopher L. Burton
Vice President
Carolina Power and Light Company
Shearon Harris Nuclear Power Plant
P. O. Box 165, Mail Code: Zone 1
New Hill, North Carolina 27562-0165

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT - NRC INTEGRATED INSPECTION REPORT 05000400/2010002, 05000400/2010501

Dear Mr. Burton:

On March 31, 2010, the US Nuclear Regulatory Commission (NRC) completed an inspection at your Shearon Harris reactor facility. The enclosed integrated inspection report documents the inspection results, which were discussed on April 26, 2010, with you and other members of your staff.

The inspections 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.

Based on the results of this inspection, the NRC has determined that one Severity Level IV violation of NRC requirements occurred. Additionally, this report documents two NRC-identified findings of very low safety significance (Green). These findings were determined to involve a violation of NRC requirements. However, because of the very low safety significance and because they are entered into your corrective action program (CAP), the NRC is treating these findings as non-cited violations (NCVs) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest any NCV, 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 Shearon Harris facility. In addition, if you disagree with the characterization of the findings 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 Shearon Harris facility. The information you provide will be considered in accordance with Inspection Manual Chapter 0305.

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

Sincerely,

/RA/

Randall A. Musser, Chief
Reactor Projects Branch 4
Division of Reactor Projects

Docket Nos.: 50-400
License No.: NPF-63
Enclosure: NRC Inspection Report 05000400/2010002
w/Attachment: Supplemental Information

cc w/encl: (See page 3)

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Letter to Christopher L. Burton from Randall A. Musser dated April 30, 2010

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT - NRC INTEGRATED INSPECTION REPORT 05000400/2010002, 05000400/2010501

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No.:	50-400
License No.:	NPF-63
Report No.:	05000400/2010002 05000400/2010501
Licensee:	Carolina Power and Light Company
Facility:	Shearon Harris Nuclear Power Plant, Unit 1
Location:	5413 Shearon Harris Road New Hill, NC 27562
Dates:	January 1, 2010 through March 31, 2010
Inspectors:	J. Austin, Senior Resident Inspector P. Lessard, Resident Inspector L. Miller, Senior Emergency Preparedness Inspector (1EP2, 1EP3, 1EP4, 1EP5, 4OA1, 4OA5) J. Beavers Emergency Preparedness Inspector (1EP2, 1EP3, 1EP4, 1EP5, 4OA1, 4OA5) A. Nielsen, Health Physicist (4OA5)
Approved by:	Randall A. Musser, Chief Reactor Projects Branch 4 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000400/2010002, 05000400/2010501; January 1, 2010 – March 31, 2010; Shearon Harris Nuclear Power Plant, Unit 1; Identification and Resolution of Problems, and Other Activities.

The report covers a three month period of inspection by resident inspectors, announced baseline inspection by two emergency preparedness inspectors and closure of an unresolved item by a region based health physicist. Three NRC-identified findings of very low safety significance (Green) 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 0305, Operating Reactor Assessment Program. 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

- SL-IV. The inspectors identified a Severity Level IV, non-cited violation (NCV) of 10 CFR 50.73(a)(2)(i)(B) due to the licensee's failure to recognize that the inability of the "B" Emergency Service Water (ESW) Discharge Valve (1SW-271) to open on the start of "B" ESW pump caused a reportable condition. Consequently, the licensee failed to submit a licensee event report (LER) within 60 days as required by 10 CFR 50.73. The licensee entered this issue into the corrective action program (CAP) as Action Request (AR) #361821 and AR #358062. The licensee took corrective action by reporting this event in LER 05000400/2010-001, Clearance Error Results in Equipment Becoming Inoperable.

  The licensee's failure to recognize that the inability of 1SW-271 to open caused a reportable condition and submit an LER as required by 10 CFR 50.73 was a performance deficiency. This issue was dispositioned as traditional enforcement, instead of the Significance Determination Process, because it had the potential for impacting the NRC's ability to perform its regulatory function. However, because this violation was of very low safety significance, was not repetitive or willful, and was entered into the licensee's CAP as AR #361821 and AR #358062, the NRC has characterized the significance of this violation as a Severity Level IV NCV in accordance with section IV.A.3 and supplement I of the NRC Enforcement Policy. The cause of this event was directly related to the cross-cutting aspect in the area of problem identification and resolution within the CAP component because the licensee did not adequately evaluate the need to submit an LER per the requirements of 10 CFR 50.73. (P.1(c)) (Section 4OA2.2).

Cornerstone: Mitigating Systems

- Green. The inspectors identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion XVI, Corrective Action, when the licensee failed to promptly evaluate operating experience (OE) received October 22, 2008 and identify potential steam voiding in the residual heat removal (RHR) system as a condition adverse to quality.

During the evaluation, which was not completed until July 16, 2009, the licensee learned that the suction lines for the RHR pumps are susceptible to steam voiding at temperatures as low as 240°F. If the steam void flowed to an RHR pump, that pump could fail causing the associated train of the Emergency Core Cooling System (ECCS) to fail. The delay in evaluating the OE resulted in a delay of determining and implementing appropriate corrective actions. Specifically, the failure to promptly evaluate this OE enabled the licensee to violate Technical Specification (TS) 3.0.4 when the plant transitioned from Mode 4 to Mode 1 with only one operable train of ECCS after refueling outage (RFO) 15 on May 9, 2009. The licensee entered this issue into the CAP as AR #345425. The licensee took corrective action by changing procedures to avoid exposing the suction lines to excessive temperatures during Modes when it is required to be operable for ECCS, thereby preventing potential steam voiding.

The inspectors determined that the failure to promptly evaluate OE received on October 22, 2008, and identify potential steam voiding as condition adverse to quality was a performance deficiency. The performance deficiency was more than minor because it was associated with the equipment performance attribute of the Mitigating Systems Cornerstone and it adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, it could have potentially caused one or more RHR pumps and associated ECCS trains to be inoperable due to steam voiding. Using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," Phase 1 Worksheet, the inspectors concluded that a Phase 2 evaluation was required because this finding represented a potential loss of safety function of the RHR system. The inspectors performed a Phase 2 analysis using IMC 0609 Appendix A, "Determining the Safety Significance of Reactor Inspection Findings for At-Power Situations" and the site specific risk informed inspection notebook. Due to the site specific risk informed inspection notebook not containing appropriate target sets to accurately estimate the risk input of the finding, it was determined that a Phase 3 analysis was required. A regional Senior Reactor Analyst performed the Phase 3 evaluation and concluded the finding was of very low safety significance (Green). The NRC's most current Standardized Plant Analysis Risk Model was used for the evaluation. The evaluation assumed that the "B" RHR Pump always failed to start for the exposure time of seventy hours. Also, there was a potential increase in the common cause failure of the RHR pumps. The dominant accident sequence was a postulated Small Break LOCA with initial success of the ECCS via High Pressure Injection, but the ECCS failed in the recirculation mode. The SDP performed for this violation considered the potential loss of safety function of the RHR system and therefore bounded all violations described in LER 05000400/2009-002 which is further discussed in Section 4OA3.2.

This finding was determined to have a cross-cutting aspect in the OE component of the Problem Identification and Resolution area, in that the licensee failed to evaluate OE in a timely manner (P.2(a)) (Section 4OA2.3).

- <u>Green.</u> The inspectors identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," in that the licensee failed to maintain the "A" ESW pump power cables in an environment for which they were designed. Specifically, the cables were submerged in water in manway 73B-SA, a condition for which they were not qualified. The licensee entered this issue into the CAP as AR #376709. As immediate corrective action, the licensee pumped the manway dry.

