UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
61 FORSYTH STREET, SW, SUITE 23T85
ATLANTA, GEORGIA 30303-8931

April 30, 2009

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

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT - NRC INTEGRATED INSPECTION REPORT NOS.: 05000325/2009002, 05000324/2009002, AND 07200006/2009001

Dear Mr. Waldrep:

On March 31, 2009, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Brunswick Unit 1 and 2 facilities, and the Brunswick Independent Spent Fuel Storage Installation. The enclosed integrated inspection report documents the inspection findings, which were discussed on April 17, 2009, with you and other members of your staff.

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

Based on the results of this inspection, the inspectors identified five issues of very low safety significance (Green). These issues were determined to involve violations of NRC requirements. The inspectors determined that one of the issues was a Severity Level IV violation of NRC requirements. However, because of their very low safety significance and because they have been entered into your corrective action program (CAP), the NRC is treating these issues as non-cited violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you contest any NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Brunswick Steam Electric Plant. In addition, if you disagree with the characterization of 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 Resident Inspector at the Brunswick Steam Electric Plant. The information you provide will be considered in accordance with Inspection Manual Chapter 0305.

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

Sincerely,

/RA/

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

Docket Nos.: 50-325, 50-324, 72-06
License Nos.: DPR-71, DPR-62
Enclosure: Inspection Report 05000325, 324/2009002, 07200006/2009001
w/Attachment: Supplemental Information

cc w/encl: (See page 3)

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Sincerely,

/RA/

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

Docket Nos.: 50-325, 50-32472-06
License Nos.: DPR-71, DPR-62
Enclosure: Inspection Report 05000325, 324/2009002, 07200006/2009001
w/Attachment: Supplemental Information

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Letter to Benjamin C. Waldrep from Randall A. Musser dated April 30, 2009

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT - NRC INTEGRATED INSPECTION REPORT NOS.: 05000325/2009002, 05000324/2009002, AND 07200006/2009001

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-325, 50-324, 72-06
License Nos.:	DPR-71, DPR-62
Report Nos.:	05000325/2009002, 05000324/2009002, 07200006/2009001
Licensee:	Carolina Power and Light (CP&L)
Facility:	Brunswick Steam Electric Plant, Units 1 & 2 Brunswick Independent Spent Fuel Storage Installation
Location:	8470 River Road, SE Southport, NC 28461
Dates:	January 1, 2009 through March 31, 2009
Inspectors:	P. O'Bryan, Senior Resident Inspector G. Kolcum, Resident Inspector G. Kuzo, Senior Health Physics Inspector (2OS1, 2OS2, 2PS2) A. Nielsen, Health Physics Inspector (2OS1, 2OS2, 2PS2) K. Ellis, Project Engineer (1R20) M. Corsey, Reactor Inspector (1R08) R. Chou, Reactor Inspector (4OA5) R. Jackson, Reactor Inspector (4OA5) B. Davis, Reactor Inspector Intern (4OA5)
Approved by:	Randall A. Musser, Chief Reactor Projects Branch 4 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000325/2009002, 05000324/2009002, 07200006/2009001; 01/01/09 – 03/31/09; Brunswick Steam Electric Plant, Units 1 & 2; Flood Protection Measures, Plant Modifications, Outage Activities, Identification and Resolution of Problems, Follow-up of Events.

This report covers a three-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Four Green findings and one Severity Level IV violation were identified by the inspectors. The findings and the Severity Level IV violation were considered Non-Cited Violations of NRC regulations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

Cornerstone: Barrier Integrity

- Green. The inspectors identified a Green non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action" which states in part, that for conditions adverse to quality, measures shall assure that the cause of the condition is determined and corrective action taken. Specifically, the licensee failed to correct a condition that allowed leakage through a penetration seal in the Unit 1 reactor building supply air ventilation room floor onto the 1B standby gas treatment (SBGT) train control panel, rendering the 1B SBGT inoperable. The licensee entered the issue into their corrective action program and repaired the degraded penetration.

  The deficiency associated with this event is not adequately sealing the floor penetration in the Unit 1 reactor building supply air ventilation room. The finding is more than minor because it was associated with the containment barrier performance attribute of the barrier integrity cornerstone to provide reasonable assurance that physical design barriers provide protection against radionuclide releases caused by accidents or events. In accordance with Inspection Manual Chapter (IMC) 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspectors conducted a Phase I significance determination process (SDP) screening and determined the finding to be of very low safety significance (Green). The finding was of very low safety significance (Green) because the finding only represents a degradation of the radiological barrier function provided for the SBGT system. The cause of the finding is not related to a cross-cutting aspect because the occurrence was greater than three years ago and is not indicative of current licensee performance. (Section 1R06)

- Green. A self-revealing Green NCV of Technical Specification (TS) 5.4.1, Procedures, was identified for inadequate maintenance procedures for the control room air conditioning and emergency ventilation system instrument air dryer. As a result, on January 21, 2009, the control room air conditioning and emergency ventilation instrument air system lost air pressure, rendering the control room air conditioning (AC) system and the control room emergency ventilation (CREV) system inoperable. The licensee entered the issue into their corrective action program and changed maintenance and operating procedures to prevent recurrence.

