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

January 28, 2011

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

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT - NRC INTEGRATED INSPECTION REPORT 05000400/2010005

Dear Mr. Burton:

On December 30, 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 January 20, 2011, with Mr. John Dufner 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.

The report documents two NRC-identified findings and four self-revealing findings of very low safety significance (Green). These findings were determined to involve a violation of NRC requirements. Additionally, one licensee-identified violation which was determined to be of very low safety significance is listed in this report. However, because of their very low safety significance and because they were entered into your corrective action program (CAP), the NRC is treating these findings as non-cited violations (NCV) consistent with 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 Senior Resident Inspector at the Shearon Harris facility. In addition, if you disagree with the cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region II, and the NRC Senior Resident Inspector at the Shearon Harris facility.

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 No.: 50-400
License No.: NPF-63

Enclosure: NRC Inspection Report 05000400/2010005
w/Attachment: Supplemental Information

cc w/encl: (See page 3)

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Letter to Christopher L. Burton from Randall A. Musser dated January 28, 2011

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT - NRC INTEGRATED INSPECTION REPORT 05000400/2010005

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No.:	50-400
License No.:	NPF-63
Report No.:	05000400/2010005
Licensee:	Carolina Power and Light Company
Facility:	Shearon Harris Nuclear Power Plant, Unit 1
Location:	5413 Shearon Harris Road New Hill, NC 27562
Dates:	October 1, 2010, through December 31, 2010
Inspectors:	J. Austin, Senior Resident Inspector P. Lessard, Resident Inspector M. Bates, Senior Operations Engineer (Section 1R11) J. Rivera-Ortiz, Senior Reactor Inspector (Section 1R08) M. Coursey, Reactor Inspector (Sections 1R08, 4OA5) R. Hamilton, Senior Health Physicist (Sections 2RS2, 4OA1, 4OA5) W. Loo, Senior Healthy Physicist (Section 2RS1) A. Nielsen, Senior Health Physicist (Section 2RS8) R. Kellner, Health Physicist (Section 2RS2) S. Walker, Senior Reactor Inspector (Section 1R17) J. Eargle, Reactor Inspector (Section1R17) N. Childs, Resident Inspector (Section 1R17) D. Mas-Penaranda, Reactor Inspector (Section 1R17)
Approved by:	Randall A. Musser, Chief Reactor Projects Branch 4 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000400/2010005; October 1, 2010 – December 31, 2010; Shearon Harris Nuclear Power Plant, Unit 1; Equipment Alignment, Maintenance Effectiveness, Plant Modifications, Post-Maintenance Testing, Surveillance Testing.

The report covers a three month period of inspection by resident inspectors, senior health physicists, health physicist, senior operations engineer, senior reactor inspector, reactor inspector, and announced baseline inspection by regional inspectors. Two NRC-identified and four self-revealing 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 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: Mitigating Systems

- Green. A self-revealing Green NCV of Technical Specifications (TS) 6.8.1, Procedures, was identified for the licensee's failure to follow procedure MST-I0073, Train "B" 18 Month Manual Reactor Trip, Solid State Protection System Actuation Logic & Master Relay Test. Specifically, step 7.4.14 of MST-I0073 required the licensee to place the Master Relay Selector Switch (MRSS) in the "Off" position. Contrary to this requirement on October 28, 2010, the licensee failed to place the MRSS in the "Off" position at step 7.4.14. Instead, at step 7.5.85, the technicians noticed that the MRSS remained in Position "3" and then placed the MRSS in the "Off" position. This action combined with the current plant condition caused an invalid "B" train safety injection signal (SIS) and "B" Emergency Safeguards Sequencer (ESS) actuation while the plant was in Mode 6. The licensee entered this issue into their corrective action program (CAP) as action request (AR) #430289. As corrective action, the licensee restored the plant to the pre-actuation condition and conducted training for the maintenance technicians.

  The failure to follow procedure MST-I0073 for the proper operation of the MRSS was a performance deficiency. The finding was more than minor because it is similar to the more than minor example 4.b from MC 0612 Appendix E in that an operator incorrectly operated a switch causing a plant transient. Additionally, it is associated with the human performance attribute of the Mitigating Systems cornerstone, and it affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, it resulted in an invalid SIS causing the ESS to start the "B" ESW and "B" CCW pumps. Using IMC 0609, Significance Determination Process, Phase 1 screening worksheet and Appendix G (Shutdown Operations), Attachment 1, Checklist 4, this finding was determined to be of very low safety significance because it did not meet any of the guidelines which require quantitative assessment. The finding has a cross-cutting aspect of Human Error Prevention, as described in the Work Practices component of the Human Performance cross-cutting area because the technicians proceeded in the face of uncertainty without consulting supervision when they encountered unexpected plant conditions (H.4(a)). (Section 1R12)

- Green. The inspectors identified a Green NCV of TS 3.1.2.6, Borated Water Sources, for the failure to comply with the limiting conditions for operation, while the Refueling Water Storage Tank (RWST) was aligned to the non-seismic Fuel Pool Purification system (FPPS) for purification, causing the RWST to be inoperable. Specifically, when FPPS was aligned to the RWST, the licensee did not declare the RWST inoperable. The licensee took corrective actions (AR #422180) and revised OP-116.1, FPPS, to remove the capability to purify the RWST in Modes 1 through 4.

  The failure to comply with the actions of TS Limiting Condition for Operation (LCO) 3.1.2.6 while the Refueling Water Storage Tank (RWST) was aligned to the non-seismic FPPS for purification on May 24, 2010, causing the RWST to be inoperable, was a performance deficiency. The performance deficiency was more than minor because it affected the Design Control attribute of the Mitigating System cornerstone objective of ensuring the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage). Specifically, when the FPPS was aligned to the RWST, the licensee did not declare the RWST inoperable. The inspectors evaluated the significance of this finding Using Attachment 4 of IMC 0609, the significance of this finding was determined to be of very low safety significance (Green) because it was a design or qualification deficiency confirmed not to result in loss of operability or functionality, did not represent a loss of system safety function, did not represent actual loss of safety function of a single train for longer than its TS Allowed Outage Time, did not represent an actual loss of safety function of one or more non-TS Trains of equipment designated as risk-significant, and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The finding had a cross-cutting aspect of Conservative Assumptions, as described in the Decision Making component of the Human Performance cross-cutting area because, assumptions used in the justification to support the procedure change (i.e. a license amendment was not deemed required to support the procedure change) to OP-116.01 were non-conservative and the review of the issue in May 2010 did not adequately validate the assumptions (H.1(b)). (Section 1R18)

- Green. A self-revealing Green NCV of TS 6.8.1, Procedures, was identified for the licensee's failure to develop an adequate procedure for the post maintenance test of the recently replaced main generator lockout relay (MGLR). Specifically, the licensee failed to ensure that the post maintenance testing (PMT) was within the clearance boundary that was established for the MGLR replacement. This resulted in the inadvertent deenergization of the "B" Safety Bus and the "B" Residual Heat Removal (RHR) pump, which was the only pump providing decay heat removal (DHR). As corrective action, the licensee entered AOP-25, Loss of One Emergency

AC Bus, and restored DHR with the “B” RHR pump after approximately three minutes. The resultant increase in Reactor Coolant System temperature was approximately one degree. Additionally, the licensee plans to revise PLP-400, Post Maintenance Testing, to provide the work planner with additional guidance in the development of PMT for protective relays. The licensee entered this issue into their CAP as AR #431732.

The licensee’s failure to develop an adequate procedure for the post maintenance test of the recently replaced MGLR was a performance deficiency. The performance deficiency was more than minor because it is associated with the procedure quality attribute of the Mitigating Systems cornerstone, and it affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, it resulted in the inadvertent deenergization of the “B” Safety Bus and loss of DHR. Using IMC 0609, “Significance Determination Process,” Phase 1 screening worksheet of the SDP, the inspectors determined that the use of Appendix G, Shutdown Operations Significance Determination Process, was necessary. Using Checklist 3 of Attachment 1 of Appendix G, the inspectors determined that this issue affected both the DHR equipment guidelines and the emergency electrical bus guidelines and therefore required a Phase 2 analysis. Using Worksheet 8 of Attachment 2 of Appendix G, the inspectors determined that recovery credit was appropriate because 1) sufficient time was available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training was conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use. Using a time to boil of greater than one hour and the fact that the steam generators were not available for cooling, the result of the Phase 2 was that a Phase 3 was necessary. A regional Senior Reactor Analyst evaluated the performance deficiency under the Phase 3 protocol of the Significance Determination Process. Based upon the results of that evaluation, the performance deficiency was characterized as of very low safety significance (Green). The finding has a cross-cutting aspect of Work Coordination, as described in the Work Control component of the Human Performance cross-cutting area because the licensee did not understand the potential operational impact of the work activities or adequately account for current plant conditions (H.3(b)). (Section 1R19).

- Green. A self-revealing Green NCV of TS 6.8.1, Procedures, was identified for the licensee’s failure to correctly implement Section D.2.10 of Engineering Change (EC) #74866R1 when aligning the Mechanism Operated Cell (MOC) switch for the “A” Main Feed Water Pump (MFP) breaker 1A-6. Specifically, the misalignment of the MOC resulted in the inadvertent auto actuation of the “B” Motor Driven Auxiliary Feed Water (MDAFW) pump. As corrective action (AR #432568), the licensee realigned MOC switch contacts under task 3 of Work Order (WO) #01658137 per the instructions of EC #74866R1. Post Modification testing verified contact continuity in both the breaker open and closed and was completed satisfactory.
  The failure to follow Section D.2.10 of EC #74866R1 on WO #01658137 task 1 was a performance deficiency. The performance deficiency was more than minor

because it is associated with the human performance attribute of the Mitigating System cornerstone, and it affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the misalignment of the MOC resulted in the inadvertent automatic start of the “B” MDAFW pump. Using IMC 0609, “Significance Determination Process,” Phase 1 screening worksheet of the SDP, this finding was determined to be very low safety significance because it was not a design or qualification deficiency confirmed to result in a loss of operability or functionality, did not represent a loss of system safety function, did not result in a loss of safety system function for a single train for greater than TS allowed outage time, did not result in a loss of safety function of one or more non-TS trains of equipment designated as risk significant for greater than 24 hours, and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The finding has a cross-cutting aspect of Human Error Prevention, as described in the Work Practices component of the Human Performance cross-cutting area because the licensee did not apply sufficient human error prevention tools to ensure the correct alignment of the MOC switch contacts associated with vacuum circuit breaker 1A-6 (H.4(a)). (Section 1R19)

- Green. A self-revealing Green NCV of Technical Specification (TS) 6.8.1, Procedures, was identified for the licensee’s failure to establish and implement procedural requirements that would ensure the Program “C” relay wiring configuration in the “A” Sequencer remained in accordance with plant drawings following maintenance. Procedure OPS-NGGC-1303, Independent Verification, did not require the use of plant drawings to verify the “As Built” configuration when lifting and landing leads, which ultimately led to the deenergization of the “A” 6.9kV Safety bus during a surveillance test. The licensee took corrective action (AR #424668) and replaced the 86UV/SA relay, tested components within the circuit that could be affected, corrected the wiring issue and issued a memo to set expectations for utilizing plant design drawings when lifting/landing leads.

  The failure to establish and properly implement procedural guidance to maintain the Program “C” relay in the “A” Sequencer wired in accordance with plant drawings following maintenance on April 28, 2009, was a performance deficiency. The performance deficiency was more than minor because it affected the procedure quality attribute of the Mitigating System cornerstone objective of ensuring the availability and reliability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage). Specifically, the leads being incorrectly landed would have prevented the “A” EDG from automatically re-energizing the “A” 6.9kV Bus. Using IMC 0609, “Significance Determination Process,” Phase 1 Worksheet, the inspectors concluded that a Phase 2 evaluation was required because this finding represented a loss of safety function of the “A” 6.9kV safety bus. 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, 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 Green. The finding has a cross-cutting aspect of Documentation and Component Labeling, as described in the Resources component of the Human Performance cross-cutting area because the licensee did not effectively communicate expectations regarding the utilization of design drawings to aid in the proper completion of the verification sign-off form (OPS-NGGC-1303) (H.2(c)). (Section 1R22)

Cornerstone: Barrier Integrity

- Green. The inspectors identified a Green NCV of Technical Specification (TS) 6.8.1, Procedures, for the licensee's failure to properly implement procedural guidance to maintain the Fuel Handling Building Emergency Exhaust System (FHBEES) boundary. Specifically, the licensee failed to properly implement procedural guidance to maintain the FHBEES boundary while two doors were propped open on October 21, 2010 and October 22, 2010. This was apparent when the inspectors identified one individual unaware of their responsibilities and another individual inattentive. The licensee entered this issue into their CAP as action request (AR) #428580 and AR #428858. The licensee took corrective action to relieve the inattentive individual and conducted additional training for all of the other individuals responsible for closing the doors.

  The failure to properly implement procedural guidance to maintain the FHBEES boundary while two doors were propped open from October 21, 2010 until October 22, 2010 was a performance deficiency. The performance deficiency was more than minor because it was associated with the Barrier Performance attribute of the Barrier Integrity cornerstone and affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. The potential safety consequence is that if spent fuel had been damaged in the spent fuel pool during this time, the FHBEES may not have been able to properly filter and monitor a radioactive release. Using IMC 0609, "Significance Determination Process," Phase 1 Worksheet, the inspectors determined this issue to be of very low safety significance because it only represented a degradation of the radiological barrier function provided for the fuel handling building. The finding has a cross-cutting aspect of Training and Work Hours, as described in the Resources component of the Human Performance cross-cutting area because the licensee did not effectively train the individuals regarding their procedural responsibilities when the FHBEES doors were propped open (H.2(b)). (Section 1R04)

B. Licensee-Identified Violations

A violation of very low safety significance which was identified by the licensee was reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's CAP. That violation and corrective action tracking number are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Unit 1 entered this reporting period at approximately 93 percent power and was preparing to enter a refueling outage (RFO) 16. On October 2, 2010, the unit was shut down and commenced RFO-16. The unit entered mode 1 on November 11, 2010, achieved rated thermal power (RTP) on November 19, 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

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 their CAP in accordance with station corrective action 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:

- Refueling Water Storage Tank (RWST) piping and instrumentation
- Emergency Service Water System

b. Findings

No findings were identified.