The inspectors determined that the failure to ensure that the “A” ESW pump power cables were maintained in an environment for which they were designed was a performance deficiency. The finding was more than minor because, if left uncorrected, it had the potential to lead to a more significant safety concern. Specifically, it could have caused the “A” ESW pump to become inoperable in the event that the cable failed due to long term degradation as a result of continuous submergence. The finding affected the equipment performance attribute of the Mitigating Systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors evaluated the significance of this finding using IMC 0609, “Significance Determination Process,” Phase 1 Worksheet. The finding was of very low safety significance because it was a qualification deficiency that did not result in a loss of operability. This finding was determined to have a cross-cutting aspect in the CAP component of the Problem Identification and Resolution area associated with timely and effective corrective actions (P.1(d)) (Section 1R06).

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status

Unit 1 began the inspection report at or near rated thermal power (RTP). On January 13, 2010, Unit 1 reduced power to approximately 60 percent RTP to repair containment fan cooler (AH-3). The plant returned to RTP on January 14, 2010, and remained there for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection

.1 Winter Seasonal Readiness Preparations

a. Inspection Scope

On January 29, 2010, 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 reviewed 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 their CAP in accordance with station procedures. Specific documents reviewed during this inspection are listed in the Attachment. The inspectors' reviews focused specifically on the following plant systems due to their risk significance or susceptibility to cold weather issues:

- "A" and "B" Emergency Diesel Generators (EDGs)
- Plant and Instrument Air Compressors

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Quarterly Partial System Walkdowns

a. Inspection Scope

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

- Electrical Switchyard while it was protected due to inoperability of the "A" EDG on January 13, 2010;
- Instrument air system while the "B" and "C" Air Compressors were unavailable due to maintenance and troubleshooting on February 8, 2010; and
- "B" DC Electrical Distribution system following planned maintenance on February 17, 2010.

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, applicable portions of 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 system components and support equipment were aligned correctly and 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. Documents reviewed are listed in the attachment.

The inspectors reviewed the following ARs associated with this area to verify that the licensee identified and implemented appropriate corrective actions:

- AR #382202, Full Electrolyte Level Indication on Safety Related Batteries is Difficult to Interpret
- AR #381075, Temporary Air Compressor Load/Unload Setpoint

b. Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Quarterly Resident Inspector Tours

a. Inspection Scope

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

- Reactor Auxiliary Building (RAB) Exhaust Fan Area;
- “A” Cable Spreading Room;
- “B” Cable Spreading Room;
- “A” Switchgear and Battery Rooms and Non-Safety Battery Room;
- “B” Switchgear and Battery Rooms and Alternate Control Panel Room; and
- Diesel Fuel Oil Storage Building Yard Area and the “A” and “B” Diesel Fuel Oil Transfer Pump Rooms.

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 fire risk as documented in the plant’s Individual Plant Examination of External Events with later additional insights, 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 were determined to be in satisfactory condition. The inspectors also verified that minor issues identified during the inspection were entered into the licensee’s CAP.

The inspectors reviewed the following ARs associated with this area to verify that the licensee identified and implemented appropriate corrective actions:

- AR #379214, Cigarette Butts Found in Diesel Fuel Oil Storage Building;
- AR #381036, Interam Fire Wrap Outer Layer Not Secured Properly;
- AR #381714, Auxiliary Control Panel Interam Fire Wrap Outer Layer not Properly Secured; and
- AR #383957, B.5.b Battery Chargers Found with Tripped Surge Protectors.

b. Findings

No findings of significance were identified.

.2 Annual Fire Protection Drill Observation

a. Inspection Scope

On February 19, 2010, the inspectors observed fire brigade performance during a simulated oil fire in the safety related "B" Startup Transformer. The observation was used to determine the readiness of the plant fire brigade to fight fires in safety related equipment. The inspectors verified that the licensee staff identified deficiencies, openly discussed them in a self-critical manner at the drill debrief, and took appropriate corrective actions. Specific attributes evaluated were:

- Proper wearing of turnout gear and self-contained breathing apparatus;
- Proper use and layout of fire hoses;
- Employment of appropriate fire fighting techniques;
- Sufficient firefighting equipment brought to the scene;
- Effectiveness of fire brigade leader communications, command, and control;
- Utilization of pre planned strategies;
- Adherence to the pre planned drill scenario; and
- Fulfillment of drill objectives.

The inspectors reviewed the following AR associated with this area to verify that the licensee identified and implemented appropriate corrective actions:

- AR #382522, Fire Pre-Plan Improvement Item to Prevent Confusion when Applying Foam to Liquid Combustible Fires

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

Unresolved Item (URI) 05000400/2009005-02, "A" ESW Pump Power Supply Cables Submerged in Water.

a. Inspection Scope

In the fourth quarter 2009, the inspectors examined potential degradation of power cables for the "A" emergency service water (ESW) pump due to submergence in water. The inspectors opened an URI for this issue pending the licensee's evaluation of the basis for qualification of these cables in submerged conditions. This inspection was conducted to evaluate that information.

b. Findings

Introduction: The inspectors identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," in that the licensee failed to maintain the "A" ESW pump power cables in an environment for which they were designed. Specifically, the cables were submerged in water, a condition for which they were not qualified.

Description: On December 10, 2009, the inspectors observed the opening of manway 73B-SA, to complete the NRC baseline inspection activities. This manway included cables from the ESW Building to the Class 1E power supply. During the activity, the inspectors noted that the “A” ESW pump power supply cables were submerged in approximately 2.5 feet of standing water. As immediate corrective actions, the licensee pumped the manway dry.

A review of the licensing basis and licensing documentation revealed the cables were selected and purchased for dry or wet conditions. After discussions with additional NRC specialists, the inspectors determined that a cable designed for wet conditions includes water resistance but does not include continuous submerged conditions.

The actual environmental conditions in the manway can be dry, wet, and submerged in water. A review of the licensee’s underground cable duct drawings showed that the manway was constructed below grade and expected to accumulate water. The original design of the plant included sump pump installation in the manway to prevent this condition; however during final construction, the sump pumps were removed from the design and the licensee failed to provide a means to maintain the cables in their rated environment. Preventive Maintenance (PM) activities were developed to pump down the manways that contain 6.9kV safety related cables. Additionally, PMs were developed to conduct cable insulation testing on all of the 6.9kV wetted cables to monitor and trend degradation. Currently not all of the safety related 6.9kV cable insulation testing PMs have been completed. The remaining PMs are scheduled to be completed this year. Although the cables were submerged, the inspectors concluded that there was not an immediate operability concern because the licensee had satisfactory test results from the completed cable insulation testing on the “A” ESW pump.

For corrective actions, the frequency for completing the PMs to inspect the manways for water and to pump out the water, as needed, is being evaluated and revised. The licensee plans to complete the remaining safety related cable insulation testing PMs during the upcoming refueling outage in October 2010. In addition, engineering personnel were evaluating permanent solutions to prevent the manways from filling with water, which would eliminate the need for manual pumping.

Analysis: The inspectors determined that failure to ensure that the “A” ESW pump power cables were maintained in an environment for which they were designed was a performance deficiency. The finding was more than minor because, if left uncorrected, it had the potential to lead to a more significant safety concern. The finding affected the equipment performance attribute of the Mitigating Systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, it could have caused the “A” ESW pump to become inoperable in the event that the cable failed due to long term degradation as a result of continuous submergence. The inspectors evaluated the significance of this finding using IMC 0609, “Significance Determination Process,” Phase

1 Worksheet. The finding was of very low safety significance because it was a qualification deficiency that did not result in a loss of operability. This finding was determined to have a cross-cutting aspect in the CAP component of the Problem Identification and Resolution area associated with timely and effective corrective actions. Specifically, corrective actions to adjust the frequency of inspecting underground manways for water were not implemented in a timely manner to address safety-related cables from repeatedly being submerged. (P.1(d)).

Enforcement: 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” requires, in part, that measures shall be established to ensure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. Contrary to the above, in December 2009 the licensee failed to maintain safety-related cables in an environment for which they were designed. Specifically, the cables in manway 73B-SA were designed to be moisture resistant, but not designed to be completely submerged in water. As immediate corrective action, the licensee pumped the manway dry. Because this finding was of very low safety significance, and it was entered into the licensee’s CAP as AR #376709, this violation is being treated as an NCV, consistent with the NRC Enforcement Policy and is designated as NCV 05000400/2010002-03, “A” ESW Pump Power Supply Cables Submerged in Water.