The failure to implement adequate maintenance procedures for the control room air conditioning and emergency ventilation instrument air system is a performance deficiency. This performance deficiency is more than minor because it is associated with structure, system, and component (SSC), and barrier performance attribute of the Barrier Integrity Cornerstone. It also adversely affected the cornerstone objective of maintaining a radiological barrier for the control room. The finding was determined to be of very low safety significance because it only affected the radiological barrier function of the control room, and does not represent a degradation of the smoke or toxic atmosphere barrier function of the control room. The cause of the finding is related to the cross-cutting area of human performance, resources component, complete and accurate documentation aspect, because the licensee did not incorporate adequate guidance for maintaining the control room AC and CREV instrument air dryer in their maintenance procedures. (H.2(c)) (Section 4OA2)

Cornerstone: Mitigating Systems

- <u>Severity Level IV</u>. The inspectors identified a severity level IV NCV of 10 CFR 50.59, "Changes, Tests, and Experiments" for failing to perform a written safety evaluation prior to implementing a change to the facility as described in the Updated Final Safety Analysis Report (UFSAR), when the Unit 1 and Unit 2 reactor building instrument air standby compressors were permanently abandoned. The licensee entered the issue into their corrective action program and performed a written safety evaluation of the condition.

  The inspectors determined that, until identified by NRC inspectors, the licensee had not performed a 10 CFR 50.59 safety evaluation for the abandonment of the instrument air standby compressors, and this is a performance deficiency. Because this is a violation of 10 CFR 50.59, it is considered to be a violation which potentially impedes or impacts the regulatory process. Therefore, such violations are dispositioned using the traditional enforcement process instead of the Significance Determination Process. This finding was determined to be more than minor because their was a reasonable likelihood that the change requiring a 10 CFR 50.59 safety evaluation would require Commission review and approval prior to implementation in accordance with 10 CFR 50.59(c)(2). This likelihood is based on the increased likelihood of loss of reactor building instrument air, reactor scram, and closure of the outboard MSIVs, which is an occurrence of a malfunction of a structure, system, or component (SSC) that is analyzed in the UFSAR. To determine the significance of the violation, the inspectors completed a significance determination review using IMC 0609, Appendix A, Significance Determination of Reactor Inspection Findings for At Power Situations. The finding impacted the initiating events cornerstone. Because the finding does not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available, this finding has very low safety significance. The cause of the finding is not related to a cross-cutting aspect because the performance deficiency is not indicative of current licensee performance. (Section 1R18)

- Green. A self-revealing Green NCV of TS 5.4.1, Procedures, was identified when reactor head piping was disconnected prior to swapping shutdown range reactor water level transmitters resulting in inaccurate water level indication. The plant procedure for disconnection of the reactor head piping, 0SMP-RPV501, Reactor Vessel Disassembly, used in conjunction with 0GP-06, Cold Shutdown to Refueling, specifies that prior to removal of head piping, the Shutdown Range Reactor Water Level Transmitters shall be swapped from level transmitters, B21-LT-NO27A and B21-LT-NO27B, to level transmitters, B21-LT-7468A and B21-LT-7468B. Contrary to this requirement, the common reference leg to the level indicators was disconnected prior to swapping transmitters which resulted in loss of accurate indication of current reactor vessel water level. The licensee reinstalled the disconnected piping and entered the issue into their corrective action program.

  The disconnection of the reference leg flange of the reactor vessel head piping prior to realignment of level instrumentation as required per procedure was identified as a performance deficiency. The performance deficiency was more than minor because it is associated with the configuration control 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. The level indication inaccuracy degraded the plant operators' ability to control the reactor vessel water level in the prescribed procedural band and would inhibit their ability to diagnose and prevent a Loss of RHR scenario. In accordance with NRC Inspection Manual Chapter (IMC) 0609, Appendix G, "Shutdown Operations Significance Determination Process," Attachment 1, Checklist 8, the inspectors conducted a Phase 1 SDP screening and determined the finding to require a Phase 2 analysis. The Phase 2 analysis determined the finding to be of very low safety significance (Green) because adequate mitigation capability was maintained. The cause of this finding was directly related to the work activity coordination cross-cutting aspect in the work control component of the Human Performance cross-cutting area because the plant operators and maintenance personnel failed to effectively communicate and coordinate the activities associated with the vessel head disassembly (H.3(b)). (Section 1R20)

- Green. A self-revealing Green NCV of TS 5.4.1, Procedures, was identified when the licensee changed the position of 2-E11-F009, the shutdown cooling (SDC) inboard suction throttle valve, without following a procedure. On March 26, Unit 2 was in Mode 5 in a refueling outage with the reactor refueling cavity flooded and fuel pool gates removed. Decay heat removal was being provided by protected systems, RHR loop B and supplemental spent fuel pool cooling. ADM-NGGC-0104, Work Management Process, states the maintenance that has an impact on system operation must be performed according to written instructions. Contrary to this requirement, a maintenance technician working without written instructions, operated the 2-E11-F009 valve locally in the drywell in the close direction, tripping the only operating RHR pump due to an electrical interlock. The licensee restored the RHR system to operation and entered the issue into their corrective action program.