.2 External Flooding

a. Inspection Scope

The inspectors evaluated the design, material condition, and procedures for coping with the design basis probable maximum flood. The evaluation included a review to check for deviations from the descriptions provided in the UFSAR for features intended to mitigate the potential for flooding from external factors. As part of this evaluation, the inspectors checked for obstructions that could prevent draining to verify that barriers required to mitigate the flood were in place and operable. Additionally, the inspectors performed a walkdown of the protected area to identify any modification to the site which would inhibit site drainage during a probable maximum precipitation event or allow water ingress past a barrier. The inspectors also reviewed the abnormal operating procedure (AOP) for mitigating the design basis flood to ensure it could be implemented as written. Additionally, the inspectors selected the new Dedicated Shutdown Diesel Generator as a plant area for a focused review. Documents reviewed are listed in the attachment.

b. Findings

No findings 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:

- The “A” Spent Fuel Pool Cooling while it was protected due to the “B” Spent Fuel Pool Cooling being down for maintenance on October 20, 2010;
- The Fuel Handling Building Emergency Exhaust System (FHBEES) while a compensatory action was in place to maintain the boundary on October 21 and 22, 2010;
- “A” Emergency Diesel Generator (EDG) while it was protected due to the “B” EDG out of service for maintenance on October 28, 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 WOs, 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.

b. Findings

Introduction: The inspectors identified a Green NCV of TS 6.8.1, Procedures, for the licensee's failure to properly implement procedural guidance to maintain the FHBEES boundary while two doors were propped open from October 21, 2010 until October 22, 2010.

Description: Between October 21, 2010 and October 22, 2010, the licensee had propped open doors 893 and 894 to allow for easier passage into the Fuel Handling Building. These doors serve as a part of the FHBEES boundary, and are allowed to be propped open by licensee procedure ADM-NGGC-0116, Nuclear Planning. When these doors are propped open, ADM-NGGC-0116 directs the licensee to station an individual to close the doors when notified by the main control room (MCR) in order to maintain an adequate FHBEES boundary. The responsible individual (RI) is also required to have a means of continuous communication with the MCR and be able to rapidly close the doors. The means of continuous communication used is a hand held radio located with the RI and another in the MCR.

During a walkdown of the FHBEES system on October 21, 2010, the inspector interviewed the RI to determine if that person was capable of meeting the requirements of ADM-NGGC-0116. During the interview, it was determined that the RI was unaware of their responsibility to rapidly close the doors when directed by the MCR. The RI was also unaware of the need for the handheld radio. On October 22, 2010, the inspectors performed a follow-up walkdown of the system. During this walkdown, the inspectors found the RI inattentive. A member of the licensee's staff who was accompanying the inspectors during the walkdown returned the individual to an attentive status and, shortly thereafter, had a new RI stationed. Additionally, the licensee conducted training for all of the other individuals responsible for closing the doors.

Analysis: The failure to properly implement procedural guidance to maintain the FHBEES boundary while two doors were propped open from October 21, 2010 to October 22, 2010 was identified as a performance deficiency. The performance deficiency was more than minor because it was associated with the Barrier Performance attribute of the Barrier Integrity cornerstone and affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Using IMC 0609, "Significance Determination Process," Phase 1 Worksheet, the inspectors determined this issue to be of very low safety significance because it only represented a degradation of the radiological barrier function provided for the fuel handling building. The finding has a cross-cutting aspect of Training and Work Hours, as described in the Resources component of the Human Performance cross-cutting area because the licensee did not effectively train the individuals regarding their procedural responsibilities when the FHBEES doors were propped open (H.2(b)).

Enforcement: TS 6.8.1, Procedures, requires that written procedures shall be established, implemented, and maintained, covering applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978. Section 3 of Appendix A of Regulatory Guide 1.33, Revision 2, February 1978 states that there will be procedures for operating Atmosphere Cleanup Systems, such as the FHBEES. Licensee procedure ADM-NGGC-0116 governs the use of an RI with a means of continuous communication to rapidly close the FHBEES boundary doors when directed by the MCR. Contrary to this requirement, the RI was found to be unaware of the required responsibilities on October 21, 2010 and inattentive on October 22, 2010, respectively. The potential safety consequence is that if spent fuel had been damaged in the spent fuel pool on these days when the doors were propped open, the FHBEES may not have been able to properly filter and monitor a radioactive release. The licensee took corrective action to relieve the inattentive individual and conducted additional training for all of the other individuals responsible for closing the doors. Because the finding is of very low safety significance and has been entered into the CAP as AR #428580 and AR #428858, and consistent with the NRC Enforcement Policy, this violation is being treated as a non-cited violation, and is designated as NCV 05000400/2010005-01, “Failure to Properly Implement Procedural Guidance to Maintain the FHBEES Boundary.”

.2 Semi-Annual Complete System Walkdown

a. Inspection Scope

On November 2, 2010 the inspectors performed a complete system alignment inspection of the Emergency Service Water System to verify the functional capability of the system. This system was selected because it was considered risk significant in the licensee’s probabilistic risk assessment. The inspectors walked down the system to review mechanical and electrical equipment line ups, electrical power availability, system pressure and temperature indications, component labeling, component lubrication, component and equipment cooling, hangers and supports, operability of support systems, and to ensure that auxiliary equipment or debris did not interfere with equipment operation. A review of a sample of past and outstanding WOs was performed to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the CAP database to ensure that system equipment alignment problems were being identified and appropriately resolved. The documents used for the walkdown and issue review 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 #431905, Loose Bolts Found on Air Handler-4 Hanger;

- AR #430613, “B” ESW Pump Start without Screen Wash.

b. Findings

No findings were identified.

1R05 Fire Protection

.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:

- Fuel Handling Building (FHB), 286’ Elevation
- FHB, 261’ Elevation
- FHB, 236’ Elevation
- Reactor Auxiliary Building (RAB), 236' Elevation, Mechanical Penetration Area
- RAB, 236’ Elevation, Chemical and Volume Control System and Boron Thermal Regeneration System Heat Exchanger Area

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 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.

1R06 Flood Protection Measures

.1 Review of Areas Susceptible to Internal Flooding

a. Inspection Scope

The inspectors reviewed selected risk important plant design features and licensee procedures intended to protect the plant and its safety related equipment from internal flooding events. The inspectors reviewed flood analyses and design documents, including the UFSAR, engineering calculations, and abnormal operating procedures (AOPs), for licensee commitments. In addition, the inspectors reviewed licensee drawings to identify areas and equipment that may be affected by internal flooding caused by the failure or misalignment of nearby sources of water, such as the fire suppression or the circulating water systems. The inspectors also reviewed the licensee's corrective action documents with respect to past flood-related items identified in the CAP to verify the adequacy of the corrective actions. The inspectors performed a walkdown of the following plant area to assess the adequacy of watertight doors and verify drains and sumps were clear of debris and were operable, and that the licensee complied with its commitments:

- PRA-F-E-0004, Reactor Auxiliary Building 216' Flooding Area

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

- AR #439768, Reactor Auxiliary Building 216' Flooding Calculation is Incomplete

b. Findings

No findings were identified.

1R07 Heat Sink Performance

a. Inspection Scope

The inspectors reviewed the licensee's testing of the "B" Emergency Services Chilled Water heat exchangers 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 verified that test acceptance criteria considered differences between test conditions, design conditions, and testing criteria.

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

- AR #357595, Low Oil and Refrigerant in “B” Chiller;
- AR #368053, “A” Chiller Low Flow Alarm with Normal Service Water Flow;
- AR #382443, “B” Chiller Low Flow Alarm Lit for no Reason;
- AR #384074, “A” Chiller Chemical Addition not Performed;
- AR #404443, Unusual Sequence of Alarms Received on “B” Chiller when Swapping;
- AR #416570, “B” Chiller Refrigerant Pressure.

b. Findings

No findings were identified.

1R08 Inservice Inspection (ISI) Activities

.1 Non-Destructive Examination (NDE) Activities and Welding Activities

a. Inspection Scope

From October 11-22, 2010, the inspectors reviewed the implementation of the licensee’s In-service Inspection (ISI) program for monitoring degradation of the reactor coolant system (RCS) boundary and risk significant piping boundaries. The inspectors’ activities consisted of an on-site review of NDE and welding activities to evaluate compliance with the applicable edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section XI (Code of record: 2001 Edition with 2003 Addenda), and to verify that indications and defects (if present) were appropriately evaluated and dispositioned in accordance with the requirements of the ASME Code, Section XI acceptance standards.

The inspectors’ review of NDE activities specifically covered examination procedures, NDE reports, equipment and consumables certification records, personnel qualification records, and calibration reports (as applicable) for the following examinations:

- Ultrasonic Testing of Reactor Vessel Stud II-RV-001RVSTUD(20-38) 6” Diameter and 57 ⅜” length;
- Liquid Penetrant Testing of Integral Attachment of 11715-WMKS-0113A-1/14-RH-2/71H; and
- Ultrasonic Testing of Elbow to Nozzle for 11715 WMKS-0102C/16-WFPD-22/18A.

The inspectors’ review of welding activities specifically covered the welding activity listed below in order to evaluate compliance with procedures and the ASME Code. The inspectors reviewed the WO, repair and replacement plan, weld data sheets, welding procedures, procedure qualification records, welder qualification records, and NDE reports.

- WO #59102018991, Replace Valve 1-RC-105

b. Findings

No findings were identified.

.2 PWR Vessel Upper Head Penetration (VUHP) Inspection Activities

a. Inspection Scope

The licensee inspection for the Reactor Head Program during this outage was a visual examination conducted above the reactor pressure vessel upper head to identify potential boric acid leaks from pressure-retaining components. The inspectors specifically reviewed examination procedures, personnel training and qualification records, report VT-09-088 for the visual inspection of pressure-retaining components above the head performed during this outage, and reviewed the licensee's calculations for effective degradation years (EDYs) and reinspection years (RIYs). No reactor vessel augmented examination required by 10 CFR 50.55a(g)(6)(ii)(d) was required to be performed during this outage.

b. Findings

No findings were identified.

.3 Boric Acid Corrosion Control (BACC) Inspection Activities

a. Inspection Scope

The inspectors reviewed the licensee's BACC program activities to ensure implementation with commitments made in response to NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," and applicable industry guidance documents. Specifically, the inspectors performed an on-site record review of procedures and the results of the licensee's containment walk-down inspections performed during the Unit 1 Fall 2010 outage. The inspectors also interviewed the BACC program owner and conducted a walkdown of the reactor building to evaluate compliance with licensee's BACC program requirements and verify that degraded or non-conforming conditions, such as boric acid leaks identified during the containment walkdown, were properly identified and corrected in accordance with the licensee's BACC and CAP.

The inspectors reviewed a sample of engineering evaluations completed during the last outage for evidence of boric acid found on systems containing borated water to verify that the minimum design code required section thickness had been maintained for the affected components.

The inspector selected the following evaluations for review:

- AR #424663, Brown Residue on 1SI-251 (Accumulator "B" Discharge Check Valve);

- AR #424599, Pipe Cap at 1SI-54 (High Head Safety Injection to Reactor Coolant System Cold Legs Test Connection Isolation Valve) has Active Leak;
- AR #424602, White and Light-Brown Residue Found on Pipe Cap Downstream of 1SI-133 (High Head Safety Injection to Reactor Coolant System Cold Leg “B” Test Connection Isolation Valve).

b. Findings

No findings were identified.

.4 Steam Generator (SG) Tube Inspection Activities

a. Inspection Scope

The inspectors reviewed the Unit 1 eddy current testing (ECT) examination activities in SGs “A”, “B”, and “C” and evaluated them against the licensee’s TS, commitments made to the NRC, ASME Section XI, and Nuclear Energy Institute (NEI) 97-06 (Steam Generator Program Guidelines). The inspectors conducted the following inspection activities:

- Reviewed ECT status reports on a daily basis and discussed the results with the licensee lead Level III analyst to ensure that all tubes with relevant indications were appropriately screened for in-situ pressure testing in accordance with the applicable industry standards. In particular, the inspectors assessed whether assumed NDE flaw sizing accuracy was consistent with data from the Electrical Power Research Institute (EPRI) examination technique specification sheets (ETSS) or other applicable performance demonstrations. None of the SG tubes examined met the criteria for in-situ pressure testing.

- Reviewed the last Condition Monitoring and Operational Assessment report in conjunction with the ECT status reports to assess the licensee prediction capability for maximum tube degradation.

- Reviewed the latest Degradation Assessment report to assess the scope of the inspection and verify it included potential areas of tube degradation identified in prior outage SG tube inspections, industry operating experience, and NRC generic communications. The inspectors also verified that appropriate inspection scope expansion criteria were planned based on inspection results of active and new degradation mechanisms. Based on the ECT examination results, no new degradation mechanisms were identified and no ECT scope expansion was required.

- Reviewed the licensee’s repair criteria and repair process to ensure they were consistent with plant TS and industry guidelines.