1R11 Licensed Operator Requalification Program

.1 Quarterly Review

a. Inspection Scope

On March 9, 2010, the inspectors observed a crew of licensed operators in the plant’s simulator during licensed operator requalification examinations to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems and training was being conducted in accordance with licensee procedures. The licensed operators responded to a main turbine generator faulty voltage regulator, steam generator tube rupture, rod control malfunction, pressurizer power operated relief valve failure and a main turbine generator manual trip. 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 Emergency Plan 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 of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the systems associated with the following components. In addition, the inspectors verified maintenance effectiveness issues were entered into the CAP with the appropriate significance characterization. The inspectors evaluated degraded performance issues involving the following three risk significant components:

- AR #379340, Wire Terminal Found Disconnected in 1RH-31 ("A" RHR pump mini-flow valve) Switch Compartment;
- AR #370899, Potential Past Operability Concern with Reactor Auxiliary Building Emergency Exhaust Damper #29; and
- AR #380543, Unplanned Inoperability of the "B" EDG.

The inspectors focused on the following attributes:

- 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 structures, systems, and components (SSCs)/functions classified as (a)(2) or appropriate and adequate goals and corrective actions for systems classified as (a)(1).

The inspectors reviewed the following ARs associated with this area to verify that the licensee identified and implemented appropriate corrective actions:

- AR #374541, "A" ESW Booster Pump Breaker will not Rack-in;
- AR #375126, "C" Air Compressor Trip;
- AR #385497, Incorrect Setting on Agastat Time Delay Drop Out Relay;
- AR #383822, Damper-21 Needed Slight Assistance at Open Limit; and
- AR #385821, B Sequencer Loose Terminal Screws.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors reviewed the licensee's evaluation and management of plant risk for the five 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:

- Qualitative yellow risk condition during a downpower to 60% combined with “A” ESW inoperability resulting in a quantitative yellow situation for Containment Fan Cooler 3 (AH-3) repair on January 13, 2010;
- Yellow risk condition for Total Reactor Makeup Water Flow Indicating Switch 114 (FIS-114) repair and 1CS-155 (isolation valve for makeup to the Volume Control Tank) work on February 1, 2010;
- Elevated green risk condition while the “B” EDG was inoperable for troubleshooting on 1DFO-191 (Fuel Oil Day Tank Isolation Valve) on February 18, 2010;
- Yellow risk condition during a scheduled surveillance with “B” Main Feed Regulating Valve in manual control on March 2, 2010; and
- Elevated green risk condition while the “B” EDG was inoperable for scheduled maintenance on March 17, 2010.

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.

The inspectors reviewed the following AR associated with this area to verify that the licensee identified and implemented appropriate corrective actions:

- AR #378949, Improvement Opportunity for Yellow Risk Activities

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors selected the following six 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 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 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 selected the following six potential operability issues:

- AR #375529, Foreign Material Concern in Boric Acid Batch Tank;
- AR #376824, Turbine Driven Auxiliary Feed Pump (TDAFW) Differential Pressure Response Time Exceeds Limits;
- AR #381734, "B" Battery Room Temperature High Out of Band;
- AR #367900, "B" EDG Control Panel High Temperature Alarm;
- AR #367901, "B" EDG, the Wrong Cotter Pin Was Used; and
- AR #360019, Guidance for Loss of Reactor Primary Shield Cooling.

The inspectors reviewed the following ARs associated with this area to verify that the licensee identified and implemented appropriate corrective actions:

- AR #368733, "A" ESW Pump Seal and Bearing Flow Lower than Expected
- AR #385171, "B" Chiller Pressure Relief Valve Actuator not in Desired Position

b. Findings

No findings of significance were identified.

1R18 Plant Modifications

a. Inspection Scope

The following engineering design package was reviewed and selected aspects were discussed with engineering personnel:

- Temporary modification, Engineering Change (EC) 76187 to revise the Core Operating Limits Report (COLR)

This document and related documentation were reviewed for adequacy of the associated 10 CFR 50.59 safety evaluation screening, consideration of design parameters, implementation of the modification, post-modification testing, and relevant procedures, design, and licensing documents were properly updated. The inspectors observed ongoing and completed work activities to verify that installation was consistent with the design control documents. The modification relaxed the requirements for the minimum number of operable detector thimbles from 38 to 25 for the remainder of the operating cycle. Compensatory measures were put into place to decrease the peaking factor limits and increase the number of detector thimbles per quadrant of the core.

The inspectors reviewed the following ARs associated with this area to verify that the licensee identified and implemented appropriate corrective actions:

- AR #385493, Minimum Number of Thimbles not Obtained During Flux Map;
- AR #373563, Setpoint Change for Reactor Supports Area Temperatures; and
- AR #378363, Pressure Switch Sensing Lines not Vented per EC 60063 (Pressure Switch Replacement).

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing

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:

Test Procedure	Title	Related Maintenance Activity	Date Inspected
OST-1046 and OST-1808	Main Steam Isolation Valve (MSIV) Operability Test Quarterly Interval Mode 3 to 5 and Main Steam Isolation: Engineered Safety Features Response Time 18 Month Interval Modes 3 – 5	Work Order (WO) 1655708, “B” MSIV Failed to Completely Close	February 23, 2010 (PMT occurred November 20, 2009, however some inspection material was not available until this quarter)

OST-1010	Containment Cooling System Operability Test Monthly Interval	Repair of Containment Cooler (AH-3)	January 13
OST-1040	Essential Services Chilled Water Systems Operability Quarterly Interval	Filter and Regulator Replacement on 1CH-116 Emergency Services Chilled Water Supply	January 14
OST-1124	Train "B" 6.9 kV Emergency Bus Undervoltage Trip Actuating Device Operational Test And Contact Check Modes 1-6	WO 1506763, Replacement of a Turbine Driven Auxiliary Feedwater Pump Driver Card that Controls Main Control Board (MCB) Indication	January 21
OST-1041	"A" Train HVAC Safety Related Essential Services Chilled Water Temperature Control Valves In Service Testing Operability Test Quarterly Interval Modes: 1-6	WO 1152333, Replace Solenoid Valve on 1CH-279 and WO 698105, 1CH-279 - Replace Solenoid and Positioner	March 26
OP-112	Containment Spray System	WO 1349680, Replace "A" Containment Spray Pump Breaker per PM-E0044, 480 VAC Siemens Type RLN(F) Load Center Breaker and Cubicle Preventative Maintenance	March 30

These activities were selected based upon the structure, system, or component'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 TS and the UFSAR to ensure 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. Documents reviewed are listed in the attachment.

The inspectors reviewed the following ARs associated with this area to verify that the licensee identified and implemented appropriate corrective actions:

- AR #366175, “B” MSIV Failed to Completely Close;
- AR #389348, AH-12-SA (A Switchgear Room Air Handler) Declared Inoperable;
- AR #389374, 1CH-279 (AH-12-SA Temperature Control Valve) Failed Code Criteria During OST-1041;
- AR #389349, Temperature Limits Exceeded for “A” Battery Room;
- AR #374226, Unexpected Chiller Equipment Response; and
- AR #378378, EC 75427 (“A” EDG Output Breaker Configuration) Post Maintenance Test did not Pass.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

.1 Routine Surveillance Testing

a. Inspection Scope

For the three surveillance tests below, the inspectors observed the surveillance tests and/or reviewed the test results for the following activities to verify the tests met TS surveillance requirements, UFSAR commitments 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.

- OST-1122, Train “A” 6.9 KV Emergency Bus Undervoltage Trip Activating Device Operational Test on January 5, 2010;
- OPT-1512, Essential Chilled Water Turbopak Units Quarterly Inspection on January 8, 2010; and
- OST-1004, Power Range Heat Balance, Computer Calculation, Daily Interval, Mode 1 (Above 15% Power) on February 12, 2010.