  The operation of 2-E11-F009, the shutdown cooling (SDC) inboard suction throttle valve, during a maintenance activity was identified as a performance deficiency. The finding is more than minor because it affects the human performance attribute of the Initiating

Events cornerstone and the objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The inspectors evaluated this finding using Attachment 1 of IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process." This finding is of very low safety significance because the finding did not represent a loss of control and did not require quantitative assessment per Checklist 7 of Attachment 1 to IMC 0609, Appendix G. Specifically, the reactor time-to-boil during this event was approximately 36 hours and RHR was restored in 17 minutes. Additionally, during the time that RHR was secured, the supplemental spent fuel pool cooling system provided sufficient decay heat removal. 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 maintenance supervision and the maintenance technician failed to follow the station's policy for work on protected train equipment and use the human error prevention tools associated with the protected train concept. (H.4(a)). (Section 1R20)

REPORT DETAILS

Summary of Plant Status

Unit 1 began the inspection period at rated thermal power. Power was reduced to 65% on February 5, to perform maintenance on the 1A reactor feed pump. Unit 1 returned to rated thermal power on February 8 after a control rod improvement. Unit 1 power was reduced to 65% on March 7 due to control rod improvements. Power was raised to 76% on March 8 to perform leak repairs on the level control valve for the west 2nd stage reheater drain tank. Maintenance was deferred due to interference problems on the valve and Unit 1 returned to rated thermal power on March 9. Unit 1 operated at or near full power for the remainder of the inspection period.

Unit 2 began the inspection period at rated thermal power. On January 11, Unit 2 reduced power to 89% for a rod pattern adjustment and then returned to rated thermal power. On January 16, Unit 2 reduced power to 89% for a rod pattern adjustment and then returned to rated thermal power. On January 20, Unit 2 reduced power to 88% for a rod pattern adjustment and then returned to rated thermal power. On January 23, Unit 2 reduced power to 65% for testing and rod pattern adjustment and then returned to rated thermal power on January 25. On January 25, Unit 2 reduced power to 93% for a rod pattern adjustment and then returned to rated thermal power. On January 26, Unit 2 reduced power to 96% for a rod pattern adjustment and then returned to rated thermal power. On February 1, Unit 2 reduced power to 94% for a rod pattern adjustment and then returned to rated thermal power. On February 4, Unit 2 reduced power to 95% for a rod pattern adjustment and then returned to rated thermal power. On February 8, Unit 2 reduced power to 75% for a rod pattern adjustment and then returned to rated thermal power. On February 15, Unit 2 reduced power to 75% for a rod pattern adjustment and then returned to rated thermal power. On February 22, Unit 2 reduced power to 75% for a rod pattern adjustment and then returned to rated thermal power. On February 24, Unit 2 reduced power to 95% due to failure of #4 turbine control valve. On February 27, a power reduction was commenced in preparation for a refueling outage. On February 28, Unit 2 was shut down for B219R1 refueling and remained in that condition through the end of the report period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection

.1 Extreme Cold Weather Conditions

a. Inspection Scope

The inspectors conducted a review of the licensee's preparations for extreme cold weather conditions prior to the site experiencing unusually cold temperatures on January 15, 2009 to verify that the plant's design features and implementation of procedures were sufficient to protect mitigating systems from the effects of extreme cold weather.

During the inspection, the inspectors focused on plant specific design features and the licensee's procedures used to mitigate or respond to cold weather conditions. Cold weather protection, such as heat tracing and area heaters, was verified to be in operation where applicable. The inspectors also reviewed corrective action program items to verify that the licensee was identifying adverse weather issues at an appropriate threshold and entering them into their corrective action program in accordance with station corrective action procedures. Specific documents reviewed during this inspection are listed in the Attachment.

b. Findings

No findings of significance 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 Updated Final Safety Analysis Report (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, checked that the roofs did not contain obvious loose items that could clog drains in the event of heavy precipitation, and determined 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 performed a walkdown of the vent lines associated with the EDG fuel oil storage tanks for their potential to be susceptible to flooding. The inspectors also reviewed the abnormal operating procedure (AOP) for mitigating the design basis flood to ensure it could be implemented as written.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Quarterly Partial System Walkdowns

a. Inspection Scope

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

- 1A loop of the residual heat removal system with the 1B loop out of service on January 22, 2009.

- 2B standby liquid control system with the 2A standby liquid control system out of service on February 10, 2009.
- 2B loop of the residual heat removal system with the 2A loop of the residual heat removal system out of service on February 17, 2009.
- Unit 2 supplemental spent fuel pool cooling system with the A and B residual heat removal systems out of service during refueling on March 20, 2009.
- 2A loop of the residual heat removal system with the 2B loop of the residual heat removal system out of service on March 24, 2009.