- Reviewed the primary to secondary leakage (e.g., SG tube leakage) history for the last operating cycle. The inspectors found that primary to secondary leakage was below three gallons per day, or the detection threshold, during the previous operating cycle.

- Reviewed documentation to ensure that data analysts, ECT probes, and equipment configurations were qualified to detect the existing and potential SG tube degradation mechanisms in accordance with the applicable industry standards.

- Reviewed a sample of site-specific Examination Technique Specification Sheets (ETSSs) to ensure that their qualification was consistent with Appendix H, “Performance Demonstration for Eddy Current Examination,” or Appendix I, “NDE System Measurement Uncertainties for Tube Integrity Assessments,” of EPRI Pressurized Water Reactor Steam Generator Examination Guidelines, Revision 7.

- Directly observed a sample of ECT data acquisition in SG “A” Hot Leg side (Rotating Pancake Coil and Bobbin probes), SG “A” Cold Leg side (Bobbin Probe), SG “B” Hot Leg side (Rotating Pancake Coil), and SG “B” Cold Leg side (Bobbin Coil).

- Reviewed ECT data with a qualified analyst for the following tubes: SG “A” (tubes R87C26, R88C33, and R49C72), SG “B” (tubes R3C20, R36C47, R26C133, R26C135, and R28C135) and SG C (tubes R63C62, R60C85, and R69C16).

- Reviewed licensee’s Foreign Object Search and Retrieval (FOSAR) activities on the secondary side of SG “B” (top of tube-sheet area) in response to ECT indications of potential loose parts.

b. Findings

No findings were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a review of ISI-related problems, including welding and BACC, which were identified by the licensee and entered into the CAP as ARs. The inspectors reviewed the ARs to confirm that the licensee had appropriately described the scope of the problems and had initiated corrective actions. The review also included the licensee’s consideration and assessment of operating experience events applicable to the plant. The inspectors performed this review to ensure compliance with 10CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” requirements. The corrective action documents reviewed by the inspectors are listed in the report attachment.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification/Training Program

.1 Quarterly Review

a. Inspection Scope

On November 24, 2010, the inspectors observed a crew of licensed operators in the plant's simulator during a licensed operator training scenario 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 scenario tested the operators' ability to respond to the loss of an emergency electrical bus, reactor trip and an oil fire in the turbine building. 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
- 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 were identified.

.2 Annual Review of Licensee Requalification Examination Results

a. Inspection Scope

On March 26, 2010, the licensee completed the comprehensive biennial requalification written examinations and annual requalification operating tests 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 written examinations, individual operating tests and the crew simulator operating tests. These results were compared to the thresholds established in Manual Chapter 609 Appendix I, Operator Requalification Human Performance Significance Determination Process.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

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. Documents reviewed are listed in the Attachment.

The inspectors evaluated degraded performance issues involving the following three risk significant components:

- AR #423460, “B” Component Cooling Water Pump has Lost Run Indication on Main Control Board and at Breaker;
- AR #430289, Unexpected Safety Injection Signal and Sequencer Actuation during Testing; and
- AR #434129, Rod Control Urgent Failure Alarm.

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
- 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 AR associated with this area to verify that the licensee identified and implemented appropriate corrective actions:

- AR #424399, Main Control/Auxiliary Control Board Relay 43T-22SA/1081, Needed Assistance Transferring back to Normal.

b. Findings

Introduction: A self-revealing Green NCV of TS 6.8.1, Procedures, was identified for the licensee’s failure to follow procedure MST-I0073, Train “B” 18 Month Manual Reactor Trip, Solid State Protection System Actuation Logic & Master Relay Test. Specifically, on October 28, 2010, the licensee rotated the Master Relay Selector Switch (MRSS) to the “Off” position during the wrong step in the procedure, causing an inadvertent “B” train safety injection signal (SIS) and “B” Emergency Safeguards Sequencer (ESS) actuation while the plant was in Mode 6.

Description: While in Mode 6 on October 28, 2010, licensee night shift personnel commenced MST-I0073 and completed through step 7.5.19. However, at step 7.4.14, the procedure required the licensee to place the MRSS in the “Off” position. For an indeterminate reason, this step was not completed and the MRSS remained in Position “3.” Due to the length of the procedure, the night shift personnel determined that the surveillance procedure would have to be turned over to the day shift personnel at step 7.5.19. Without discovering the error, the night shift personnel turned over the procedure to the day shift personnel.

The day shift personnel continued the procedure without issue until step 7.5.85. At this point, the technicians identified that they were receiving unexpected indications. Upon investigation, they determined that the cause of the unexpected indications was the MRSS being in the wrong position. Without consulting supervision and with no procedural guidance, the technicians rotated the MRSS to the “Off” position. This action resulted in an invalid “B” train SIS causing the “B” ESS to start the “B” ESW and “B” CCW pumps.

The Safety Injection System (SIS) block cards were installed at the time of this issue; however, the SIS actuation still occurred, because the actuation signal was developed within the logic downstream of the block cards. Due to the plant configuration, the “B” Residual Heat Removal Pump and “B” Charging Safety Injection Pump did not start and therefore no water flowed into the reactor coolant system as a result of the SIS actuation signal. Additionally, the reactor head was removed for refueling, the refueling cavity level was greater than 23 feet above the reactor vessel flange and decay heat removal was being provided by the other train and was uninterrupted.

Analysis: The failure to follow procedure MST-I0073 for the proper operation of the MRSS was a performance deficiency. The finding was more than minor because it is similar to the more than minor example 4.b listed in MC 0612 Appendix E in where an operator incorrectly operated a switch causing a plant transient. Additionally, it is associated with the human performance attribute of the Mitigating Systems cornerstone, and it affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, it resulted in an invalid SIS causing the ESS to start the “B” ESW and “B” CCW pumps. Using IMC 0609, “Significance Determination Process,” Phase 1 screening worksheet and Appendix G (Shutdown Operations), Attachment 1, Checklist 4 of the SDP this finding was determined to be of very low safety significance because it did not meet any of the guidelines which require quantitative assessment. The finding has a cross-cutting aspect of Human Error Prevention, as described in the Work Practices component of the Human Performance cross-cutting area because the technicians proceeded in the face of uncertainty without consulting supervision when they encountered unexpected plant conditions (H.4(a)).

Enforcement: TS 6.8.1, Procedures, requires that written procedures shall be established, implemented, and maintained, covering applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978.

Section 8 of Appendix A of Regulatory Guide 1.33, Revision 2, February 1978, states that there will be procedures for performing surveillance tests on safety related equipment. The plant procedure governing this testing was a maintenance surveillance test, MST-I0073. Step 7.4.14 of MST-I0073 required the licensee to place the MRSS in the "Off" position. Contrary to this requirement on October 28, 2010, the licensee failed to place the MRSS in the "Off" position at step 7.4.14. Instead, at step 7.5.85, the technicians noticed that the switch was in the "3" position and then placed the MRSS in the "Off" position. This action resulted in an invalid "B" train SIS actuation signal causing the "B" ESS to start the "B" ESW and "B" CCW pumps. As corrective action, the licensee restored the plant to the pre-actuation condition and conducted training for the maintenance technicians. Because the finding is of very low safety significance and has been entered into the CAP (AR #430289), and consistent with the NRC Enforcement Policy, this violation is being treated as a non-cited violation, and is designated as NCV 05000400/2010005-02, "Failure to Follow Procedure Results in Emergency Safeguards Sequencer Actuation and Safety Injection Signal (SIS) while the Plant was in Mode 6."

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 three 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:

- Plant risk evaluated for restoration of the non-essential CCW header being supplied by the "B" CCW Pump, following the loss of the "A" CCW Pump on October 3, 2010. Risk remained green;
- Plant Risk evaluated for trip of the "C" plant Air Compressor on November 12, 2010. The plant risk remained yellow for the plant startup (the quantitative analysis for the "C" compressor out-of-service was green);
- Plant risk evaluated for the replacement of a burned out Safe Shutdown Light above the Reactor Trip Breakers on November 29, 2010. Risk remained green.

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

a. Inspection Scope

The inspectors selected the following five 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 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. Documents reviewed are listed in the attachment.

- AR #426652, Reactor Trip Switchgear has Incorrect Bolting Hardware;
- AR #427960, Incorrect Lugs on Line and Load Side of Diesel Building Exhaust Fan Breaker;
- AR #428788, 1SW-271 (Emergency Service Water Return Isolation Valve) Breached Boundary Opened;
- AR #424906, E-88 "B" SB Breaker (Emergency Service Water Structure) Tripped;
- AR #430301, Possible Water Hammer Event on the "B" ESW System.

b. Findings

No findings were identified.

1R17 Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed selected samples of evaluations to confirm that the licensee had appropriately considered the conditions under which changes to the facility, Updated Final Safety Analysis Report (UFSAR), or procedures may be made, and tests conducted, without prior NRC approval. The inspectors reviewed evaluations for six changes and additional information, such as drawings, calculations, supporting analyses, the UFSAR, and TS to confirm that the licensee had appropriately concluded that the changes could be accomplished without obtaining a license amendment. The six evaluations reviewed are listed in the List of Documents Reviewed.

The inspectors reviewed samples of changes for which the licensee had determined that evaluations were not required, to confirm that the licensee's conclusions to "screen out" these changes were correct and consistent with 10CFR50.59. The 16 "screened out" changes reviewed are listed in the List of Documents Reviewed.

The inspectors evaluated engineering design change packages for 10 material, component, and design based modifications to evaluate the modifications for adverse effects on system availability, reliability, and functional capability. The 10 modifications and the affected cornerstones are as follows:

- EC #69252, ESW Pump Motor Alternate Replacement, Rev. 5 (Mitigating Systems)
- EC #74408, Removal of 1SW-272; Make Temp EC #73305 Permanent, Rev. 0 (Mitigating Systems)
- EC #58448, Pipe Replacement of ESW Lines to CSIP Coolers and Installation of Flow Instrumentation, Rev. 0 (Mitigating Systems)
- EC #64641, Modify Transfer Scheme and Power Supply Scheme for 1SW-1208, Rev. 7 (Mitigating Systems)
- EC #62649, Evaluated New Power Supply for DRPI & RPS, Rev. 1 (Mitigating Systems)
- EC #74866R1, MOC Switch Set-up for Installed 6.9kV Vacuum Breaker, Rev. 2 (Initiating Events)
  EC #66427, 6.9kV Breaker Replacement, Rev. 11(Initiating Events)
- EC #69501, Design and Installation of Incipient Fire Detection, Rev. 5 (Mitigating Systems)
- EC #62967, Relocate ESCW Expansion Tanks, Rev. 3 (Mitigating Systems)
- EC #73402, Turbine Driven AFW Pump Casing Vents, Rev. 0 (Mitigating Systems)

Documents reviewed included procedures, engineering calculations, modification design and implementation packages, WOs, site drawings, corrective action documents, applicable sections of the living UFSAR, supporting analyses, TS, and design basis information. The inspectors additionally reviewed test documentation to ensure adequacy in scope and conclusion. The inspectors review was also intended to verify that all appropriate details were incorporated in licensing and design basis documents and associated plant procedures.

The inspectors also reviewed selected corrective action documents associated with modifications and screening/evaluation issues to confirm that problems were identified at an appropriate threshold, were entered into the corrective action process, and appropriate corrective actions had been initiated and tracked to completion.

b. <u>Findings</u>

No findings were identified.

1R18 Plant Modifications

a. Inspection Scope

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

- EC #79004, Temporary Modification to Cap on the “A” Reactor Coolant Pump (RCP) Seal Injection Isolation Valve (1CS-347);
- EC #63236 Permanent Modification to Account for Risk Increase during Proposed RWST On-Line Cleanup.

These documents 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 temporary modification completed under EC #79004 placed a temporary cap on 1CS-347 after its internals had been removed. The 1CS-347 internals were removed after the licensee identified that the valve seat was damaged and would not isolate flow during a test. This valve is a normally open valve which supplies seal injection flow to the “A” RCP. The safety function of 1CS-347 is to maintain the RCS pressure boundary and there was no pressure boundary leakage. The temporary cap was installed to maintain the RCS pressure boundary after the internals had been removed, until the valve could be replaced.

The permanent modification completed under EC #63236 evaluated and provided justification to change OP-116.01 to allow manual action/compensatory measures cross connect the RWST to the FPPS for purification.

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

- AR #430564, Failed Leak Rate Test due to Suspected Leakage by 1CS-347;
- AR #430762, Valve 1CS-347 Disc Found Stuck In Body.

b. Findings

Introduction: The inspectors identified a Green NCV of TS 3.1.2.6, Borated Water Sources, for the failure to comply with the limiting conditions for operation, while the RWST was aligned to the non-seismic Fuel Pool Purification System (FPPS) for purification, causing the RWST to be inoperable.

Description: The FPPS is a non-safety, non-seismic system and is separated from the RWST by a normally closed safety related boundary valve. The RWST is seismically qualified and a safety-related system described in TS. The two systems can be mechanically cross connected via a valve CT-23, RWST to Spent Fuel Pool Pump Suction Valve. The CT-23 valve is a manually operated valve and has no automatic isolation signals.

In 2006, the licensee revised OP-116.01, Fuel Pool Purification System, to permit purification of the RWST in modes 1-4 without declaring the RWST inoperable and entering the TS LCO, with the intention of not losing critical path time due to water clarity issues. The change relied on manual operator actions as compensatory actions to close the valve and maintain the RWST operable.