The inspectors reviewed the following ARs associated with this area to verify that the licensee identified and implemented appropriate corrective actions:

- AR #382195, MST-I0304 (Reactor Auxiliary Building Service Water Return Flow Loop Calibration) as Left Flow Indication Over-Ranged
- AR #384062, Targets Failed to Trip During Testing

b. Findings

No findings of significance were identified.

.2 In-service Testing (IST) Surveillance

a. Inspection Scope

The inspectors reviewed the performance of OST-1191, Steam Generator Power Operated Relief Valves (PORV) and Isolation Valve Operability Test Quarterly Interval Modes 1 – 4 on January 22, 2010, to evaluate the effectiveness of the licensee's American Society of Mechanical Engineers (ASME) Section XI testing program for determining equipment availability and reliability. This surveillance satisfies the IST requirements for the Steam Generator PORVs and the associated PORV isolation valves. The inspectors evaluated selected portions of the following areas:

- Testing procedures and methods;
- Acceptance criteria;
- Compliance with the licensee's IST program, TS, selected licensee commitments, and code requirements; and
- Required corrective actions.

The inspectors reviewed the following ARs associated with this area to verify that the licensee identified and implemented appropriate corrective actions:

- AR #376072, "B" ESW Screen Wash Pump has negative Differential Pressure Trend
- AR #382650, 1CH-126 (AH-30 Return Isolation Valve) Stroke Time Outside Code Criteria

b. Findings

No findings of significance were identified.

1EP2 Alert and Notification System Evaluation

a. Inspection Scope

The inspectors evaluated the adequacy of the licensee's methods for testing the alert and notification system in accordance with NRC Inspection Procedure 71114, Attachment 02, "Alert and Notification System (ANS) Testing". The applicable planning standard, 10 CFR Part 50.47(b)(5) and its related 10 CFR Part 50, Appendix E, Section IV.D requirements were used as reference criteria. The criteria contained in NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response

Plans and Preparedness in Support of Nuclear Power Plants," Revision 1, were also used as a reference.

The inspectors reviewed various documents which are listed in the Attachment. This inspection activity satisfied one inspection sample for the alert and notification system on a biennial basis.

b. Findings

No findings of significance were identified.

1EP3 Emergency Preparedness Organization Staffing and Augmentation System

a. Inspection Scope

The inspectors reviewed the licensee's Emergency Response Organization (ERO) augmentation staffing requirements and process for notifying the ERO to ensure the readiness of key staff for responding to an event and timely facility activation. The qualification records of key position ERO personnel were reviewed to ensure all ERO qualifications were current. A sample of problems identified from augmentation drills or system tests performed since the last inspection was reviewed to assess the effectiveness of corrective actions.

The inspection was conducted in accordance with NRC Inspection Procedure 71114, Attachment 03, "Emergency Preparedness Organization Staffing and Augmentation System." The applicable planning standard, 10 CFR 50.47(b)(2), and its related 10 CFR 50, Appendix E requirements were used as reference criteria.

The inspectors reviewed various documents which are listed in the Attachment to this report. This inspection activity satisfied one inspection sample for the ERO staffing and augmentation system on a biennial basis.

b. Findings

No findings of significance were identified.

1EP4 Emergency Action Level and Emergency Plan Changes

a. Inspection Scope

Since the last NRC inspection of this program area, Revisions 53 and 54 of the Emergency Plan were implemented based on the licensee's determination, in accordance with 10 CFR 50.54(q), that the changes resulted in no decrease 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 inspectors conducted a sampling review of the Plan changes and implementing procedure changes made between January 1, 2009, and January, 2010, to evaluate potential decreases in effectiveness of the Plan. However, this review was not documented in a Safety Evaluation Report and does not constitute formal NRC approval of the changes. Therefore, these changes remain subject to future NRC inspection in their entirety.

The inspection was conducted in accordance with NRC Inspection Procedure 71114, Attachment 04, "Emergency Action Level and Emergency Plan Changes." The applicable planning standard, 10 CFR 50.47(b)(4) and its related 10 CFR 50, Appendix E requirements were used as reference criteria.

The inspectors reviewed various documents which 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 of significance were identified.

1EP5 Correction of Emergency Preparedness Weaknesses

a. Inspection Scope

The inspectors reviewed the corrective actions identified through the Emergency Preparedness program to determine the significance of the issues and to determine if repeat problems were occurring. The facility's self-assessments and audits were reviewed to assess the licensee's ability to be self-critical, thus avoiding complacency and degradation of their emergency preparedness program. In addition, the inspectors reviewed licensee self-assessments and audits to assess the completeness and effectiveness of all emergency preparedness related corrective actions.

The inspection was conducted in accordance with NRC Inspection Procedure 71114, Attachment 05, "Correction of Emergency Preparedness Weaknesses." The applicable planning standard, 10 CFR 50.47(b)(14) and its related 10 CFR 50, Appendix E requirements were used as reference criteria.

The inspectors reviewed various documents which are listed in the Attachment to this report. This inspection activity satisfied one inspection sample for the correction of emergency preparedness weaknesses on a biennial basis.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

.1 Initiating Events Performance Indicators

a. Inspection Scope

To verify the accuracy of the PI data reported to the NRC, the inspectors compared the licensee's basis in reporting each data element to the PI definitions and guidance contained in Nuclear Energy Institute (NEI) Document 99-02, Regulatory Assessment Performance Indicator Guideline.

Initiating Events Cornerstone

- Unplanned Scrams per 7000 Critical Hours;
- Unplanned Power Changes per 7000 Critical Hours; and
- Unplanned Scrams with Complications.

The inspectors sampled licensee submittals for the performance indicators listed above for the period from the first quarter 2009 through the fourth quarter 2009. The inspectors reviewed the licensee's operator narrative logs, issue reports, event reports and NRC inspection reports for the period to validate the accuracy of the submittals. Specific documents reviewed are described in the Attachment to this report.

b. Findings

No findings of significance were identified.

.2 Emergency Preparedness Performance Indicators

a. Inspection Scope

The inspectors sampled licensee submittals relative to the PIs listed below for the period January 1, 2009, and December 31, 2009. To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in NEI 99-02, "Regulatory Assessment Performance Indicator Guideline", was used to confirm the reporting basis for each data element.

Emergency Preparedness Cornerstone

- Emergency Response Organization (ERO) Drill/Exercise Performance;
- ERO Drill Participation; and
- Alert and Notification System Reliability.

For the specified review period, the inspector examined data reported to the NRC, procedural guidance for reporting PI information, and records used by the licensee to identify potential PI occurrences. The inspectors verified the accuracy of the PI for ERO drill and exercise performance through review of a sample of drill and event records. The inspectors reviewed selected training records to verify the accuracy of the PI for ERO drill participation for personnel assigned to key positions in the ERO. The inspectors verified the accuracy of the PI for alert and notification system reliability through review of a sample of the licensee's records of periodic system tests. The inspectors also interviewed the licensee personnel who were responsible for collecting and evaluating the PI data. Licensee procedures, records, and other documents reviewed within this inspection area are listed in the Attachment. This inspection satisfied three inspection samples for PI verification on an annual basis.

b. Findings

No findings of significance 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 of significance were identified.

.2 Selected Issue Follow-up Inspection: "Failure to Submit a Licensee Event Report for a Condition Prohibited by Technical Specifications Associated with the "B" Emergency Service Water Discharge Valve"

a. Inspection Scope

The inspectors selected AR #361821, Potential Error in AR #358062 Reportability Review Assignment, for detailed review. This AR investigated a missed maintenance rule functional failure associated with the "B" Emergency Service Water Discharge Valve. The inspectors reviewed this report to verify that the licensee identified the full extent of the issue, performed an appropriate evaluation, and specified and prioritized appropriate corrective actions. The inspectors evaluated the report against the requirements of the licensee's CAP as delineated in corporate procedure CAP-NGGC-0200, Corrective Action Program, and 10 CFR 50, Appendix B.