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, Updated Final Safety Analysis Report (UFSAR), Technical Specification (TS) requirements, outstanding work orders, condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also walked down accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the corrective action program with the appropriate significance characterization. Documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2 Semi-Annual Complete System Walkdown

a. Inspection Scope

The inspectors performed a complete system alignment inspection of the Unit 2 residual heat removal system to verify the functional capability of the system. This system was selected because it was considered both safety-significant and 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, as appropriate, component labeling, component lubrication, component and equipment cooling, hangers and supports, operability of support systems, and to ensure that ancillary equipment or debris did not interfere with equipment operation. A review of a sample of past and outstanding work orders (WOs) was performed to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the corrective action program (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.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Quarterly Resident Inspector Tours

a. Inspection Scope

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

- Unit 1 Reactor Building East 50' Elevation 1PFP-RB1-1h E
- Unit 1 Reactor Building West 50' Elevation 1PFP-RB1-1h W
- Unit 2 North RHR Room -17' Elevation 2PFP-RB2-1c
- Unit 2 South RHR Room -17' Elevation 2PFP-RB2-1d
- Unit 2 HPCI Room -17' Elevation 2PFP-RB2-2
- Unit 2 Turbine/MSR Area 70' Elevation 2PFP-TB2-1n

The inspectors reviewed areas to assess if the licensee had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant, effectively maintained fire detection and suppression capability, maintained passive fire protection features in good material condition, and had implemented adequate compensatory measures for out of service, degraded or inoperable fire protection equipment, systems, or features in accordance with the licensee’s fire plan. The inspectors selected fire areas based on their overall contribution to internal fire risk as documented in the plant’s Individual Plant Examination of External Events 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 corrective action program.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

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. The specific documents reviewed are listed in the attachment. 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 corrective action program to verify the adequacy of the corrective actions. The inspectors performed a walkdown of the following plant area(s) 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:

- Unit 1 standby gas treatment room 50' level and reactor building ventilation room 80' level

b. Findings

Failure to Identify and Correct a Condition Adverse to Quality

Introduction. The inspectors identified a Green non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action" which states in part, that for conditions adverse to quality, measures shall assure the condition be promptly identified and corrected. Specifically, the licensee failed to correct a condition that allowed leakage through a penetration seal in the Unit 1 reactor building supply air ventilation room floor onto the 1B standby gas treatment (SBGT) train control panel, rendering the 1B SBGT inoperable.

Description. On January 17, 2009, a fire protection system valve located in the Unit 1 reactor building supply air ventilation room froze and cracked. Water from the fire protection system accumulated on the floor of the ventilation room and leaked through the ventilation room floor penetration seal. The water entered the 50' elevation of the reactor building and wetted the 1B SBGT train control panel and created an electrical ground, disabling the 1B SBGT controls.

Similar leakage was noted twice previously at this penetration. On June 28, 2000, standing water in the Unit 1 reactor building supply ventilation room caused leakage through the floor penetration and caused spurious alarms on 1B SBGT train from water dripping onto the 1B SBGT panel. Licensee corrective action documents noted that the leaking drain required repair, but there were no corrective action items put in place for the repairs. On July 23, 2004, water again leaked onto 1B SBGT control panel when a floor drain became clogged in the reactor building supply air ventilation room preventing proper draining. The water backed up and began to drain through a hole in the pipe located at the penetration. These two previous occurrences of water leakage through the floor penetration demonstrated that an adverse condition existed affecting the 1B SBGT system. However, the licensee did not implement adequate corrective actions. Previous licensee corrective actions to seal the penetration concentrated on maintaining the secondary containment boundary, and did not adequately address the ability of the penetration to prevent the leakage of water onto the 1B SBGT system.

Analysis. The deficiency associated with this event is not adequately sealing the floor penetration in the Unit 1 reactor building supply air ventilation room. The finding is more than minor because it was associated with the containment barrier performance attribute of the barrier integrity cornerstone to provide reasonable assurance that physical design barriers provide protection against radionuclide releases caused by accidents or events. In accordance with Inspection Manual Chapter (IMC) 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspectors conducted a Phase I significance determination process (SDP) screening and determined the finding to be of very low safety significance (Green). The finding was of very low safety significance (Green) because the finding only represents a degradation of the radiological barrier function provided for the SBGT system. The cause of the finding is not related to a cross-cutting aspect because the occurrence was greater than three years ago and is not indicative of current licensee performance.

Enforcement. 10 CFR 50, Appendix B, Criterion XVI requires, in part, that conditions adverse to quality, such as equipment deficiencies, be promptly identified and corrected. Contrary to the above, the licensee did not implement corrective actions to correct the leaking seal around the pipe penetration in the Unit 1 reactor building ventilation room. Water leaked through the penetration and onto the 1B SBGT train control panel, rendering it inoperable. Because this finding is of very low safety significance and has been entered into the licensee's corrective action program (NCR 315118), this violation is being treated as an NCV, consistent with the NRC Enforcement Policy and is designated as NCV 05000325/2009002-01, Failure to Identify and Correct a Condition Adverse to Quality Affecting the Operability of the Standby Gas Treatment Train B.

1R07 Heat Sink Performance

a. Inspection Scope

The inspectors performed a visual inspection of the 2B residual heat removal system heat exchanger and reviewed the licensee's eddy current testing results to verify that potential deficiencies did not exist that would lead to 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 also reviewed the licensee's observations as compared against acceptance criteria, the frequency of testing, and the impact of instrument inaccuracies on test results.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities

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

a. Inspection Scope

From March 09-13, 2009, the inspector observed and 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 of Brunswick Unit 2 during refueling outage B219R1. The inspector activities consisted of an on-site review of nondestructive examination (NDE) and welding activities to evaluate compliance with Technical Specifications and the applicable edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Sections XI (Code of record: 2001 Edition through the 2003 Addenda), for Class 1, 2, and 3 systems; and to verify that indications and defects (if present) were appropriately evaluated and dispositioned in accordance with the requirements of the Section XI acceptance standards. For Brunswick Unit 2 this was the third outage of the last period of the third interval and first outage of the first period of the fourth interval due to a Relief Request issued pursuant to RR-41 issued by the NRC on December 18, 2008. The third period of the third interval ends on May 10, 2009 in accordance with the above referenced Relief Request. The inspector also reviewed a sample of inspection activities associated with components that are outside the scope of ASME Section XI requirements which are performed in accordance with commitments to follow industry guidance documents, such as the Boiling Water Reactor Vessel and Internals Project (BWRVIP).