In May 2010, shortly after the RWST had been aligned to FPPS, the inspectors questioned this practice. The licensee entered the issue into their CAP as AR #401863 and decided not to continue this practice until the issue was resolved. The licensee closed the AR, noting that additional information would need to be added to their original justification for RWST operability. In September 2010, the Residents and NRR staff discussed the issue with the licensee and concluded that the 10 CFR 50.59 evaluation for the procedure change failed to identify that opening CT-23 (RWST to Spent Fuel Pool Pump Suction Valve) would require declaring the RWST inoperable regardless of what administrative controls were in place to close the valve and that the licensee should have sought a License Amendment to perform this activity.

Analysis: The failure to comply with the actions of TS LCO 3.1.2.6 while the RWST was aligned to the non-seismic FPPS for purification, causing the RWST to be inoperable on May 24, 2010, was a performance deficiency. The performance deficiency was more than minor because it affected the Design Control attribute of the Mitigating System Cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, when the FPPS was aligned to the RWST, the licensee did not declare the RWST inoperable even though the FPPS is a non-safety, non-seismically qualified system. The inspectors evaluated the significance of this finding using Attachment 4 of IMC 0609 and determined that it was a very low safety significance (Green) because it was a design or qualification deficiency confirmed not to result in loss of operability or functionality; did not represent a loss of system safety function; did not represent actual loss of safety function of a single train for longer than its Technical Specification (TS) Allowed Outage Time; did not represent an actual loss of safety function of one or more non-TS trains of equipment designated as risk-significant; and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The finding had a cross-cutting aspect of Conservative Assumptions, as described in the Decision Making component of the Human Performance cross-cutting area because assumptions used in the justification to support the procedure change (i.e., a license amendment was not required to support the procedure change) to OP-116.01 were non-conservative and the review of the issue in May 2010 did not adequately validate the assumptions (H.1(b)).

Enforcement: TS LCO 3.1.2.6 requires the RWST to be operable in Modes 1 through 4. If the RWST is inoperable in Modes 1 though 4, then the licensee is required to enter

LCO 3.1.2.6 action statement “b.” Action statement “b” requires that the RWST be restored to operable status within 1 hour or be in at least Hot Standby within the next 6 hours and in Cold Shutdown within the following 30 hours. Contrary to the above, the licensee failed to enter LCO 3.1.2.6 action statement “b” when CT-23, RWST to Spent Fuel Pool Pump Suction Valve, was opened for approximately 24 hours in 2010. The RWST is inoperable when CT-23 is open because the FPPS is a non-safety, non-seismically qualified system. The licensee took corrective actions and revised OP-116.01 to remove the capability to purify the RWST in Mode 1-4. Because the finding is of very low significance and has been entered into the CAP as AR #422180 and consistent with the NRC Enforcement Policy, this violation is being treated as a non-cited violation and is designated as NCV 05000400/2010005-03, “Failure to comply with the limiting conditions for operation, while the Refueling Water Storage Tank was aligned to the non-seismically qualified Fuel Pool Purification System.”

1R19 Post Maintenance Testing

a. Inspection Scope

The inspectors reviewed the following four post-maintenance (PM) 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-1834 and OST-1040	Essential Services Chilled Water (ESCW) Isolation Valve Remote Position Indication Test Two Year Interval Modes 1-6 and ESCW System Operability Quarterly Interval Modes 1 – 6	WO #1863446, The Bolt that Connects the Actuator to 1SW-1055 (Return from “A” Chiller Condenser) is out	December, 16, 2010
OST-1122	Train “A” 6.9 kV Emergency Bus Under Voltage Trip Activity Device Operation	WO #1831428, Replace 86 UV/SA Lockout Relay Switch	October 14, 2010
WO #1658137-02	1A-6 (Main Feedpump) Breaker Test	WO #1658137-02	November 10, 2010
WO #1843093-02	Main Generator Lockout Relay Continuity Checks	WO #1843093-01, Replace Main Generator Lockout Relay	November 5, 2010

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.

b. Findings

.1 Deenergization of the "B" Safety Bus and Loss of Decay Heat Removal

Introduction: A self-revealing Green NCV of TS 6.8.1, Procedures, was identified for the licensee's failure to develop an adequate procedure for the post maintenance test of the recently replaced main generator lockout relay (MGLR). Specifically, the licensee failed to ensure that the post maintenance testing (PMT) was within the clearance boundary that was established for the MGLR replacement. This resulted in the deenergization of the "B" Safety Bus and the "B" Residual Heat Removal (RHR) pump, which was the only pump providing decay heat removal (DHR).

Description: During the refueling outage, on November 4, 2010, the licensee replaced the MGLR. The replacement of the MGLR was performed under Clearance Order (CO) #236677 with no issue.

While deciding what type of PMT would be appropriate, licensee personnel referred to procedure PLP-400, Post Maintenance Testing; however, this procedure did not provide guidance for developing PMTs associated with complex relays such as the MGLR. Licensee personnel determined that the PMT for the MGLR would include continuity checks to be performed under WO #1843093. The licensee determined that the PMT could be performed within the original work boundary clearance established in CO #236677. This determination proved to be wrong because the PMT for WO #1843093 directed the technicians to perform continuity checks outside of the clearance boundary on energized relays.

On November 5, 2010, the technicians performed the PMT in accordance with WO #1843093. During the continuity checks, which were performed outside of the clearance boundary, a short circuit was inadvertently developed which energized relay CR3/1748. As a result, relay CR3/1748 sensed an electrical fault and performed its intended function of deenergizing the "B" Safety Bus. The "B" RHR pump was the only pump providing DHR, and was also deenergized.

As corrective action, the licensee entered AOP-25, Loss of One Emergency AC Bus and restored DHR with the "B" RHR pump in approximately three minutes. The resultant increase in Reactor Coolant System temperature was approximately one degree Fahrenheit. Additionally, the licensee plans to revise PLP-400, Post Maintenance Testing, to provide personnel with additional guidance when developing PMTs for protective relays.

Analysis: The licensee's failure to develop an adequate PMT for the recently replaced MGLR was a performance deficiency. The performance deficiency was more than minor because it is associated with the procedure quality attribute of the Mitigating Systems cornerstone, and it affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, it resulted in the inadvertent deenergization of the "B" Safety Bus and loss of DHR. Using IMC 0609, "Significance Determination Process," Phase 1 screening worksheet of the SDP, the inspectors determined that the use of Appendix G, Shutdown Operations Significance Determination Process, was necessary. Using Checklist 3 of Attachment 1 of Appendix G, the inspectors determined that this issue affected both the DHR equipment guidelines and the emergency electrical bus guidelines and therefore required a Phase 2 analysis. Using Worksheet 8 of Attachment 2 of Appendix G, the inspectors determined that recovery credit was appropriate because 1) sufficient time was available to implement these actions, 2) environmental conditions allowed access where needed, 3) procedures existed, 4) training was conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions was available and ready for use. Using a time to boil of greater than one hour and the fact that the steam generators were not available for cooling, the result of the Phase 2 was that a Phase 3 analysis was necessary. A regional Senior Reactor Analyst evaluated the performance deficiency under the Phase 3 protocol of the Significance Determination Process. Given that the performance deficiency caused an interruption of Residual Heat Removal (RHR) while in cold shutdown, the evaluation was performed as a precursor event under NRC Manual Chapter 0609, Appendix G. The framework of the evaluation was based upon Worksheet 8, Loss of RHR with Reactor Coolant System intact. The major assumptions of the evaluation included:

- The Reactor Coolant System was closed with level in the Pressurizer
- There was water level in the secondary side of the Steam Generators with a Motor-Driven Auxiliary Feed Water Pump available, if needed to provide forced flow to the Steam Generators
- RHR via Steam Generator forced cooling was viable through operator actions with a failure probability estimated in the one in fifty demand range
- The time to boil, given no forced core cooling, was approximately four hours
- All ECCS mitigation equipment was available, including instrumentation

The initiating event frequency for loss of RHR was set to always happen as the surrogate for the performance deficiency. Worksheet 8 was solved with the steam generator (SG) function set to zero but with "2 points" being assigned to the recovery function for being able to establish this function prior to core damage. The dominant accident sequence was loss of RHR followed by operators failing to recover RHR in the short term, failing to initiate feed and bleed, or failing to place the Steam Generators into service for RHR. Based on these parameters, the phase 3 evaluation indicated that the performance deficiency was characterized as of very low safety significance (Green). The finding has a cross-cutting aspect of Work Coordination, as described in the Work Control component of the Human Performance cross-cutting area because the licensee did not understand the potential operational impact of the work activities or adequately account for current plant conditions (H.3(b)).

Enforcement: TS 6.8.1, Procedures, requires that written procedures shall be established, implemented, and maintained, covering applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978. Section 9 of Appendix A of Regulatory Guide 1.33 requires procedures for maintenance that can affect the performance of safety related systems. The licensee's procedure for developing PMTs that can affect performance of safety related systems is PLP-400, Post Maintenance Testing. The licensee's PMT for the MGLR replacement was developed using the guidance from PLP-400, and was implemented in WO#1843003. Contrary to this requirement, the licensee's procedure PLP-400 failed to provide adequate guidance for developing PMTs associated with complex relays such as the MGLR. Specifically, the PMT directed the technicians to perform continuity checks outside of the clearance boundary on energized relays. This resulted in the inadvertent deenergization of the "B" Safety Bus and the "B" Residual Heat Removal (RHR) pump, which was the only pump providing decay heat removal (DHR). As corrective action, the licensee entered AOP-25, Loss of One Emergency AC Bus and restored DHR with the "B" RHR pump in approximately three minutes. The resultant increase in Reactor Coolant System temperature was approximately one degree Fahrenheit. Additionally, the licensee plans to revise PLP-400, Post Maintenance Testing, to provide the work planner with additional guidance in the development of PMTs for protective relays. Because the finding is of very low safety significance and has been entered into the CAP (AR #431732), and consistent with the NRC Enforcement Policy, this violation is being treated as a non-cited violation, and is designated as NCV 05000400/2010005-04, "Inadequate Post Maintenance Test Procedure Results in Deenergization of the "B" Safety Bus and Loss of Decay Heat Removal."

.2 Failure to Follow Procedure to Properly Align the "a' MOC Switch Contacts Associated with Vacuum Circuit Breaker 1A-6

Introduction: A self-revealing Green NCV of TS 6.8.1, Procedures, was identified for the licensee's failure to follow Section D.2.10 of EC #74866R1. Specifically, the licensee failed to properly align the Mechanism Operated Cell (MOC) switch for the "A" MFP breaker 1A-6, resulting in the automatic actuation of the "B" MDAFW pump.

Description: On November 10, 2010, the licensee was performing Section D.2.10 of EC #74866R1, in accordance with the associated WO #01658137 Task 2. Task 1 of the WO required the "a" contacts of the MOC switch to be aligned 15 degrees advance (one notch) from 3 o'clock towards 12 o'clock (counterclockwise). Contrary to these instructions, the licensee personnel aligned the "a" contacts 30 degrees (two notches) from 3 o'clock towards 12 o'clock (counterclockwise). The independent verifier observed the position of the “a” contacts on the MOC switch and also failed to recognize that they were aligned incorrectly.

The MOC switch is a multi-stage switch which operates when the circuit breaker changes state. The circuit breaker must be racked to connect position in order for the MOC switch to operate. The MOC switch includes form “a” and “b” auxiliary contacts. The form “a” contact is open when the circuit breaker is open and is closed when the circuit breaker is closed. The form “b” contact is closed when the circuit breaker is open and is open when the circuit breaker is closed.

Because the “a” contacts were incorrectly aligned (to the 30 degrees position advanced from 3 o’clock towards 12 o’clock), when the licensee subsequently closed the MFP breaker 1A-6 to verify its functionality, the “B” MDAFW pump auto started since the “a” contacts traveled passed the stationary contact to send the start signal to the AFW pump logic.

The licensee performed a walkdown of the feeder breaker (1A-6) for the “A” MFP and identified that the “a” contacts on the MOC switch had been positioned incorrectly. The MOC switch contacts were subsequently realigned under task 3 of WO #01658137 per the instructions of the EC #74866R1. Post Modification testing to verify contact continuity in the breaker open and breaker closed positions was satisfactory. The licensee entered the issue into their CAP AR #432568.

Analysis: The failure to correctly implement Section D.2.10 of EC #74866R1 on WO #01658137 task 2 was a performance deficiency. The performance deficiency was more than minor because it is associated with the human performance attribute of the Mitigating System cornerstone, and it affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, it resulted in the automatic start of the “B” MDAFW pump. Using IMC 0609, Significance Determination Process, Phase 1 screening worksheet, this finding was determined to be very low safety significance because it was not a design or qualification deficiency confirmed to result in a loss of operability or functionality, did not represent a loss of system safety function, did not result in a loss of safety system function for a single train for greater than TS allowed outage time, did not result in a loss of safety function of one or more non-TS trains of equipment designated as risk significant for greater than 24 hours, and did not screen as potentially risk significant due to seismic, flooding, or severe weather initiating event. The finding has a cross-cutting aspect of Human Error Prevention, as described in the Work Practices component of the Human Performance cross-cutting area because the licensee did not apply sufficient human error prevention tools to ensure the correct alignment of the MOC switch “a” contacts associated with vacuum circuit breaker 1A-6 (H.4(a)).