The inspectors reviewed the following ARs associated with this area to verify that the licensee identified and implemented appropriate corrective actions:

- AR #358062, Missed Maintenance Rule Functional Failure
- AR #365286, Clearance Impact on 1SW-271

b. Findings

Introduction: The inspectors identified a Severity Level IV, non-cited violation (NCV) of 10 CFR 50.73(a)(2)(i)(B) due to the licensee's failure to recognize that the inability of the "B" ESW Discharge Valve (1SW-271) to open on the start of "B" ESW pump caused a reportable condition. Consequently, the licensee failed to submit a licensee event report (LER) within 60 days as required by 10 CFR 50.73.

Description: On October 19, 2007, while in Mode 5, 1SW-271 failed to open on the start of the "B" ESW pump. This valve is required to open on the start of the "B" ESW pump to provide a discharge path for the cooling water. Operators immediately stopped the "B" ESW pump and aligned normal service water to the safety related components in the "B" train. The licensee determined that the auto open controls for 1SW-271 had been disabled by a clearance order for unrelated work. Although the "B" ESW train is not required to be operable in Mode 5, the components cooled by "B" ESW, such as "B" EDG and "B" RHR, were being relied upon as protected train equipment. Therefore, the "B" ESW train was necessary to ensure core decay heat removal in the event that off-site power was not available. NRC inspectors wrote a self-revealing NCV of TS 6.8.1, "Programs and Procedures," for an inadequate clearance order as documented in NRC Integrated Inspection Report 05000400/2007005.

In October, 2009, a Problem Identification and Resolution (PI&R) inspection team identified that the licensee's past operability evaluation stated this condition was not reportable since operators were able to open this valve manually from the control room. The team questioned whether the operators would be able to open the valve within one minute, which is required to ensure cooling to the EDGs during an accident. The team also determined that when the valve is manually opened by the reactor operators from the control room, that the valve would automatically go closed due to the inadequate clearance. As a result of the team's questions, the licensee wrote AR #361821 to address this issue. This issue was considered unresolved (05000400/2009006-03) pending additional NRC review of the evaluation of the failure.

Subsequently, the inspectors identified that the licensee's past operability evaluation (AR #358062) performed to address the PI&R team's question was in error. The "B" RHR train was inoperable from the time the clearance was in place until the time when 1SW-271 was opened with control power removed; a total of 13 hours and 24 minutes. During this time, the plant was in Mode 5 (cold shutdown) and transitioned from loops not filled to loops filled. For these conditions, "B" ESW was not functional to support operability of the "B" RHR train. However, "B" ESW was not functional which rendered the "B" RHR system inoperable. This resulted in the licensee violating TS 3.4.1.4.1 (Mode 5 with loops filled) and TS 3.4.1.4.2 (Mode 5 with loops not filled), which was a reportable event. As corrective action, the licensee revised the past operability evaluation, submitted LER 05000400/2010-001 and performed a priority one investigation.

Analysis: The licensee's failure to recognize that the inability of 1SW-271 to open caused a reportable condition and submit an LER as required by 10 CFR 50.73 was a performance deficiency. This issue was dispositioned as traditional enforcement, instead of the Significance Determination Process, because it had the potential for impacting the NRC's ability to perform its regulatory function. However, because this violation was of very low safety significance, was not repetitive or willful, and was entered into the licensee's CAP as AR #361821 and AR #358062, the NRC has characterized the significance of this reporting violation as a Severity Level IV NCV in accordance with section IV.A.3 and supplement I of the NRC Enforcement Policy. The cause of this event was directly related to the cross-cutting aspect in the area of problem identification and resolution within the CAP component because the licensee did not adequately identify the need to submit an LER per the requirements of 10 CFR 50.73. (P.1(c))

Enforcement: 10 CFR 50.73 requires licensees to submit an LER for any operation or condition which was prohibited by TS within 60 days of discovering the event. Contrary to the above, the licensee failed to submit a report within 60 days of October 19, 2007, when the event associated with the inability of 1SW- 271 to remain open was discovered. Because this violation was of very low safety significance, was not repetitive or willful and it was entered into the licensee's corrective action program, this violation is being treated as an NCV, consistent with the NRC Enforcement Policy and is designated as NCV 05000400/2010002-01: "Failure to Submit a Licensee Event Report for a Condition Prohibited by Technical Specifications Associated with the "B" Emergency Service Water Discharge Valve."

.3 Selected Issue Follow-up Inspection: "Failure to Promptly Evaluate Operating Experience and Identify Potential Steam Voiding as a Condition Adverse to Quality"

a. Inspection Scope

The inspectors selected AR #345425, Potential Steam Voids in the RHR System, for detailed review. This AR was associated with the evaluation of industry OE highlighting potential steam voiding issues in the RHR system. The inspectors reviewed this report to verify that the licensee identified the full extent of the issue, performed an appropriate evaluation, and specified and prioritized appropriate corrective actions. The inspectors evaluated the report against the requirements of the licensee's CAP as delineated in corporate procedure CAP-NGGC-0200, Corrective Action Program, and 10 CFR 50, Appendix B.

The inspectors reviewed the following ARs associated with this area to verify that the licensee identified and implemented appropriate corrective actions:

- AR #302656, OE27625 - RHR System Inoperability in Modes 3 and 4;
- AR #306234, Industry OE for RHR Trains Inoperable During Mode Changes;
- AR #317222, OE Review Assignments Greater than 4 Weeks Old;
- AR #333882, OE Evaluation Over 470 Days Old;
- AR #368628, Risk Rank OE to Aid in Evaluating Greatest Potential Impact First; and

- AR #369182, Review of Westinghouse Nuclear Safety Advisory Letters for Impacts.

b. Findings

Introduction: The inspectors identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion XVI, Corrective Action, when the licensee failed to promptly evaluate OE received October 22, 2008 and identify potential steam voiding as a condition adverse to quality. Specifically, the evaluation was not completed until July 16, 2009, thereby delaying corrective actions and resulting in the licensee violating TS 3.0.4 while transitioning from Mode 4 to Mode 1 on May 9, 2009.

Description: The RHR system is a two train system with two primary purposes. First, RHR can be aligned to the reactor coolant system (RCS) to provide decay heat removal (DHR) while the reactor is shutdown with temperatures as high as 350°F. Secondly, each train of RHR can be aligned to serve as a low pressure water injection source to the RCS for an associated train of the Emergency Core Cooling System (ECCS) in the event of a large Loss of Coolant Accident (LOCA). In the ECCS mode, the suction source for the RHR pumps would be either the Refueling Water Storage Tank or the ECCS sump in containment. The pressure of these sources of water would be significantly below the suction pressure the RHR pump would experience while in the DHR mode. If a large LOCA were to occur, RHR would be realigned for the ECCS mode. If RHR had recently been used in the DHR mode, the water in the suction line to the RHR pump could be hot enough to flash to steam at the new lower suction pressure. If the steam void flowed to an RHR pump, that pump could fail causing the associated train of ECCS to fail. This phenomenon could occur at suction temperatures as low as 240°F.

On October 22, 2008, the licensee entered OE #27625, RHR System Inoperability in Modes 3 and 4 due to Potential Suction Line Steam Voiding, into the CAP. In accordance with CAP-NGGC-0202, Operating Experience Program, the licensee created an OE Review (OER) assignment to be completed. An OER is used to determine if the OE is applicable to the site and if a follow-up evaluation is required. Although CAP-NGGC-0202 states that OER assignments should typically be issued with due dates of two weeks or up to potentially four weeks, the initial due date was scheduled for December 4, 2008. After two due date extensions were requested, the OER was converted to an OE Evaluation (OEE) as dictated by CAP-NGGC-0202 on February 9, 2009.