The inspector reviewed NDE activities, specifically including examination procedures, NDE reports, equipment and consumables certification records, personnel qualification records, and calibration reports for compliance to requirements of ASME Section V, ASME Section XI, BWRVIP documents, and other industry standards for the following examinations:

- VT-3 of Component 2-B32-PS7505A Visual Examination of Component Supports and Snubbers
- UT of Component 2E1163-21-SWC
- UT of Component 2B32RECIRC-28-B-8

The inspector review of welding activities specifically covered the welding activity listed below in order to evaluate compliance with procedures and the ASME Code.

The inspector reviewed the work order, repair and replacement plan, weld data sheets, welding procedures, procedure qualification records, welder qualification records, and NDE reports.

- Work Order 01382250 2-B21-F028C PERFORM GUIDE PAD MODIFICATION

b. Findings

No findings of significance were identified.

.2 Reactor Vessel Internal Inspections

a. Inspection Scope

The inspector reviewed the following NDE activities associated with the inspection of Reactor Vessel internal components (Boiling Water Reactors Vessel Internals Project):

- EVT-1 of the core shroud particularly of the V2 and V6 welds and indications.
- EVT-1 of the Core Spray T-Box at 90 degrees.
- EVT-1 of the Jet Pump Riser C RS-1 weld.

b. Findings

No findings of significance were identified.

.3 Identification and Resolution of Problems

a. Inspection Scope

The inspector completed a review of ISI-related problems that were identified by the licensee and entered into the corrective action program. The inspector reviewed these corrective action documents to confirm that the licensee had appropriately described the scope of the problems, and had implemented appropriate corrective actions.

The inspector review included confirmation that the licensee had an adequate threshold for identifying issues. Through interviews with licensee staff and review of corrective action documents, the inspector evaluated the licensee's threshold for identifying lessons learned from industry issues related to ASME Section XI. The inspector performed these reviews to ensure compliance with 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. The corrective action documents reviewed by the inspector are listed in the report Attachment.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program

a. Inspection Scope

On February 4, 2009, the inspectors observed crew D of licensed operators in the plant's simulator during licensed operator requalification examinations to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems, and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

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

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

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

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

- Unit 2 high pressure core injection system inoperable due to a failed barometric condenser pump on January 27, 2009.
- Supplemental spent fuel pool cooling system spurious control circuit actuations on March 14, 2009 and March 16, 2009.

The inspectors reviewed events where ineffective equipment maintenance has resulted in invalid automatic actuations of protective features or equipment malfunctions and independently verified the licensee's actions to address system performance or condition problems in terms of the following:

- implementing appropriate work practices;
- identifying and addressing common cause failures;
- scoping of systems in accordance with 10 CFR 50.65(b) of the maintenance rule;
- characterizing system reliability issues for performance;
- charging unavailability for performance;
- trending key parameters for condition monitoring;
- ensuring 10 CFR 50.65(a)(1) or (a)(2) classification or re-classification; and
- verifying appropriate performance criteria for 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 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 corrective action program with the appropriate significance characterization. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

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

- Maintenance scheduled and conducted during the week of January 5, 2009.
- Yellow risk condition while the 1B loop of RHR was out of service on January 22, 2009
- Unit 2 A loop of residual heat removal service water system scheduled maintenance which rendered secondary containment inoperable on February 17, 2009.
- Planned maintenance on the Unit 2 nuclear service water header on March 5, 2009.
- Spurious trip of the supplemental spent fuel pool cooling system while employing natural circulation decay removal on March 16, 2009.

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 of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following issues:

- NCR #312876, Short Duration LCO Entry on EDG #3 and #4, January 1, 2009
- NCR #315863, Instrument 1-B21-XY-5949A Self Test Fault, Steam Leak Detection on January 23, 2009.
- NCR #316695, Unit 2 HPCI Barometric Condenser Pump Failure on January 27, 2009.
- NCR #321193, #3 Emergency Diesel Generator Sheared Air Start Distributor Pin on February 23, 2009.
- NCR #322413, #4 Emergency Diesel Generator Low Jacket Water Differential Temperature on March 2, 2009.

The inspectors selected these potential operability issues based on the risk-significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure that TS operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the Technical Specifications (TS) and Updated Safety Analysis Report (USAR) 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.

b. Findings

No findings of significance were identified.

1R18 Plant Modifications

a. Inspection Scope

The following modifications were reviewed and selected aspects were discussed with engineering personnel:

- Abandonment of the reactor building standby instrument air compressors (permanent modification).
- Supplemental spent fuel pool cooling secondary pump protective features removed from operation (temporary modification).