Enforcement: TS 6.8.1, Procedures, requires that written procedures shall be established, implements, and maintained covering applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978. Section 9 of Appendix A of Regulatory Guide 1.33, Revision 2, February 1978 states that there will be documented instructions for performing maintenance that can affect the performance of safety related equipment. Section D.2.10 of EC #74866R1 on WO #01658137 Task 1 described the proper set-up for the MOC switch contacts “a” and “b” followed by independent verification by a second technician. Contrary to this requirement, on October 27, 2010, the licensee failed to implement the instructions contained in WO 01658137 Task 1. Specifically, the “a” contacts in the MOC switch were aligned incorrectly, which resulted in an invalid actuation of “B” MDAFW pump. As corrective action, the licensee realigned MOC switch contacts under task 3 of WO #01658137. In accordance with the instructions of the EC 74866R1 Post Modification testing, the licensee subsequently verified the correct contact continuity for the breaker open and breaker closed positions. Because the finding is of very low safety significance and has been entered into the CAP (432568), and consistent with the NRC Enforcement Policy, this violation is being treated as a non-cited violation and is designated as NCV 05000400/2010005-05, “Failure to Follow Procedure to Properly Align the MOC Switch Contacts Associated With Breaker 1A-6 Results in Actuation of the “B” MDAFW Pump.”

1R20 Refueling and Outage Activities

a. Inspection Scope

For the outage that began on October 1, 2010 and ended on November 13, 2010, the inspectors evaluated licensee outage activities as described below to verify that licensees considered risk in developing outage schedules, adhered to administrative risk reduction methodologies they developed to control plant configuration, and adhered to operating license and technical specification requirements that maintained defense-in-depth. The inspectors also verified that the licensee developed mitigation strategies for losses of the following key safety functions:

- Decay heat removal
- Inventory control
- Power availability
- Reactivity control
- Containment integrity

Documents reviewed are listed in the Attachment.

.1 Review of Outage Plan

a. Inspection Scope

Prior to the outage, the inspectors reviewed the outage risk control plan to verify that the licensee had performed adequate risk assessments, and had implemented appropriate risk-management strategies when required by 10 CFR 50.65(a)(4).

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

- AR #426041, Boron Flow Path Temperatures Near Miss;
- AR #426500, Transportation of Stator Deviated from Approved Haul Path.

b. Findings

No findings were identified.

.2 Monitoring of Shutdown Activities

a. Inspection Scope

The inspectors observed portions of the cooldown process to verify that technical specification cooldown restrictions were followed.

b. Findings

No findings were identified.

.3 Licensee Control of Outage Activities

a. Inspection Scope

During the outage, the inspectors observed the items or activities described below to verify that the licensee maintained defense-in-depth commensurate with the outage risk-control plan for key safety functions and applicable TS when taking equipment out of service.

- Clearance activities
- Reactor coolant system instrumentation
- Electrical power
- Decay heat removal
- Spent fuel pool cooling
- Inventory control
- Reactivity control
- Containment closure

The inspectors also reviewed responses to emergent work and unexpected conditions to verify that resulting configuration changes were controlled in accordance with the outage risk control plan, and to verify that control-room operators were cognizant of the plant configuration.

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

- AR #430613, “B” ESW Pump Start without Screen Wash;

- AR #426309, 480V Bus 1E2 Outage, Impacted Refueling Communications with the Main Control Room.

b. Findings

No findings were identified.

.4 Reduced-Inventory Conditions

a. Inspection Scope

The inspectors reviewed commitments from Generic Letter 88-17and confirmed by sampling that those commitments were still in place and adequate. Periodically, during the reduced-inventory conditions, the inspectors reviewed system lineups to verify that the configuration of the plant systems were in accordance with the commitments. During reduced-inventory operations, the inspectors observed operator activities to verify that unexpected conditions or emergent activities did not degrade the operators' ability to maintain the required reactor vessel level.

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

- AR #425853**,** Unable to Detention the Last Six Reactor Vessel Head Studs

b. Findings

No findings were identified.

.5 Refueling Activities

a. Inspection Scope

The inspectors observed fuel handling operations (removal, inspection, and insertion) and other ongoing activities to verify that those operations and activities were being performed in accordance with TS and approved procedures. Also, the inspectors observed refueling activities to verify that the location of the fuel assemblies, including new fuel, was tracked from core offload through core reload. The inspectors reviewed the following ARs associated with this area to verify that the licensee identified and implemented appropriate corrective actions:

- AR #430043, Temporary Fan Unit Installed on Manipulator Crane;
- AR #430894, Polar Crane in Motion Alarm;
- AR #430681, Thread Damage Found on Four Vessel Stud Holes;
- AR #426671, Repetitive Problems Discovered with the Refueling Manipulator Crane.

b. Findings

No findings were identified.

.6 Monitoring of Heatup and Startup Activities

a. Inspection Scope

Prior to mode changes and on a sampling basis, the inspectors reviewed system lineups and/or control board indications to verify that TSs, license conditions, and other requirements, commitments, and administrative procedure prerequisites for mode changes were met prior to changing modes or plant configurations. Also, the inspectors periodically reviewed RCS boundary leakage data and observed the containment integrity controls to verify that the RCS and containment boundaries were in place and had integrity when necessary. Prior to reactor startup, the inspectors walked down the containment to verify that debris has not been left which could affect performance of the containment sumps. The inspectors reviewed reactor physics testing results to verify that core operating limit parameters were consistent with the design.

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

- AR #430819, 60kVA Inverter Swapped to Alternate Source;
- AR #434129, Rod Control Urgent Failure Alarm;
- AR #432635, “C” Charging/Safety Injection Pump Breaker Found in the Test Position.

b. Findings

No findings were identified.

.7 Identification and Resolution of Problems

a. Inspection Scope

Periodically, the inspectors reviewed the items that had been entered into the CAP to verify that the licensee had identified problems related to outage activities at an appropriate threshold and had entered them into the CAP. For significant problems documented in the CAP, the inspectors reviewed the results of the investigations to verify that the licensee had determined the root cause and implemented appropriate corrective actions, as required by 10 CFR 50, Appendix B, Criterion XVI, Corrective Action.

b. Findings

No findings were identified.

1R22 Surveillance Testing

.1 Routine Surveillance Testing

a. Inspection Scope

For the two surveillance tests listed below, the inspectors observed the surveillance tests and/or reviewed the test results to verify the tests met TS surveillance requirements, UFSAR commitments, inservice 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.

- OST-1000, Power Range Heat Balance, Online Calculation, Daily Interval, Mode 1 (Above 15% Power) on November 15, 2010;
- OST-1823, "A" Emergency Diesel Generator (EDG) Operability, on October 3, 2010.

b. Findings

Introduction: A self-revealing green NCV of TS 6.8.1, Procedures, was identified for the licensee's failure to properly establish procedural guidance to maintain the Program "C" relay wiring configuration in the "A" Sequencer following maintenance.

Description: On October 2, 2010, during the performance of OST-1823, "A" Emergency Diesel Generator (EDG) Operability, the EDG started and tied to its "A" 6.9 kV safety bus during the first under voltage signal in the surveillance test; however, during the second under voltage signal test on October 3, the EDG started but failed to tie to the safety bus as expected because the 86UV/SA relay failed to initiate load shedding on the "A" safety bus.

After replacement of the 86UV/SA relay, the licensee also identified that a wiring error existed on the Program "C" relay in the "A" Sequencer. The lead from TB7-16-10 was incorrectly wired to Program "C" relay terminal 1J, instead of terminal 1L. This wiring error had allowed negative DC current into the positive DC control circuit for breaker "A" of the "A" Safety Bus. In this configuration, a short circuit through various relay and switch contacts in the control circuit existed, and caused damage to the 86UV/SA relay during the first portion of the surveillance test on October 2, 2010.

The licensee's review of past maintenance revealed that the Program "C" relay had been replaced in accordance with WO #1312555 on April 28, 2009 as a planned replacement activity during RFO15. OPS-NGGC-1303, Independent Verification, Attachment 5 verification sign-off sheet, was used as a lift-land record during the relay replacement and indicated that wire TB7-16-10 had been lifted and re-landed at the same incorrect location at terminal 1J instead of 1L. Therefore, OPS-NGGC-1303, Independent Verification, was inadequate to ensure the required wiring configuration was maintained because it did not require the use of plant drawings to verify the "As Built" configuration when lifting and landing leads. It could not be determined whether the wiring error had existed prior to April 28, 2009, or if it occurred during the relay's replacement in RFO 15.

In either case, the wiring error resulted in the failure of the 86UV/SA relay. Further inspection indicated that the impact of the wiring error on the EDG's ability to perform its safety function was that the EDG would still automatically tie to the safety bus one time following a valid under voltage condition, at which time the 86UV/SA relay would have been damaged. Because of this 86UV/SA relay damage, the EDG failed to automatically tie to its safety bus during second under voltage condition in the surveillance test on October 3, 2010.

Analysis: The failure to properly establish procedural guidance to maintain the required Program "C" relay wiring configuration in the "A" Sequencer following maintenance on April 28, 2009, was a performance deficiency. The performance deficiency was more than minor because it affected the procedure quality attribute of the Mitigating System cornerstone objective of ensuring the availability and reliability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the leads being incorrectly landed prevented the "A" EDG from automatically re-energizing the "A" 6.9kV Bus. Using IMC 0609, "Significance Determination Process," Phase 1 Worksheet, the inspectors concluded that a Phase 2 evaluation was required because this finding represented a loss of safety function of the "A" 6.9kV safety bus. 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 and 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 Green. Major contributors to the significance were the short exposure time due to the finding's ability to be detected during the monthly EDG testing, and the requirement that the condition would have to be "set up" with a prior undervoltage condition before a failure would take place. The short duration resulted from the ability of the monthly emergency diesel generator (EDG) test to detect a failed relay that had been "set up" by a loss of offsite power (LOOP) or loss of bus prior to the occurrence of the second LOOP. The second condition required the risk analysis result from a T/2 exposure to be multiplied by the annual likelihood of a LOOP or loss of bus that would fail the relay in a way that was not detectable until a test or actual event caused a failure. This significantly reduced to risk impact of the finding. The finding has a cross-cutting aspect of Documentation and Component Labeling, as described in the Resources component of the Human Performance cross-cutting area because the licensee did not effectively communicate expectations regarding the utilization of design drawings to aid in the proper completion of the verification sign-off form (OPS-NGGC-1303) (H.2(c)).

Enforcement: TS 6.8.1, Procedures, requires that written procedures shall be established, implemented, and maintained covering applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February, 1978. Section 9 of Regulatory Guide 1.33, Appendix A states, in part, that procedures are required for Maintenance that can affect performance of safety related equipment and that this maintenance should be properly pre-planned and performed in accordance with written procedures, documented instructions or drawings appropriate to the circumstances. The licensee's procedure for verifying that leads are correctly lifted and landed during the replacement of the sequencer Program “C” relay is OPS-NGGC-1303, Independent Verification. Contrary to the above requirement, procedure OPS-NGGC-1303, Independent Verification, was inadequate to ensure the required wiring configuration because it did not require the use of plant drawings to verify the “As Built” configuration when lifting and landing leads, which ultimately led to the deenergization of the “A” 6.9kV safety bus. The licensee took corrective action and replaced the 86UV/SA relay, tested components within the circuit that could be affected, corrected the wiring issue, and issued a memo that set expectations for utilizing plant design drawings when lifting/landing leads. Additionally, the licensee planned to incorporate procedural guidance that require design drawings to be used when lifting and landing leads for all Mitigating System Performance Indicator (MSPI) systems and zero-tolerance equipment failure critical components and systems. Because the finding is of very low safety significance and has been entered into the CAP as AR #424668 and consistent with the NRC Enforcement Policy, this violation is being treated as a non-cited violation and is designated as NCV 05000400/2010005-06, “Inadequate Procedural Guidance to Properly Lift/Land Leads.”

.2 <u>In service Testing (IST) Surveillance</u>

a. <u>Inspection Scope</u>

The inspectors reviewed the performance of OST-1073, Emergency Diesel Generator (EDG) Operability Test Monthly Interval Modes 1-6 on December 17, 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 following components: “B” EDG Fuel Oil Transfer Pump, 1DF0-186 (“B” EDG Fuel Oil Transfer Pump Discharge Check Valve), and 1DFO-191 (“B” EDG Fuel Oil Day Tank Inlet Valve). 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
- Range and accuracy of test instruments
- Required corrective actions

b. <u>Findings</u>

No findings were identified.

.3 <u>Containment Isolation Valve Testing</u>

a. <u>Inspection Scope</u>

The inspectors reviewed the test results for the following activities to determine whether risk-significant systems and equipment were capable of performing their intended safety function and to verify testing was conducted in accordance with applicable procedural and TS requirements. Specifically, the operability of containment isolation valve 1MS-72 ("C" Steam Generator Steam Supply to Auxiliary Feed water Turbine) was verified via the following surveillance tests:

- OST-1311, Auxiliary Feedwater Valves Remote Position Indication Test 2 Year Interval Modes 4 – 6;
- OST-1411, Auxiliary Feedwater Pump 1X-SAB Operability Test Quarterly Interval Mode 1,2,3.

The inspectors observed in-plant activities and reviewed procedures and associated records to determine whether: any preconditioning occurred; the effects of the testing were adequately addressed by control room personnel or engineers prior to the commencement of the testing; acceptance criteria were clearly stated, demonstrated operational readiness, and were consistent with the system design basis; plant equipment calibration was correct, accurate, and properly documented; as left setpoints were within required ranges; and the calibration frequency were in accordance with TS, the UFSAR, procedures, and applicable commitments; measuring and test equipment calibration was current; test equipment was used within the required range and accuracy; applicable prerequisites described in the test procedures were satisfied; test frequencies met TS requirements to demonstrate operability and reliability; tests were performed in accordance with the test procedures and other applicable procedures; jumpers and lifted leads were controlled and restored where used; test data and results were accurate, complete, within limits, and valid; test equipment was removed after testing; where applicable, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable; where applicable for safety-related instrument control surveillance tests, reference setting data were accurately incorporated in the test procedure; where applicable, actual conditions encountering high resistance electrical contacts were such that the intended safety function could still be accomplished; prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test; equipment was returned to a position or status required to support the performance of its safety functions; and all problems identified during the testing were appropriately documented in the CAP. 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 #282356, Probable Leakage from 1MS-70 or 1MS-72;
- AR #165225, In-service Testing Program Limiting Value Discrepancies.