An OEE is a formal evaluation that is performed when it is determined that OE is applicable to the site, with the goal of learning lessons from events throughout the industry. CAP-NGGC-0202 prescribes that OEE assignments are typically expected to be completed within 90 days. The due date for the OEE was also extended three times before being completed on July 16, 2009. The conclusion of this OEE was that an adverse condition existed in that the RHR pump suction lines were potentially susceptible to steam voiding at temperatures as low as 240°F. This conclusion resulted in the licensee implementing corrective actions through procedural guidance to prevent steam voiding in susceptible areas.

Nearly six months after receiving the OE but prior to completing the formal evaluation, the licensee started refueling outage 15 (RFO-15) on April 17, 2009. When returning to normal operation on May 9, 2009, the “B” RHR train was being used in the DHR mode. It was secured in Mode 4 when the “B” RHR pump suction line temperature reached 321°F, making it susceptible to steam voiding if it was transitioned to the ECCS mode of operation. With the “B” ECCS train inoperable, the licensee proceeded to Mode 1 operation.

TS 3.5.3 requires only one train of ECCS to be operable in Mode 4. However, TS 3.5.2 requires both trains of ECCS to be operable in Modes 3, 2, and 1. If this condition can not be met, the licensee must enter an associated action statement that requires a shutdown if compliance is not restored. By proceeding into Modes 3, 2, and 1 with one train of ECCS inoperable, the licensee violated the requirements of TS 3.0.4. TS 3.0.4 prohibit entry into a Mode when the limiting conditions for operation are not met and the action statement requires a shutdown. The condition existed for approximately 70 hours, until the suction line cooled below 240°F.

Analysis: Failing to promptly evaluate OE received October 22, 2008, and identify potential steam voiding as a condition adverse to quality was identified as a performance deficiency. This performance deficiency resulted in the licensee violating the requirements of TS 3.0.4 on May 9, 2009. The performance deficiency was more than minor because it was associated with the equipment performance attribute of the Mitigating Systems Cornerstone and it adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, it could potentially cause one or more RHR pumps and associated ECCS trains to be inoperable in the event that the steam voiding travelled to the pump.

Using IMC 0609, “Significance Determination Process,” Phase 1 Worksheet, the inspectors concluded that a Phase 2 evaluation was required because this finding represented a potential loss of safety function of the RHR system. The inspectors performed a Phase 2 analysis using IMC 0609, Appendix A, “Determining the Safety Significance of Reactor Inspection Findings for At-Power Situations” and the site specific risk informed inspection notebook. Due to the site specific risk informed inspection notebook not containing appropriate target sets to accurately estimate the risk input of the finding, it was determined that a Phase 3 analysis was required.

A regional Senior Reactor Analyst performed a Phase 3 evaluation under the Significance Determination Process and concluded the finding was of very low safety significance (Green). The NRC’s most current Standardized Plant Analysis Risk Model was used for the evaluation. The evaluation assumed that the “B” RHR Pump always failed to start for the exposure time of seventy hours. Also, there was a potential increase in the common cause failure of the RHR pumps. The dominant accident sequence was a postulated Small Break LOCA with initial success of the ECCS via High Pressure Injection, but the ECCS failed in the recirculation mode. The SDP performed for this violation considered the potential loss of safety function of the RHR system and therefore bounded all violations described in LER 05000400/2009-002 which is further discussed in Section 4OA3.2.

This finding was determined to have a cross-cutting aspect in the OE component of the Problem Identification and Resolution area, in that the licensee failed to promptly evaluate OE (P.2(a)).

Enforcement: 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that conditions adverse to quality shall be promptly identified and corrected. Contrary to this requirement, the licensee failed to promptly evaluate OE and identify potential steam voiding in the RHR suction lines as a condition adverse to quality. Specifically, this resulted in the violation of TS 3.0.4 on May 9, 2009 when the licensee entered Modes 3, 2 and 1 without meeting the requirements of TS 3.5.2. The condition existed for approximately 70 hours, until the suction line cooled below 240°F.

The licensee took corrective action by changing procedures to avoid exposing the applicable portions of piping to excessive temperatures during Modes when it is required to be operable for ECCS, thereby preventing steam voiding. Because the finding is of very low safety significance and has been entered into the licensee's CAP as AR #345425, this violation is being treated as an NCV consistent with the Enforcement Policy and is designated as NCV 05000400/2010002-02, "Failure to Promptly Evaluate Operating Experience and Identify Potential Steam Voiding as a Condition Adverse to Quality."

4OA3 Follow-up of Events

.1 (Closed) LER 05000400/2010-001, Clearance Error Results in Equipment Becoming Inoperable

This LER documents a condition that occurred during Refueling Outage 14, on October 18, 2007. Specifically, a clearance was hung which inadvertently caused the required number of operable and operating RHR loops to be less than that required by TS 3.4.1.4.1 due to "B" ESW being non-functional. This issue is further discussed along with enforcement aspects in Section 4OA2.2 of this report. This condition has been entered into the licensee's CAP as AR #365286, Clearance Impact of "B" ESW Discharge Isolation Valve. This LER is closed.

.2 (Closed) LER 05000400/2009-002, Potential for RHR Trains to Be Inoperable During Mode Changes

This LER documents the susceptibility of the RHR system to steam voiding due to reduced suction pressure in the ECCS mode of operation. The licensee's investigation revealed that over the past three years, eleven violations of TS 3.0.4, TS 3/4.5.2 and TS 3/4.5.3 occurred. The inspectors reviewed these violations, including the most limiting case where both trains were determined to be inoperable resulting in the potential loss of safety function for the RHR system. The potential loss of safety function occurred on three occasions; September 29, 2007, August 13, 2008 and August 18, 2008. For the ten violations of TS that occurred prior to October 22, 2008, no performance deficiency was identified because the inspectors determined that the cause was an initial design flaw and was not reasonably within the licensee's ability to foresee and correct.

The violation of TS 3.0.4 that occurred after October 22, 2008 was determined to have a performance deficiency. Additionally, to conservatively estimate the risk of this issue the inspectors considered the potential loss of safety function for the RHR system in the SDP. Therefore, the SDP performed for the violation associated with this performance deficiency bounded all violations described in this LER, and determined that the significance was Green. This violation and SDP analysis are further discussed in Section 4OA2.3 of this report.

This condition has been entered into the licensee's CAP as AR #345425, Potential Steam Voids in the RHR System. The licensee took corrective action by changing procedures to avoid exposing the applicable portions of piping to excessive temperatures during Modes when it is required to be operable for ECCS, thereby preventing steam voiding. The inspectors reviewed the licensee's assessment and corrective actions for the event, and determined they were appropriate. 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. 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 of significance were identified.

.2 (Closed.) URI 05000400/2009003-01. Review the Significance of the Cooling Tower Blowdown Line Pathway Dose Compared to Doses from All Other Pathways.

An unresolved item (URI) was identified regarding the significance of leakage from a cooling tower blowdown line (CTBL) used to transport radioactive effluents. The licensee discharges permitted and monitored radioactive liquid effluents into the CTBL for dilution prior to release into Harris Lake. On December 15, 2008, the licensee observed water in Air Relief System Manhole (ARSM) No. 2 located on the CTBL upstream from the permitted release point. The licensee obtained water samples from ARSM No. 2 for analysis and identified tritium levels ranging from less than the detection limit to 2,120 picoCuries per liter (pCi/L). On May 1, 2009, the inspectors noted that the leakage could constitute an unanalyzed exposure pathway to a member of the public (via groundwater) and opened the URI.

The Offsite Dose Calculation Manual (ODCM) states that radioactive materials released in liquid effluents to unrestricted areas are required to demonstrate compliance with 10 CFR Part 50 Appendix I. Appendix I annual limits are 3 millirem to the total body or 10 millirem to any organ. In addition, Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I" specifies that exposure pathways that may arise due to unique conditions at a specific site should be considered if they are likely to provide a significant contribution to the total dose. A significant pathway is considered one whose additional dose increment is equal to or greater than ten percent of the total from all pathways.