This document and related documentation were reviewed for adequacy of the associated 10 CFR 50.59 safety evaluation screening, consideration of design parameters, implementation of the modification, post-modification testing, and relevant procedures, design, and licensing documents were properly updated. The inspectors observed ongoing and completed work activities to verify that installation was consistent with the design control documents.

b. Findings

Failure to Perform a 10 CFR 50.59 Evaluation for a Plant Modification

Introduction: The inspectors identified a Severity Level IV non-cited violation of 10 CFR 50.59, "Changes, Tests, and Experiments" for failing to perform a written safety evaluation prior to implementing a change to the facility as described in the Updated Final Safety Analysis Report (UFSAR), when the Unit 1 and Unit 2 reactor building instrument air standby compressors were permanently abandoned.

Description: The licensee's UFSAR describes the function of the four (two in each unit) reactor building instrument air standby compressors as providing an air make-up supply to the reactor non-interruptible air (RNA) system when system pressure lowers to 95 pounds per square inch. The RNA system provides air for various functions, including maintaining outboard main steam isolation valves (MSIVs) open and maintaining control rod drive scram valves shut.

The licensee's nuclear condition report (NCR) #75947, written in 2003, describes the history of the reactor building instrument air standby compressors up to that time and this history is useful in understanding the events leading to this violation. The instrument air standby compressors were installed in 1975 due to problems maintaining sufficient reactor building instrument air header pressure during equipment operation. In 1984, the capacity of the compressors was increased. However, after the compressor capacity was increased, the licensee identified that air filters and dryers could not operate correctly at the higher discharge temperature and flow. Therefore, the licensee downgraded the compressors to non-safety and a backup nitrogen system was installed as a backup to the RNA system to operate safety-related components that did not fail to their safe positions. After 1984, the compressors were operated with the filters and dryers bypassed. In 1991, regular performance tests of the compressors were deleted due to concern about charging air that was not filtered or dried into the RNA system. Sometime around 2000 (the exact date is unknown), the reactor building instrument air standby compressors were run for the last time.

The 2003 NCR #75947 further concluded that the instrument air standby compressors were not capable of meeting their design basis function because they had not been operated "for at least 3 years" and were therefore unreliable. In 2005, a 10 CFR 50.59 screening was completed to determine if a written 10 CFR 50.59 safety evaluation should be performed for the permanent abandonment of the instrument air standby compressors. This screening determined that the abandonment of the compressors did not adversely affect a UFSAR-described design function. Inspectors determined that the UFSAR-described design function of the instrument air standby compressors was adversely affected by their abandonment and could not conclude that a license amendment would not be required for the change since the change increases the likelihood of loss of reactor building air. Loss of reactor building air would result in a reactor scram and closure of the outboard MSIVs, which is an occurrence of a malfunction of a structure, system, or component (SSC) that is analyzed in the UFSAR.

Analysis: The inspectors determined that, until identified by NRC inspectors, the licensee had not performed a 10 CFR 50.59 safety evaluation for the abandonment of the instrument air standby compressors, and this is a performance deficiency. Because this is a violation of 10 CFR 50.59, it is considered to be a violation which potentially impedes or impacts the regulatory process. Therefore, such violations are dispositioned using the traditional enforcement process instead of the Significance Determination Process. This finding was determined to be more than minor because their was a reasonable likelihood that the change requiring a 10 CFR 50.59 safety evaluation would require Commission review and approval prior to implementation in accordance with 10 CFR 50.59(c)(2). This likelihood is based on the increased likelihood of loss of reactor building instrument air, reactor scram, and closure of the outboard MSIVs, which is an occurrence of a malfunction of a structure, system, or component (SSC) that is analyzed in the UFSAR. To determine the significance of the violation, the inspectors completed a significance determination review using IMC 0609, Appendix A, Significance Determination of Reactor Inspection Findings for At Power Situations. The finding impacted the initiating events cornerstone. Because the finding does not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available, this finding has very low safety significance. The cause of the finding is not related to a cross-cutting aspect because the performance deficiency is not indicative of current licensee performance.

Enforcement: 10 CFR 50.59(d)(1) requires, in part, that licensees maintain records of changes in the facility, of changes in procedures, and of tests and experiments. These records must include a written evaluation which provides the basis for determination that the change, test, or experiment does not require a license amendment. Contrary to the above, the licensee failed to perform a written safety evaluation prior to making a change to the facility as described in the UFSAR. Specifically, the licensee abandoned the reactor building instrument air standby compressors, which are described in section 9.3.1.2.1 of the UFSAR. The failure to perform a written safety evaluation was characterized as a severity level IV violation. This issue is in the licensee's corrective action program as NCR #316630. Because this violation was of very low safety significance, was not repetitive or willful, and was entered into the licensee's corrective action program, this violation is being treated as an NCV and is designated as NCV 05000325,324/2009002-02, Failure to Perform a 10 CFR 50.59 Evaluation for a Plant Modification.