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstones: Occupational Radiation Safety and Public Radiation Safety

2RS1 Radiological Hazard Assessment and Exposure Controls

a. Inspection Scope

Hazard Assessment and Instructions to workers: During facility tours, the inspectors directly observed labeling of radioactive material and postings for radiation areas, High Radiation Areas (HRAs), and airborne radioactivity areas established within the Radiologically Controlled Area (RCA). The inspectors independently measured radiation dose rates or directly observed conduct of licensee radiation surveys for selected RCA areas. The inspectors reviewed survey records for several plant areas including surveys for alpha emitters, hot particles, airborne radioactivity, gamma surveys with a range of dose rate gradients, and pre-job surveys for selected Refueling Outage (RFO) work activities. The inspectors also discussed changes to plant operations that could contribute to changing radiological conditions since the last inspection with cognizant licensee representatives. For selected RFO jobs, the inspectors attended pre-job briefings and reviewed Radiation Work Permit (RWP) details to assess communication of radiological control requirements and current radiological conditions to workers.

Hazard Control and Work Practices: The inspectors evaluated access barrier effectiveness for selected U1 Locked HRA and Very HRA locations. Changes to procedural guidance for Locked HRA and Very HRA controls were discussed with selected Radiation Protection (RP) supervisors. Controls and their implementation for storage of irradiated material within the Spent Fuel Pool (SFP) were reviewed and discussed in detail with cognizant licensee representatives. Established radiological controls (including airborne controls) were evaluated for selected tasks including work in auxiliary building HRAs, and radwaste processing and storage. In addition, licensee controls for areas where dose rates could change significantly as a result of plant shutdown and U1 refueling operations were reviewed and discussed.

Occupational workers' adherence to selected RWPs and RP technician (RPT) proficiency in providing job coverage were evaluated through direct observations and interviews with licensee staff. Electronic Dosimeter (ED) alarm set points and worker stay times were evaluated against area radiation survey results for selected U1 RFO activities. ED alarm logs were reviewed and worker response to dose and dose rate alarms during selected work activities was evaluated. For HRA tasks involving significant dose rate gradients, the inspectors evaluated the use and placement of whole body and extremity dosimetry to monitor worker exposure.

Control of Radioactive Material: The inspectors observed surveys of material and personnel being released from the RCA using small article monitor, personnel contamination monitor, and portal monitor instruments. The inspectors reviewed records for selected release point survey instruments and discussed equipment sensitivity, alarm setpoints, and release program guidance with cognizant licensee staff. The inspectors compared recent 10 CFR Part 61 results for the Dry Active Waste (DAW) radioactive waste (radwaste) stream with radionuclides used in calibration sources to evaluate the appropriateness and accuracy of release survey instrumentation. The inspectors also reviewed records of leak tests on selected sealed sources and discussed nationally tracked source transactions with cognizant licensee staff.

Problem Identification and Resolution: Nuclear Condition Reports (NCRs) associated with radiological hazard assessment and control were reviewed and assessed. The inspectors evaluated the licensee's ability to identify and resolve the issues in accordance with procedure CAP-NGGC-0200, Condition Identification and Screening process, Revision (Rev.) 33. The inspectors also evaluated the scope of the licensee's internal audit program and reviewed recent assessment results.

RP activities were evaluated against the requirements of UFSAR Section 12; TS Sections 5.4 and 5.7; 10 CFR Parts 19 and 20; and approved licensee procedures. Licensee programs for monitoring materials and personnel released from the RCA were evaluated against 10 CFR Part 20 and IE Circular 81-07, Control of Radioactively Contaminated Material. Documents reviewed are listed in Section 2RS1 of the Attachment.
The inspectors completed all specified line-items detailed in Inspection Procedure (IP) 71124.01 (sample size of 1).

b. Findings

No findings were identified.

2RS2 Occupational ALARA Planning and Controls

a. Inspection Scope

Radiological Work Planning: The inspectors reviewed a number of ALARA Work Plans (AWP) associated with the previous refueling outage and the current refueling outage. The AWPs were reviewed with respect to activity evaluation, exposure estimates, and exposure mitigation requirements. The inspectors verified that the plans identified appropriate mitigation features and incorporated lessons learned from previous outages, and defined reasonable dose goals. For AWPs from the previous outage, the inspectors compared the results achieved in terms of actual dose vs. planned dose and actual hours vs. estimated hours, reviewed in-progress and post-job ALARA reviews, and discussed the job planning, performance, and reviews with ALARA staff. For AWPs associated with the current refueling, the inspectors tracked dose-to-date on select jobs, comparing estimates with actual and observed development of selected in-progress reviews.

Verification of Dose Estimates and Exposure Tracking Systems: For the ALARA work plans reviewed, the inspectors reviewed the assumptions and basis for the dose rate and man-hour estimates. The inspectors discussed with ALARA staff the means by which wrench-hours were derived from the WO hours provided by craft supervision to

ALARA staff. The inspectors verified the licensee had established several means to track and trend doses for ongoing work activities. The inspectors observed discussions between ALARA staff and job owners related to in-progress reviews and re-planning work when dose/hour budgets were exceeded or when emergent work and/or changes in scope were encountered.

Source Term Reduction and Control The inspectors determined the historical trends and current status of the plant source term through review of records. Through interviews and document review, the inspectors assessed the licensee's current activities and future plans related to source term reduction, including shutdown chemistry and response to problems with fuel in previous cycles.

Radiation Worker Performance: The inspectors observed radiation worker performance through direct observation, via remote camera monitoring, and via telemetry. Jobs observed were associated with the refueling outage.

Problem Identification & Resolution: Licensee CAP documents associated with ALARA planning and controls were reviewed and assessed. This included review of selected Action Requests (ARs), self-assessments, and audits. The inspectors evaluated the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with procedure CAP-NGGC-0200, Condition Identification and Screening Process, Rev. 33. Licensee CAP documents reviewed are listed in Section 2RS2 of the Attachment.

Radiation protection activities were evaluated against the requirements of UFSAR Section 12; 10 CFR Parts 19 and 20; and approved licensee procedures. Records reviewed are listed in Section 2RS2 of the report Attachment.

The inspectors completed all specified line-items detailed in IP 71124.02 (sample size of 1).

b. Findings

No findings were identified.

2RS8 Radioactive Solid Waste Processing and Radioactive Material Handling, Storage, and Transportation

a. Inspection Scope

Waste Processing and Characterization During inspector walk-downs, accessible sections of the liquid and solid radioactive waste (radwaste) processing systems were assessed for material condition and conformance with system design diagrams. Inspected equipment included radwaste storage tanks; resin transfer piping, resin and filter packaging components; and abandoned reverse osmosis equipment. The inspectors discussed component function, processing system changes, and radwaste program implementation with licensee staff.

The 2009, Effluent Report and radionuclide characterizations from 2008 – 2010, for each major waste stream were reviewed and discussed with radwaste staff. For primary resin, radwaste filters, and Dry Active Waste (DAW) the inspectors evaluated analyses for hard-to-detect nuclides, reviewed the use of scaling factors, and examined quality assurance comparison results between licensee waste stream characterizations and outside laboratory data. Waste stream mixing and concentration averaging methodology for resins and filters was evaluated and discussed with radwaste staff. The inspectors also reviewed the licensee's program for monitoring changes in waste stream isotopic mixtures.

Radwaste processing activities and equipment configuration were reviewed for compliance with the licensee's Process Control Program and UFSAR, Chapter 11. Waste stream characterization analyses were reviewed against regulations detailed in 10 CFR Part 20, 10 CFR Part 61, and guidance provided in the Branch Technical Position on Waste Classification (1983).

Radioactive Material Storage During walk-downs of indoor and outdoor radioactive material storage areas, the inspectors observed the physical condition and labeling of storage containers and the posting of Radioactive Material Areas. The inspectors also reviewed licensee procedural guidance for storage and monitoring of radioactive material. Radioactive material and waste storage activities were reviewed against the requirements of 10 CFR Part 20.

Transportation The inspectors directly observed preparation activities for a shipment of contaminated outage equipment. The inspectors noted package markings and labeling, performed independent dose rate measurements, and interviewed shipping technicians regarding Department of Transportation (DOT) regulations.

Selected shipping records were reviewed for consistency with licensee procedures and compliance with NRC and DOT regulations. The inspectors reviewed emergency response information, DOT shipping package classification, waste classification, radiation survey results, and evaluated whether receiving licensees were authorized to accept the packages. Licensee procedures for opening and closing shipping casks were compared to Certificate of Compliance requirements and vendor manual recommendations. In addition, training records for selected individuals currently qualified to ship radioactive material were assessed.

Transportation program implementation was reviewed against regulations detailed in 10 CFR Part 20, 10 CFR Part 71 (which requires licensees to comply with DOT regulations in 49 CFR Parts 107, 171-180, and 390-397), as well as the guidance provided in NUREG-1608. Training activities were assessed against 49 CFR Part 172 Subpart H.

Problem Identification and Resolution The inspectors reviewed NCRs in the area of radwaste/shipping. The inspectors evaluated the licensee's ability to identify and resolve the issues in accordance with procedure CAP-NGGC-0200, "Condition Identification and Screening Process", Rev. 33. The inspectors also evaluated the scope of the licensee's internal audit program and reviewed recent assessment results.

The inspectors completed one sample as required by inspection procedure 71124.08. Documents reviewed during the inspection are listed in Section 2RS8 of the report Attachment.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

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.

Mitigating Systems Cornerstone

- Mitigating Systems Performance Index, Residual Heat Removal
- Mitigating Systems Performance Index, Cooling Water Systems

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index performance indicators (MSPI) listed above for the period from the third quarter 2009 through the third quarter 2010. 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 reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. 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 and none were identified. Specific documents reviewed are described in the Attachment.

Occupational Radiation Safety Cornerstone: The inspectors reviewed Performance Indicator (PI) data collected from January 1, 2010, through September 30, 2010, for the Occupational Exposure Control Effectiveness PI. For the reviewed period, the inspectors assessed CAP records to determine whether HRA, VHRA, or unplanned exposures, resulting in TS or 10 CFR 20 non-conformances, had occurred during the review period. In addition, the inspectors reviewed selected personnel contamination event data, internal dose assessment results, and ED alarms for cumulative doses and/or dose rates exceeding established set-points. The reviewed data were assessed against guidance contained in NEI 99-02, "Regulatory Assessment Indicator Guideline." The reviewed documents relative to these PI reviews are listed in Sections 2RS1 and 4OA1 of the report Attachment.

Public Radiation Safety Cornerstone: The inspectors reviewed the Radiological Control Effluent Release Occurrences PI results for the Public Radiation Safety Cornerstone from January 1, 2010, through September 30, 2010. For the assessment period, the inspectors reviewed cumulative and projected doses to the public and NCR documents related to Radiological Effluent Technical Specifications/Offsite Dose Calculation Manual issues. Documents reviewed are listed in section 4OA1 of the Attachment.

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 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a review of the licensee's CAP 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 through December 31, 2010, although some examples expanded beyond those dates where the scope of the trend warranted.

The review also included issues documented outside the normal CAP; i.e., in major equipment problem lists, repetitive and/or 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

No findings were identified. The inspectors observed that the licensee performed adequate trending reviews. The licensee routinely reviewed cause codes, involved organizations, key words, and system links to identify potential trends in the CAP data. The inspectors compared the licensee process results with the results of the inspectors' daily screening to identify any discrepancies or potential trends in the CAP data that the licensee had failed to identify.

The inspectors identified that an adverse trend in the area of Human Error prevention. Specifically, the failure to adequately implement maintenance activities have resulted in the inadvertent actuation of safety related systems. The following issues illustrate the presence of this trend:

- AR #381672, Failure to Follow Procedure while Reinstalling a Relay in the "B" Sequencer Cabinet Results in the "B" Motor Driven Auxiliary Feedwater (AFW) Pump Actuation;
- AR #430289, Inadvertent Safety Injection Signal During Testing;
- AR #432568, AFW Pump Automatically Started when Main Feedwater Pump Started.

This trend was entered into the licensee's CAP as AR #441282 to address the need for increased management attention.

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 were identified.

.2 (Closed) Reactor Coolant System Dissimilar Metal Butt Welds (TI 2515/172, Revision 1)

a. Inspection Scope

Based on the schedule of dissimilar metal butt weld (DMBW) examinations under MRP-139, no examinations were required for the current Unit 1 refueling outage (N1R21) and hence none were performed. Additionally, the licensee had not made any changes to the MRP-139 inspection program since the NRC had previously reviewed this program.

b. Observations

This completes the TI-2515/172 requirements for Harris Unit 1.

In accordance with requirements of TI 2515/172, Revision 1, the inspectors evaluated and answered the following questions:

(1) Implementation of the MRP 139 Baseline Inspections

- Have the baseline inspections been performed or are they scheduled to be performed in accordance with MRP 139 guidance?

  Yes. All baseline inspections have been placed on the schedule per MRP-139 guidance.

- Is the licensee planning to take any deviations from the MRP 139 baseline inspection requirements of MRP 139? If so, what deviations are planned, what is the general basis for the deviation, and was the NEI 03 08 process for filing a deviation followed?

  The licensee has taken no deviations and has no plans to take any deviations within the scope of the MRP-139 requirements.