The licensee contracted a vendor to construct groundwater monitoring wells and perform an analysis of the hydrological transport properties in the vicinity of the CTBL leaks. The licensee also performed calculations of doses to hypothetical members of the public through alternate release pathways as a result of the leaking CTBL. The location of the leakage was on a peninsula projecting into Harris Lake and groundwater transport studies showed that any contamination would ultimately migrate toward the lake (the permitted release location). Local vegetation was analyzed and no tritium was detected, thereby showing that vegetation-human ingestion or vegetation-animal-human ingestion pathways are not significant. Although no drinking wells are located in the vicinity of the contaminated plume, and there is no credible pathway to the public, calculations were performed to conservatively estimate the dose to a member of the public who used the contaminated water as their primary drinking water for an entire year. The results were below 10 CFR Part 50 Appendix I annual limits. The inspectors noted that tritium was the only reactor-produced radionuclide that was detected in the monitoring well and environmental media samples. The inspectors also noted that all tritium levels discovered in the CTBL leakage plume and the levels currently existing in Harris Lake are below EPA limits for safe drinking water (<20,000 pCi/L). Through review of licensee documents and discussions with licensee personnel and Nuclear Reactor Regulation (NRR) staff, the inspectors determined that no new significant exposure pathways were created as a result of the CTBL leakage. The licensee has initiated corrective actions that include replacement of the leaking CTBL with new piping.

.3 <u>(Closed) URI 05000400/2009006-03, Unresolved Item Associated with the Evaluation of the Failure of Emergency Service Water Valve 271.</u>

In inspection report 05000400/2009006, an URI was identified associated with the evaluation of the failure of 1SW-271 ("B" Emergency Service Water Discharge Valve). Specifically, this URI was opened to enable continued inspection of whether or not this issue was reportable. The LER that resulted from this continued inspection is addressed in Section 4OA3.1 and the issue is further discussed along with enforcement aspects in Section 4OA2.2 of this report. This URI is closed.

4OA6 Management Meetings

.1 Exit Meeting Summary

On March 4, 2010, the lead inspector presented the Emergency Preparedness inspection results to Mr. C. Burton, and other members of your staff. The inspector confirmed that proprietary information was not provided or reviewed during the inspection.

On April 26, 2010, the inspector presented the inspection results to Mr. C. Burton, and other members of the licensee staff. The inspectors confirmed that proprietary information was not provided or examined during the inspection period.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

B. Bernard, Superintendent, Security
C. Burton, Vice President Harris Plant
D. Corlett, Supervisor, Licensing/Regulatory Programs
J. Dills, Manager, Operations
K. Harshaw, Manager, Outage and Scheduling
K. Henderson, Plant General Manager
G. Kilpatrick, Training Manager
S. O'Connor, Manager, Engineering
M. Parker, Superintendent, Radiation Protection
H. Curry, Manager, Nuclear Oversight
J. Robinson, Superintendent, Environmental and Chemistry
J. Warner, Manager, Support Services

NRC personnel

R. Musser, Chief, Reactor Projects Branch 4, Division of Reactor Projects, Region II

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000400/2010002-01	NCV	Failure to Submit a Licensee Event Report for a Condition Prohibited by Technical Specifications Associated with the “B” Emergency Service Water Discharge Valve (Section 4OA2.2)
05000400/2010002-02	NCV	Failure to Promptly Evaluate Operating Experience and Identify Potential Steam Voiding as a Condition Adverse to Quality (Section 4OA2.3)
05000400/2010002-03	NCV	“A” ESW Pump Power Supply Cables Submerged in Water (Section 1R06)

Opened

05000400/2010-002	LER	Manual Actuation of the Reactor Protection System due to Hydrogen Seal Oil Leak

Closed

05000400/2010-001	LER	Clearance Error Results in Equipment Becoming Inoperable (Section 4OA3.1)
05000400/2009-002	LER	Potential for RHR Trains to Be Inoperable During Mode Changes (Section 4OA3.2)
05000400/2009003-01	URI	Review the Significance of the Cooling Tower Blowdown Line Pathway Dose Compared to Doses from all other Pathways (Section 4OA5.2)
05000400/2009005-02	URI	“A” ESW Pump Power Supply Cables Submerged in Water (Section1R06)
05000400/2009006-03	URI	Unresolved Item Associated with the Evaluation of the Failure of Emergency Service Water Valve 271 (Section 4OA5.3)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Procedures

- ORT-1415, Electric Unit Heater Check Monthly Interval
- OP-161.01, Operations Freeze Protection and Temperature Maintenance Systems
- AP-300, Severe Weather
- AP-301, Seasonal Weather Preparations and Monitoring

Section 1R04: Equipment Alignment

Partial System Walkdown

Instrument Air system:
- Procedure OP- 151.01, Compressed Air System,
- Drawing 2165-S-0800, Simplified Flow Diagram Service Air System
- Drawing 2165-S-0801, Simplified Flow Diagram Instrument Air Systems

Electrical Switchyard system:
- Procedure OP- OP-156.02, AC Electrical Distribution System,
- Drawing 5165-B-C-0001, AC Electrical Distribution System
- FSAR 8.3.1 Onsite Power

“B” DC Electrical Distribution system:
- Procedure OP- 156.01, DC Electrical Distribution System,
- Stationary Battery Guide: Design, Application, and Maintenance

Section 1R05: Fire Protection

- FPP-001 Fire Protection Program Manual
- FPP-004, Transient Combustible Control
- FPP-013, Fire Protection – Minimum Requirements, Mitigating Actions and Surveillance Requirements
- FPP-012-05-DFOSB, Diesel Fuel Oil Storage Building Fire Pre-Plan, O01, Fuel Oil Transfer Pump Room “A”
- FPP-012-05-DFOSB, Diesel Fuel Oil Storage Building Fire Pre-Plan, O02, Fuel Oil Transfer Pump Room “B”
- FPP-012-05-DFOSB, Diesel Fuel Oil Storage Building Fire Pre-Plan, O05, Mezzanine Above Pump Rooms
- FPP-012-05-DFOSB, Diesel Fuel Oil Storage Building Fire Pre-Plan, O08, Balance
- FPP-012-05-DFOSB, Diesel Fuel Oil Storage Building Fire Pre-Plan, O09, Yard
- FPP-012-02-RAB286, Reactor Auxiliary Building Elevation 286 Fire Pre-Plan, A40, Cable Spreading Room “A”
- FPP-012-02-RAB286, Reactor Auxiliary Building Elevation 286 Fire Pre-Plan, A42, Cable Spreading Room “B”

- FPP-012-02-RAB286, Reactor Auxiliary Building Elevation 286 Fire Pre-Plan, A34, Switchgear Room "A"
- FPP-012-02-RAB286, Reactor Auxiliary Building Elevation 286 Fire Pre-Plan, A35, Switchgear Room "B"
- FPP-012-02-RAB286, Reactor Auxiliary Building Elevation 286 Fire Pre-Plan, A36, Battery Room "A"-SA
- FPP-012-02-RAB286, Reactor Auxiliary Building Elevation 286 Fire Pre-Plan, A37, Non-Safety Battery Room
- FPP-012-02-RAB286, Reactor Auxiliary Building Elevation 286 Fire Pre-Plan, A38, Battery Room "B"-SB
- FPP-012-02-RAB286, Reactor Auxiliary Building Elevation 286 Fire Pre-Plan, A44, Auxiliary Control Room
- FPP-012-02-RAB305-324, Reactor Auxiliary Building Elevations 305 and 324 Fire Pre-Plan, A57, Reactor Auxiliary Building Exhaust Fan Area
- FPT-3580, Fire Wrap Inspection - Interam 18-Month Interval Modes: All
- Drawing 2166-S-2600, Three-Hour Fire Barrier System 3M-Interam Application Guide And Installation Details, Sheets 1-56
- EC 48802, Make Auxiliary Control Panel Room Separate Fire Area
- Fire Drill Scenario, "B" Startup Transformer Fire, Revision 3