1R19 Post Maintenance Testing

a. Inspection Scope

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

- 0PT-12.2D, #4 Diesel Generator Monthly Load Test to monitor the performance of the fuel rack limiting cylinder on January 5, 2009

- 2OP-19, HPCI system Operating Procedure, to verify proper operation of the HPCI barometric condenser condensate pump after replacement of brushes on January 28, 2009
- 2PT-08.2.2C, LPCI/RHR Operability Test, Loop A after planned maintenance on February 18, 2009
- 0OP-13.1, Supplemental Spent Fuel Pool Cooling Operating Procedure, after temporary modification to remove automatic protective features on March 19, 2009
- 0PT-08.2.2b, LPCI/RHR System Operability Test – Loop B after planned maintenance on March 24, 2009

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 corrective action program and that the problems were being corrected commensurate with their importance to safety. Documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

1R20 Outage Activities

.1 Refueling Outage Activities

a. Inspection Scope

The inspectors reviewed the outage plan and contingency plans for the Unit 2 refueling outage, which commenced on February 28, 2009 to confirm that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. During the refueling outage, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below. The inspectors also reviewed the loss of 2B RHR shutdown cooling and orange risk condition on March 26, 2009. Documents reviewed during the inspection are listed in the attachment.

- Licensee configuration management, including maintenance of defense-in-depth commensurate with the OSP for key safety functions and compliance with the applicable TS when taking equipment out of service.
- Implementation of clearance activities and confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing.
- Installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication, accounting for instrument error.
- Controls over the status and configuration of electrical systems to ensure that TS and outage safety plan requirements were met, and controls over switchyard activities.
- Monitoring of decay heat removal processes, systems, and components.
- Controls to ensure that outage work was not impacting the ability of the operators to operate the spent fuel pool cooling system.
- Reactor water inventory controls including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss.
- Controls over activities that could affect reactivity.
- Maintenance of secondary containment as required by TS.
- Refueling activities, including fuel handling and sipping to detect fuel assembly leakage.
- Licensee identification and resolution of problems related to refueling outage activities.

b. Findings

(1) Loss of Accurate Reactor Level Indication

Introduction: A self-revealing Green NCV of TS 5.4.1, Procedures, was identified when reactor head piping was disconnected prior to swapping shutdown range reactor water level transmitters resulting in inaccurate water level indication. The plant procedure for disconnection of the reactor head piping, 0SMP-RPV501, Reactor Vessel Disassembly, used in conjunction with 0GP-06, Cold Shutdown to Refueling, specifies that prior to removal of head piping, the Shutdown Range Reactor Water Level Transmitters shall be swapped from level transmitters, B21-LT-NO27A and B21-LT-NO27B, to level transmitters, B21-LT-7468A and B21-LT-7468B. Contrary to this requirement, the common reference leg to the level indicators was disconnected prior to swapping transmitters which resulted in loss of accurate indication of current reactor vessel water level.

Description: On March 2, 2009, with Unit 2 in Mode 4, reactor water level was being maintained between 400 and 420 inches, as indicated on the Shutdown Range Reactor Water Level Transmitters (B21-LT-NO27A and B21-LT-NO27B). The indicated water level suddenly increased from approximately 410 to 470 inches. In response, the plant operators lowered water level 10 inches, according to the transmitters now indicating 470 inches, to verify positive control of the parameter. The control room was then notified that the reference leg flange of the level transmitters currently being used had been loosened. When the flange was loosened, the reference leg lost water which resulted in the rise in indicated water level. The water level indicators were realigned from level transmitters, B21-LT-NO27A and B21-LT-NO27B, to level transmitters, B21-LT-7468A and B21-LT-7468B. Upon realignment, the water level initially indicated 360 inches, and after several minutes, the indicators settled out at 390 and 400 inches. The initial reading of 360 inches was the result of a vacuum above the water in the vessel that was caused by the operator action to lower the water level 10 inches. However, because the vessel was vented, the indicated level rose when the vessel's pressure equalized to atmosphere.

Procedure 0GP-06, "Cold Shutdown to Refueling", step 5.1.14, states that the level transmitters shall be swapped from level transmitters, B21-LT-NO27A and B21-LT-NO27B, to level transmitters, B21-LT-7468A and B21-LT-7468B, and authorization shall be given prior to removing reactor water level piping per procedure 0SMP-RPV501. The miscommunication between the control room operators and the maintenance group performing the disassembly of the reactor vessel head piping lead to loosening the flange of the common reference leg prior to the control room operators swapping from the level transmitters. This procedure violation caused inaccurate reactor water level indication for approximately 15 minutes.

Analysis: The disconnection of the reference leg flange of the reactor vessel head piping prior to realignment of level instrumentation as required per procedure was identified as a performance deficiency. The performance deficiency was more than minor because it is associated with the configuration control 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. The level indication inaccuracy degraded the plant operators' ability to control the reactor vessel water level in the prescribed procedural band and would inhibit their ability to diagnose and prevent a Loss of RHR scenario. In accordance with NRC Inspection Manual Chapter (IMC) 0609, Appendix G, "Shutdown Operations Significance Determination Process," Attachment 1, Checklist 8, the inspectors conducted a Phase 1 SDP screening and determined the finding to require a Phase 2 analysis. The Phase 2 analysis determined the finding to be of very low safety significance (Green) because adequate mitigation capability was maintained. The cause of this finding was directly related to the work activity coordination cross-cutting aspect in the work control component of the Human Performance cross-cutting area because the plant operators and maintenance personnel failed to effectively communicate and coordinate the activities associated with the vessel head disassembly (H.3(b)).