(2) Volumetric Examinations

This portion of the TI was inspected during the RFO16 outage within the guidelines of the MRP-139 baseline inspection for hot leg components. There are further requirements for inspection that are expected to be fulfilled for the cold leg during RFO17. This portion was previously covered in NRC Inspection Report 05000400/2008004.

(3) Weld Overlays

This portion of the TI was not inspected during the period of this inspection report, but was previously covered in NRC Inspection Report 05000400/2008004.

(4) Mechanical Stress Improvement (SI)

The licensee has some stress improvement activities planned during R16 on hot leg components. The scope of these activities is within the guidelines of MRP-139.

(5) Application of Weld Cladding and Inlays

There were no weld cladding or inlay activities performed or planned by this licensee to comply with their MRP 139 commitments.

(6) Inservice Inspection Program

- Has the licensee prepared an MRP 139 inservice inspection program? If not, briefly summarize the licensee's basis for not having a documented program and when the licensee plans to complete preparation of the program.

  No. The licensee did not have a standalone MRP-139 inservice inspection program document. However, the licensee's MRP-139 inservice inspection program was included in their ASME Section XI Inservice Inspection Program (ISI Program) and also attached as augmented inspections to the inservice inspection program. The inspectors reviewed the Harris Nuclear Plant ISI Plan. The licensee had revised the ISI Plan to reflect the examination methods and frequencies for the MRP-139 ISI requirements.

- In the MRP 139 inservice inspection program, are the welds appropriately categorized in accordance with MRP 139? If any welds are not appropriately categorized, briefly explain the discrepancies.

  Yes. The welds were appropriately categorized by the licensee responsible engineer.

- In the MRP 139 inservice inspection program, are the inservice inspection frequencies, which may differ between the first and second intervals after the MRP 139 baseline inspection, consistent with the inservice inspections frequencies called for by MRP 139?

  Yes. The licensee plans inspection frequencies for welds in the MRP-139 ISI program to be consistent with the requirements of MRP-139.

- If any welds are categorized as H or I, briefly explain the licensee's basis of the categorization and the licensee's plans for addressing potential PWSCC.

  There are no DMBWs categorized as H or I.

- If the licensee is planning to take deviations from the MRP - 139 inservice inspection guidelines, what are the deviations and what are the general bases for the deviations? Was the NEI 03 08 process for filing deviations followed?

The licensee had not planned to take any deviations from MRP-139 requirements.

c. Findings

No findings were identified.

.3 (Closed) TI 2515/179 Verification of Licensee Responses to NRC Requirement for Inventories of Materials Tracked in the National Source Tracking System (NSTS) Pursuant to Title 10, Code of Federal Regulations, Part 20.2207 (10 CFR 20.2207)

a. Inspection Scope

The inspectors performed the TI concurrent with IP 71124.01 Radiation Hazard Analysis. The inspectors reviewed the licensee's source inventory records and identified the sources that met the criteria for reporting to the NSTS. The inspectors visually identified the sources contained in various calibration systems and verified the presence of the source by direct radiation measurement using a calibrated portable radiation detection survey instrument. The inspectors reviewed the physical condition of the irradiation device. The inspectors reviewed the licensee's procedures for source receipt, maintenance, transfer, reporting and disposal. The inspectors reviewed documentation that was used to report the sources to the NSTS. Documents reviewed are listed in sections 2RS1 of the Attachment.

b. Findings

No findings were identified.

4OA6 Management Meetings

.1 Exit Meeting Summary

On January 20, 2011, the inspector presented the inspection results to Mr. John Dufner and other members of the licensee staff. The inspectors confirmed that proprietary information was not provided or examined during the inspection period.

On October 15, 2010 and October 22, 2010 the inspectors presented the results of the ISI inspections to licensee management. The licensee acknowledged the inspection results. The inspectors confirmed that none of the potential report input discussed was considered proprietary. Proprietary material received during the inspection was returned to the licensee.

On October 29, 2010 and December 9, 2010 the inspectors discussed results of the onsite radiation protection inspections with Mr. Chris Burton, Vice President Shearon Harris Plant, and other responsible staff. The inspectors noted that no proprietary information was reviewed during the course of the inspection. The inspectors noted that some personally identifiable information was reviewed during the course of the inspection and that it was returned to the licensee prior to the exit.

An interim exit with licensee management and staff was conducted on December 16, 2010 and an exit meeting was conducted on January 20, 2011 to discuss the results of the Modifications inspection. Proprietary information reviewed by the team as part of routine inspection activities was returned to the licensee in accordance with prescribed controls.

4OA7 Licensee-Identified Violations

The following violation of very low safety significance (Green) was identified by the licensee and is a violation of NRC requirements which meet the criteria of the NRC Enforcement Policy, for being dispositioned as a Non-Cited Violation.

During Mode 4 operation, TS 3.5.3 requires that one complete train of Emergency Core Cooling System (ECCS) shall be operable. During Mode 3 operation, TS 3.5.2 requires that two complete trains of ECCS shall be operable. Additionally, TS 3.0.4 prohibits transitioning into a Mode when the licensee has not met all of the limiting conditions for operation when the TS action would require a shutdown. Contrary to these requirements, between November 9, 2010 and November 10, 2010, the licensee operated in Mode 4 and transitioned to Mode 3 with both trains of ECCS inoperable. The licensee determined that the root cause of this issue was an operating procedure which incorrectly directed the operator to remove control power to both of the Residual Heat Removal Header Isolation Valves which provided suction to the Charging Safety Injection Pump. As corrective action, the licensee restored control power to the affected valves and revised the procedure. The licensee determined that this issue was reportable and will issue a Licensee Event Report which will be addressed in a future inspection report. This issue was identified in the licensee’s CAP as AR 432567. A regional Senior Reactor Analyst evaluated the performance deficiency under the Phase 3 protocol of the Significance Determination Process. Based upon the results of that evaluation, the performance deficiency was characterized as of very low safety significance (Green). The NRC's most current Probabilistic Risk Assessment model for the Harris plant was used. The surrogates for the performance deficiency were basic events RHR-MOV-CC-25 and RHR-MOV-CC-26, i.e., the “piggyback” motor operated valves, which were set to always be closed for the evaluation. The resulting dominant accident sequence was a Small Break Loss of Coolant Accident with operators failing to depressurize the Reactor Coolant System allowing core cooling via low pressure recirculation and high pressure recirculation failing due to the performance deficiency. The major assumptions for the evaluation included a thirty six hour exposure time and no recovery credit from the performance deficiency.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

C. Burton, Vice President Harris Plant
J. Caves, Supervisor, Licensing/Regulatory Programs
H. Curry, Manager, Nuclear Oversight
J. Dills, Manager, Operations
J. Doorhy, Licensing
J. Dufner, Manager, Engineering
K. Harshaw, Manager, Outage and Scheduling
K. Henderson, Plant General Manager
D. Hooten, Design Engineering Supervisor
G. Kilpatrick, Training Manager
M. Parker, Superintendent, Radiation Protection
J. Price, Design Engineering Manager
J. Robinson, Superintendent, Environmental and Chemistry
T. Slake, Superintendent, Security
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

05000400/2010005-01	NCV	Failure to Properly Implement Procedural Guidance to Maintain the FHBEES Boundary. (Section 1R04)
05000400/2010005-02	NCV	Failure to Follow Procedure Results in Emergency Safeguards Sequencer Actuation and Safety Injection Signal (SIS) while the Plant was in Mode 6. (Section 1R12)
05000400/2010005-03	NCV	Failure to comply with the limiting conditions for operation, while the Refueling Water Storage Tank was aligned to the non-seismically qualified Fuel Pool Purification System. (Section 1R18)
05000400/2010005-04	NCV	Inadequate Post Maintenance Test Procedure Results in Deenergization of the “B” Safety Bus and Loss of Decay Heat Removal. (Section 1R19)
05000400/2010005-05	NCV	Failure to Follow Procedure to Properly Align the MOC Switch Contacts Associated With Breaker 1A-6 Results in Actuation of the “B” MDAFW Pump. (Section 1R19)
05000400/2010005-06	NCV	Inadequate Procedural Guidance to Properly Lift/Land Leads. (Section 1R22)

Closed

2515/172	TI	Reactor Coolant System Dissimilar Metal Butt Welds (Section 4OA5.2)
2515/179	TI	Verification of Licensee Responses to NRC Requirement for Inventories of materials Tracked in the National Source Tracking System (NSTS) Pursuant to Title 10, Code of Federal Regulations, Part 20.2207 (10 CFR 20.2207) (Section 4OA5.3)

LIST OF DOCUMENTS REVIEWED

<u>Section 1R01: Adverse Weather Protection</u>

<u>Procedures</u>
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
EC #70350, Alternate Seal Injection And Back-Up Diesel Generator System
FSAR 2.4, Hydrologic Engineering
FSAR 3.4, Flood Design

<u>Section 1R04: Equipment Alignment</u>

<u>Partial System Walkdown</u>
<u>Fuel Pool Cooling system</u>:
Procedure OP-116 Fuel Pool Cooling
Drawing 2165-S-0805, Simplified Flow Diagram Fuel Pool Cooling System
FSAR 9.1.3, Fuel Pool Cooling

<u>Fuel Handling Building Emergency Exhaust system</u>:
Procedure OP-170, Fuel Handling Building Heating Ventilation and Air Conditioning System,
Drawing 2168-G-0533, Simplified Flow Diagram Fuel Handling Building Heating Ventilation and Air Conditioning System
FSAR 7.3.1.3.4, Emergency Exhaust Systems
OST-1035, Fuel Handling Building Emergency Exhaust Train “A” Operability Test Monthly Interval Whenever Irradiated Fuel Is In The Storage Pool
OST-1047, Fuel Handling Building Emergency Exhaust Train “B” Operability Test Monthly Interval Whenever Irradiated Fuel Is In The Storage Pool
OST-1048, Fuel Handling Building Emergency Exhaust System Operability 18 Month Interval At All Times

<u>Complete System Walkdown</u>
Procedure OP-139 Service Water System
Design Basis Document- 128 Service Water System
Drawing 2165-S-0547, Simplified Flow Diagram Service Water System
FSAR 9.2.1, Service Water System
WO #678670, Replace "B" ESW Pump Wear Rings - Install New Pump
ISI-801, In-service Testing of Valves
PLP-106, Technical Specification Equipment List Program and Core Operating Limits Report

<u>Section 1R05: Fire Protection</u>

FPP-001 Fire Protection Program Manual
FPP-004, Transient Combustible Control

FPP-013, Fire Protection – Minimum Requirements, Mitigating Actions and Surveillance Requirements
FPP-012-03-FHB, Fuel Handling Building Fire Pre-Plan, F07-FHB Operating Floor including New Fuel Pools, Spent Fuel Pools, Fuel transfer Canals, Main transfer Canal, New Fuel Storage Area, and Cask Loading Pool
FPP-012-03-FHB, Fuel Handling Building Fire Pre-Plan, F06-FHB Emergency Exhaust Room and Fuel Pool Demineralizer Room “A” and B
FPP-012-03-FHB, Fuel Handling Building Fire Pre-Plan, F05-FHB Emergency Exhaust Electrical Room
FPP-012-03-FHB, Fuel Handling Building Fire Pre-Plan, F02-Fuel Pool Cooling Equipment Room
FPP-012-02-RAB 236, Reactor Auxiliary Building Elevation 236 Fire Pre-Plan, A14-CVCS and BTRS Heat Exchanger Area
FPP-012-02-RAB 236, Reactor Auxiliary Building Elevation 236 Fire Pre-Plan, A11-Mechanical Penetration Area

Section 1R06: Flood Protection Measures

UFSAR Sections
FSAR 2.4.10, Flooding Protection Requirements
FSAR 3.6A.6, Flooding Analysis
FSAR 3.4, Flood Design

Procedures
AOP-022, Loss of Service Water
OP-139, Service Water System

Section 1R07: Heat Sink Performance

Procedures
EPT- 163, Generic Letter 89-13 Inspections
PLP- 620, Service Water Program (Generic Letter 89-13)
OPT-1512, Essential Chilled Water Turbopak Units Quarterly Inspection/Checks Modes 1-6
OST-1040, Essential Services Chilled Water Systems Operability Quarterly Interval Modes 1 – 6
OPT-1512, Essential Chilled Water Turbopak Units Quarterly Inspection/Checks Modes 1-6
OST-1040, Essential Services Chilled Water Systems Operability Quarterly Interval Modes 1 – 6
MPT-M0091, Heat Exchanger Opening/Closure for NRC Generic Letter 89-13 Inspections
WO #1526926, Assist Engineering With Performance Of EPT-163. Perform MPT-M0091
WO #1615007, Retube “B” Chiller Condenser
WO #1656049, Service Water Leakage Alarm In
WO #746659, Perform Loop Calibration Water Chiller WC-2 (“A”) Flow

Section 1R11: Licensed Operator Requalification

AOP-025, Loss of One Emergency AC Bus (6.9V) or One Emergency DC Bus (125V)
EOP-EPP-004, Reactor Trip Response

Section 1R12: Maintenance Effectiveness

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

Section 1R13: Maintenance Risk Assessments and Emergent Work Evaluation

OMP-003, Outage Shutdown Risk Management
WCM-001, On-line Maintenance

ADM-NGGC-0006, Online Equipment Out of Service (EOOS) Models for Risk Assessment

Section 1R15: Operability Evaluations

OPS-NGGC-1305, Operability Determinations

Section 1R17: Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications

Full Evaluations
EC #53878, De-Energization Charging Pump Discharge Cross-Connect Valves, REG 00171657, 50.59 Screen & Evaluation, Rev. 0
EC #69264, Ultrasonic Feedwater Flow Meter Installation, REG 00419633, 50.59 Screen & Evaluation, Rev. 3
EC #60828, De-Energization of Charging Pump Suction Cross-connect Valves 1CS-168, 1CS-169, 1CS-170, 1CS-171, AR 00176038, 50.59 Screen & Evaluation, Rev. 0
EC #60257, Manual Transfer of C CSIP, AR 00192783, 50.59 Screen & Evaluation, Rev. 5
EC #60541, ESCW Expansion Tanks Check Valve Replacement And Alternate Pressure Makeup, Rev. 1
EC #67999, Manual Action Compensatory Measures for SSD, Rev. 0