<u>Section 1R12: Maintenance Effectiveness</u>

- NUMARC 93-01, Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants
- ADM-NGGC-0101, Maintenance Rule Program

<u>Section 1R13: Maintenance Risk Assessments and Emergent Work Evaluation</u>

- OMP-003, Outage Shutdown Risk Management
- WCM-001, On-line Maintenance
- ADM-NGGC-0006, Online Equipment Out of Service (EOOS) Models for Risk Assessment
- Calculation HNP-F/PSA-0011, Online Equipment Out of Service Probability Safety Analysis Model

<u>Section 1R15: Operability Evaluations</u>

- OPS-NGGC-1305, Operability Determinations
- Vendor Technical Manual – Boric Acid Transfer Pump

<u>Section 1R19: Post Maintenance Testing</u>

- Drawing SA-A081, "B" MSIV Pneumatic and Hydraulic Control Circuit Schematic
- Smart Maintenance Report, "B" MSIV
- Vendor Manual, VM-MEE, Actuators, Section 4.0.0.0, MSIV Principles of Operation
- Westinghouse Specification-G-678842, 3.3.4, MSIV Control System
- MSIV Testing Plan, November 18, 2009
- Drawing 1364-94593, Air Valve Assembly Electro/Pneumatic Air Control Circuit

- Drawing 2166-B-401, Sheets 1003 and 1004, Control Wiring Diagram, “B” MSIV Trains “A” and “B”
- ISI-801, In-Service Testing of Valves
- Valve Test Deviation Record, 1CH-279, March 26, 2010
- OPS-NGGC-1303, Independent Verification
- PM-E0044, 480 VAC Siemens Type RLN(F) Load Center Breaker and Cubicle Preventative Maintenance

Section 1R22: Surveillance Testing

- Part 9900: Technical Guidance; Maintenance - Preconditioning of Structures, Systems, and Components Before Determining Operability
- FSAR #10.3 Main Steam Supply System
- Drawing 2165-S-0542, Simplified Flow Diagram Main Steam Supply System
- ISI-801, In-Service Testing of Valves
- ISI-800, In-Service Testing of Pumps
- OST-1000, Power Range Heat Balance, ERFIS On-Line Calculation, Daily Interval, Mode 1 (Above 15% Power)

Section 1EP2: Alert and Notification System Evaluation

Procedures

- EPM-400, Public Notification and Alerting System, Rev. 13
- EPM-500, Public Education and Information Program, Rev. 1
- WPS-2900 Series High Power Voice and Siren System Operating and Troubleshooting Manual

Records and Data

- 2009 Annual Tone Alert Radio Test Survey
- 2008 Annual Tone Alert Radio Test Survey

Section 1EP3: Emergency Preparedness Organization Staffing and Augmentation System

Procedures

- PEP-230, Control Room Operations, Rev. 17
- PEP-240, Activation and Operation of the Technical Support Center, Rev. 12
- PEP-260, Activation and Operation of the Operations Support Center, Rev. 12
- PEP-270, Activation and Operation of the Emergency Operations Facility, Rev, 21
- PEP-310, Notifications and Communications, Rev. 24
- PEP-350, Protective Actions, Rev. 7
- EPM-200, ERO Training Program, Rev. 10
- EPM-201, EP Staff Training Program, Rev. 6
- EMP-602, Routine Maintenance and Testing of the Dialogic System, Rev. 1
- EPL-001, Emergency Phone List, Rev. 70
- Pager Call Out Codes, 08/03

- ERO Expectations Memo, 07/1/2009

Records and Data

- 2009 augmentation drill; 09/23/2009
- ERO Training Modules 1 and 2

Section 1EP4: Emergency Action Level and Emergency Plan Changes

Change Packages

- PLP-201, Emergency Plan, Rev. 53 and 54
- EPM-100, EP Program Administration, Rev. 8
- EPM-210, EP Drill and Exercise Program, Rev. 15
- EPM-400, Public Notification and Alerting System, Rev. 11
- EPM-110, Emergency Classification and Protective Action Recommendations, Rev. 17

Section 1EP5: Correction of emergency Preparedness Weaknesses

Procedures

- CAP-NGGC-0200, Corrective Action Program, Rev. 30
- CAP-NGGC-0201, Self-Assessment/Benchmark Programs, Rev. 13
- PI-AA-204, Condition Identification and Screening Process, Rev. 5

Corrective Actions – Condition Report (CR)

- AR #118913, removing striking personnel from facility entrance
- AR #247133, tracking and trending of supplemental dose projection training
- AR #247140, emergency facility and equipment standards
- AR #292413, NRC KPI data input not verified
- AR #307766, late PAR
- AR #323495, EP Drill Follow-up ENF wind direction did not match PARs
- AR #322922, Sample analysis guidance needed for the different chemistry samples needed
- AR #323082, simulator data for WRGM not tracking with scenario timeline caused confusion
- AR #323099, SEC and security director pursued 50.54(x) evaluate security plan, cancelled see NRC 323236
- AR #323101, some cases of controllers not meeting expectations during interactions with players
- AR #323207, post accident sampling capability should be verified
- AR #323219, need guidance for issuing KI to security in OSC
- AR #323221, 3 of 4 portable gas generators would not start used by HP field teams
- AR #323223, Scenario items are becoming to predictable
- AR #323236, improved opportunity for RIS 2008-26
- AR #323240, dose assessments were not formally approved and included on the ENFs

- AR #323440, EPM-400, Public Notification and Alerting System needs revision to ensure maximum of two attempts allowed to be considered successful
- AR #348835, incorrect emergency notification form
- AR #348991, OSC relocation drill issues
- AR #349003, high rad in post accident sample area
- AR #357883, unannounced drill attendance
- AR #368307, missed classification in licensed operator continuing training

Records and Data

- H-EP-08-01, Harris Nuclear Plant Emergency Preparedness Assessment, 09/26/08
- H-EP-09-01, Harris NOS Emergency Preparedness Mid-Cycle Review, 08/10/09 – 08/19/09
- Assessment 343816, Quick Hit Self Assessment, 06/15/09 – 07/15/09
- Assessment 339544, Quick Hit Self Assessment, 06/01/09 – 07/14/09
- Assessment 314106, Self Assessment, 12/17/08 – 01/08/09
- Assessment 310262, Quick Hit Self Assessment, 03/03/09 and 08/0/09
- Assessment 308955, Self Assessment, 11/17/08 – 12/08/08
- 11/02/2009 Drill Package
- 09/23/2009 Drill Package
- 08/04/2009 Drill Package
- 03/03/2009 Drill Package

Section 4OA1: Performance Indicator Verification

- NEI 99-02, Regulatory Assessment Performance Indicator Guideline

Procedures

- EPM-100, EP Program Administration, Rev. 8
- REG-NGGC-0009, NRC Performance Indicators and Monthly Operating Report Data Rev.9
- EP-EAL, Emergency Action Level Guidelines, Rev. 7

Records and Data

- DEP opportunities documentation for 1st, 2nd, 3rd, and 4th Quarters 2009
- Drill and exercise participation records of ERO personnel 1st, 2nd, 3rd, and 4th Quarters 2009
- Siren test data 1st, 2nd, 3rd, and 4th Quarters 2009
- Various ERO Personnel Qualification and Participation records

Section 4OA2: Identification and Resolution of Problems

- CAP-NGGC-0200, Corrective Action Program
- CAP-NGGC-0202, Operating Experience Program

Section 4OA5: Other Activities

- Radiological Environmental Operating Report, 2008
- Annual Radioactive Effluent Release Report, 2008
- Shearon Harris Offsite Dose Calculation Manual, Rev. 20
- Impact of Cooling Tower Blowdown Line Leak upon a member of the Public, 1/20/10
- Cooling Tower Blowdown Line Assessment Report, April 2009
- AR 00328551, Leak in Cooling Tower Blowdown Line, 4/1/09