Screened Out Items
EC #79004, Put Bonnet Cap on RCP Pump Seal Injection Isolation, REG 00431007, 50.59 Screen, Rev. 0
EC #48021, Eval Design Pressure Increase/CSIP Pump Impeller Replacement, REG 00205474, 50.59 Screen, Rev. 3
EC #64889, Replacement of Containment Isolation Valve Internal Parts, REG 00220061, 50.59 Screen, Rev. 0
EC #66533, 1-SP214, -291 not in Agreement with FSAR, REG 00242290, 50.59 Screen, Rev. 0
EC #66848, 2 & 3B-SB Spent Fuel Pool Cooling Pump Thermal Overloads, AR 00231654, 50.59 Screen, Rev. 0
EC #69249, 1RH-2 & 1RH-40 – Replace Motors to Improve Margin, AR 00281855, 50.59 Screen, Rev. 0
EC #79056, Replace Barton Pressure Transmitter Model 752 with Rosemount Model 1154

Pressure Transmitter for FT-01CS-0124SW, AR 00431585, 50.59 Screen, Rev. 0
EC #63160, System Tie-Ins For The New ESCW Surge Tanks…RFO-13, Rev. 0
EC #65081, Evaluation That Shows That HNP Meets The Requirements Of Generic Letter 2004-02, Rev. 0
EC #69450, Tech Spec 3/4.7.5 Revision (Main Reservoir Level Change – 215 FT to 206 FT), Rev. 3
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RFN H-RP-10-01, S/N HNOS 10-067, Assessment of Radiation Protection, Dated 08/18/10
RFN H-RP-09-01, HNOS 09-088, Assessment of Harris Radiation Protection, Dated 09/16/09

Section 2RS2: Occupational ALARA Planning and Controls

Procedures, Guidance Documents, and Manuals
HPP-602, Radiation Protection Work Planning Process, Rev.4
HPS-NGGC-0014, Radiation Work Permits, Rev. 7
HPS-NGGC-0016, Access Control, Rev.6
PLP-511, Radiation Control and Protection Program, Rev. 23

Reports, Records, and Data
HNP Five Year Dose Reduction Plan 2009-2013
Memo: 2008 Dose Budget, 12/18/2007
Memo: 2009 Dose Budget, 12/11/2009
Memo: 2010 Dose Budget, 1/27/2010
White Paper: Source Term Reduction Summary
Spreadsheet: HNP Historical Performance (2003-2008) and Predictions (2009-2013)
Temporary Shielding Requests (TSR) 10-001, 1RC4-231SN-1
TSR-10-002, 2085 LHSI & RHR: 2070 –CT: 2080 HHSI
TSR 10-003, RCB Pressurizer Platform at Elevation 250' near Azimuth 291
TSR 10-006, Pressurizer Spray Line Riser
TSR 10-007, Incore Sump
TSR 10-009, Pressurizer
TSR 10-014, Shield fuel transfer gap for offloading and reloading the core
TSR 10-022, Reactor Vessel and Piping below permanent cavity seal ring
TSR 10-029, Steam Generator Sludge Lance and FOSAR Activities at the Secondary Hand Holes in all 3 S/Gs
Presentation: ALARA Techniques integrated into the work schedule
Harris Nuclear Plant ALARA Committee Meeting Minutes, May 24, 2010
Harris Nuclear Plant ALARA Committee Meeting Minutes, June 28, 2010
Harris Nuclear Plant ALARA Committee Meeting Minutes, August 13, 2010
ALARA Work Plan (AWP) 09-006, RF0-15 Reactor Headwork / Refueling
AWP 09-007, R15 Seal Table Maintenance Activities
AWP 09-013, 'C' Reactor Coolant Pump Motor Replacement & RCP Preventative Maintenance Activities
AWP 09-029, ISI Activities
AWP 09-006 Reactor Headwork/Refueling-Post-Job ALARA Critique
AWP 09-007, Seal Table Maintenance Activities-Post-Job ALARA Critique
AWP 09-013, 'C' Reactor Coolant Pump Motor Replacement & RCP Preventative Maintenance

Activities Post Job ALARA Critique
AWP 09-029, ISI Activities Post Job ALARA Critique
AWP 10-002, Radiation Protection Activities
AWP 10-007, RF0-16 Reactor Headwork / Refueling
AWP 10-009, Scaffold R-16
AWP 10-010, Insulation R16
AWP 10-010, Shielding R16
AWP 10-014, “A” Reactor Coolant Pump Motor Replacement & RCP Preventative Maintenance Activities
AWP 10-022, Alloy 600 Reactor Vessel Nozzle Mitigation

CAP Documents
Self-Assessment: H-OM-FR-09-01, Focused Review of ALARA Work Plans (AWPs) and Radiation Work Permits (RWPs) to support R15, 3/11/2009
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AR 00343002, There needs to be a better process in the planning stage to evaluate radiological impact from erecting scaffolding prior to the scaffolding foreman/crew arriving at the RCC window to begin work.
AR 00343005, Software familiarity problems
AR 00343017, ALARA briefing work load
AR 00414159, ALARA Committee functioning
AR 00414160, Individual dose ownership

2RS8: Radioactive Material Processing and Transportation

Procedures, Manuals, and Guides
HPS-NGGC-0001, “Radioactive Material Receipt and Shipping Procedure”, Rev. 30
HPS-NGGC-0002, “Vendor Cask Utilization Procedure”, Rev. 17
HPS-NGGC-0003, “Radiological Posting, Labeling and Surveys”, Rev. 15
HPP-800, “Handling Radioactive Material”, Rev. 54
OP-120.04, “Spent Resin Storage and Transfer System”, Rev. 26
PLP-300, “Process Control Program”, Rev. 11
HPP-830, “Process Control Program Implementation”, Rev. 0
CAP-NGGC-0200, “Condition Identification and Screening Process”, Rev. 33

Shipping Records and Radwaste Data
Shipment 09-061, LSA, Filters
Shipment 09-034, LSA, DAW
Shipment 10-051, LSA, Disposable protective clothing
Shipment 09-032, SCO, Outage equipment
Shipment 10-009, LSA, Resin
Shipment 10-041, Limited Quantity, DAW
10 CFR Part 61 Analyses, DAW, 11/6/08 and 11/21/06
10 CFR Part 61 Analysis, F-23 Demin Skid Filters, 11/6/08
10 CFR Part 61 Analyses, Sample Data Set Evaluation, Radwaste Resin, Primary Resin, Spent Fuel Pool Resin, 1/26/10
2009 Annual Radioactive Effluent Release Report

CAP Documents
H-RP-09-01, Assessment of Harris Radiation Protection
AR 437452, Review practice of storing rusted metal RAM boxes on gravel
AR 366763, Secondary side triturated water shipped offsite without required paperwork
AR 424489, Bag of radioactive trash mishandled by processing vendor and found on public road
AR 352576, Hose containing radioactive material found in truck bay

Section 4OA1: Performance Indicator Verification

NEI 99-02, Regulatory Assessment Performance Indicator Guideline
Calculation HNP-F/PSA-0068, NRC Mitigating System Performance Index Basis Document for Harris Nuclear Plant

Procedures
REG-NGGC-0009, NRC Performance Indicators and Monthly Operating Report Data, Rev. 10

Records and Data Reviewed
RETS/ODCM Radiological Effluent Occurrence Data January 2010 – September 2010
Occupational Exposure Control Effectiveness Data January 2010 - September 2010
Reviewed a sampling DRD alarm investigations for 6 individuals occurring between January and September 2010
CAP Documents
AR 315269, Area Found LHRA Upon Entry by RP Tech
AR 332049, Dosimetry Telemetry Communication Gap
AR 333852, Personnel Erecting Scaffold On Wrong RWP
AR 335087, Elevated Dose Rates In the “A” CSIP Room
AR 355183, DRD Alarm Lessons Learned

Section 4OA2: Identification and Resolution of Problems

CAP-NGGC-0200, Corrective Action Program

Section 4OA5: Other Activities

Procedures
Anatec-08, Certification of NDT Personnel, Revision 22
Attachment T of NDE Appendix B, Rev. 7, Preservice and Inservice Inspection of Nuclear Power Plant Components
EGR-NGGC-0208, Steam Generator Integrity Program, Revision 10
ER-AA-MAT-11 Alloy 600 Management Plan, Revision 6
EST-216, Steam Generator Tube Integrity, Revision 19
EST-216, Steam Generator Tube Integrity, Revision 19
Mistras-100-QC-005.2.C, Addendum C: Qualification and Certification of EPRI Qualified Data Analyst (QDA) Personnel, Revision 0
MRS 2.3.2 GEN-13, Mechanical Ribbed Plugging of Steam Generator Tubes, Revision 25
NDEP-0201, Rev. 28, Liquid Penetrant Examination

NDEP-0425, Rev. 8, Ultrasonic Examination of Austenitic Pipe Welds (PDI)
NDEP-0437, Rev. 3, Manual Ultrasonic Examination Procedure for Ferritic Pipe Welds (PDI)
PQP-003, Training & Examining Personnel for Qualified Data Analyst (QDA) Certification, Revision 0
QAP-102, Training, Qualification, and Certification of Nondestructive Testing Personnel to Qualified Data Analyst, Revision 3
QAP-9.2, QDA Qualification, Revision 10
WEC 2.10, Qualification, Training, and Certification of Nondestructive Personnel, Revision 1
WEC 2.10.1, Addendum B: Certification of EPRI Qualified Data Analyst (QDA) Personnel, Revision 1

Calculations
Calculation HNP-M/MECH-1091, Effective Degradation Years for the Reactor Vessel Head

Corrective Action Documents- Nuclear Condition Reports (NCRs)
NCR 350404, Through-wall ESW pipe leak at 1SW-1365
NCR 350617, 1SW-240 has negative stroke time closed trend
NCR 424599, White and light-brown residue on pipe cap downstream of 1SI-54
NCR 424602, White and light-brown residue found on pipe cap downstream of 1SI-133
NCR 424663, Brown residue on 1SI-251 Accumulator "B" Discharge Check Valve
NCR 428620, Possible Loose Part Identified in Steam Generator "B"

Other
Analysis Technique Specification Sheet (ANTS) CQL_ANL3CRPC_10 Rev. 0
Analysis Technique Specification Sheet (ANTS) CQL_ANL3CRPC_10 Rev. 0
Analysis Technique Specification Sheet (ANTS) CQL_ANLBOB_10 Rev. 0
Analysis Technique Specification Sheet (ANTS) CQL_ANLBOB_10 Rev. 0
Calibration Standard Specifications for Serial Nos.: Z-18932 through Z-18935, Z-18937, Z-18938, Z-19232 thorough Z-19242, Z-16965 through Z-16967, Z-19259, Z-21873, Z-16974, Z-21874 through Z-21876, Z-18946, and Z-23104
Certificate of Calibration for Eddy Current Tester Corestar Model Omni-200, Serial Numbers 221035, 224328, 224324, 224321, 224314, 221077, 221063, 221059, 221056, 221055, 221043, 221039, 221041, 221036
Certificate of Calibration for Eddy Current Tester Model MIZ-80iD, Serial Numbers 015, 021, 032, 048, 011, 043, 011, 032, 021, 015, 032, 039, 043, and 048
Certificate of Compliance for Eddy Current Probes Serial Numbers: S/N 506202, 466540, 433597, 422037, 0062-0709, 0094-0709, and 507782
Certificates of Personnel Qualification for 13 Qualified Data Analysts
Condition Monitoring Assessment for the RFO 13 – Steam Generator Eddy Current Inspection Results (April 2006) and Operational Assessment for Cycles 14, 15, 16, 08/08/06
DDM-96-009, Documentation of Appendix H Compliance and Equivalency, Revision 0 Material Qualifications for Valve 1-RC-105
MRS-TRC-1772, OMNI 200 to MIZ-70 Tester Equivalency, Revision 0
MRS-TRC-2032, Use of Appendix H & I Qualified Techniques at Shearon Harris Unit 1 16th Refueling, Revision 0
NDE Personnel Qualifications for 4 NDE Examiners

RST-207, Attachment 1, Secondary Coolant System Radiochemistry Surveillance Test Data Sheet, 9/27/10
RST-207, Attachment 1, Secondary Coolant System Radiochemistry Surveillance Test Data Sheet, 9/29/10
RST-207, Attachment 1, Secondary Coolant System Radiochemistry Surveillance Test Data Sheet, 10/1/10
Site Specific Performance Demonstration records for 13 Qualified Data Analysts
Steam Generator Degradation Assessment for Harris Nuclear Plant RFO-16 (October 2010), Revision 1, September 2010
Steam Generator Eddy Current Inspection Multi-Frequency Eddy Current Parameters, ACTS# CQL-001-10 through CQL-012-10
Westinghouse Mechanical Plugging Process Data Sheet 11.5 – Tubes SG "B" R26C133, R26C135, and R28C135, 10/22/10
Westinghouse Process Data Sheet 11.3 and 11.2 – Tubes SG "B" R26C133, R26C135, and R28C135, 10/22/10
WO 59102018991, Replaced Valves 1-RC-105 Replace 2" RE w/ Yarway dated 9/23/2010