Enforcement: TS 5.4.1, Administrative Control (Procedures), requires that written procedures shall be established, implemented, and maintained, covering applicable procedures recommended in Regulatory Guide 1.33, Appendix A, November 1972 (Safety Guide 33, November 1972). Section I.1 of Regulatory Guide 1.33, Appendix A, November 1972 (Safety Guide 33, November 1972) states that maintenance that can affect the performance of safety-related equipment should be properly planned and performed in accordance with written procedures, documented instructions, or drawings appropriate to the circumstances. In addition, Section B.10 of Regulatory Guide 1.33, Appendix A, November 1972 (Safety Guide 33, November 1972) states that procedures used in preparation for refueling, refueling equipment operation, and core alterations should be covered by written procedures. Contrary to the above, the licensee failed to follow the maintenance and integrated operations procedures which require the operators to swap level transmitters prior to mechanics removal of reactor water level piping. The removal of the reference leg piping, prior to swapping level transmitters, resulted in inaccurate Unit 2 reactor water level indication for approximately 15 minutes on March 2, 2009. Because the finding is of very low safety significance and has been entered into the CAP (NCR 322354), and consistent with the NRC Enforcement Policy, this violation is being treated as a non-cited violation, and is designated as NCV 05000324/2009002-03, Failure to Follow Procedures During Reactor Head Disassembly.

(2) Failure to Follow Plant Procedures for Operation of Shutdown Cooling Inboard Suction Throttle Valve

Introduction: A self-revealing Green NCV of TS 5.4.1, Procedures, was identified when the licensee changed the position of 2-E11-F009, the shutdown cooling (SDC) inboard suction throttle valve, without following a procedure. On March 26, Unit 2 was in Mode 5 in a refueling outage with the reactor refueling cavity flooded and fuel pool gates removed. Decay heat removal was being provided by protected systems, RHR loop B and supplemental spent fuel pool cooling. ADM-NGGC-0104, Work Management Process, states the maintenance that has an impact on system operation must be performed according to written instructions. Contrary to this requirement, a maintenance technician working without written instructions, operated the 2-E11-F009 valve locally in the drywell in the close direction, tripping the only operating RHR pump due to an electrical interlock.

Description: On March 26, 2009, Unit 2 was in Mode 5 in a refueling outage with the reactor refueling cavity flooded and fuel pool gates removed. Decay heat removal was being provided by protected systems, RHR loop B and the supplemental spent fuel pool cooling. A maintenance technician was tasked with replacing a thread protector on motor-operated valve 2-E11-F009, the inboard suction valve for the RHR system in the shutdown cooling mode. Although this valve was a protected piece of equipment, the technician's supervisor viewed the thread protector replacement as a non-intrusive task and directed the technician to perform the task. While replacing the thread protector, the technician attempted to ensure the valve was fully open by manually operating the valve. The technician did not have permission from control room licensed operators to manipulate the valve, and doing so is in violation of station policy for operating protected equipment. In performing this task, he inadvertently operated the valve in the close direction which tripped the operating RHR 'D' pump due to an electrical interlock. Operators took prompt actions to restore shutdown cooling flow. RHR shutdown cooling flow was secured for 17 minutes. However, reactor coolant temperature remained at 99°F before and after the event because the supplemental spent fuel pool cooling system was in operation with sufficient capacity to remove decay heat from the core and from the spent fuel pool.

Analysis: The operation of 2-E11-F009, the shutdown cooling (SDC) inboard suction throttle valve, during a maintenance activity was identified as a performance deficiency. The finding is more than minor because it affects the human performance attribute of the Initiating Events cornerstone and the objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The inspectors evaluated this finding using Attachment 1 of IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process." This finding is of very low safety significance because the finding did not represent a loss of control and did not require quantitative assessment per Checklist 7 of Attachment 1 to IMC 0609, Appendix G. Specifically, the reactor time-to-boil during this event was approximately 36 hours and RHR was restored in 17 minutes. Additionally, during the time that RHR was secured, the supplemental spent fuel pool cooling system provided sufficient decay heat removal. 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 maintenance supervision and the maintenance technician failed to follow the station's policy for work on protected train equipment and use the human error prevention tools associated with the protected train concept. (H.4(a))

Enforcement: TS 5.4.1, Administrative Control (Procedures), requires that written procedures shall be established, implemented, and maintained, covering applicable procedures recommended in Regulatory Guide 1.33, Appendix A, November 1972 (Safety Guide 33, November 1972). Section I.1 of Regulatory Guide 1.33, Appendix A, November 1972 (Safety Guide 33, November 1972) states that maintenance that can affect the performance of safety-related equipment should be properly planned and performed in accordance with written procedures, documented instructions, or drawings appropriate to the circumstances. Contrary to the above, on March 26, 2009, a maintenance technician operated a valve in the residual heat removal system without written instructions. Specifically, the licensee operated 2-E11-F009, the shutdown cooling (SDC) inboard suction throttle valve, locally in the drywell in the close direction which tripped the operating RHR 'D' pump. Because the finding is of very low safety significance and has been entered into the CAP (AR 327475), and consistent with the NRC Enforcement Policy, this violation is being treated as a non-cited violation, and is designated as NCV 05000324/2009002-04, Unauthorized Maintenance Results in Loss of Shutdown Cooling.

1R22 Surveillance Testing

.1 Routine Surveillance Testing

a. Inspection Scope

The inspectors either observed surveillance tests or reviewed the test results for the following activities to verify the tests met TS surveillance requirements, UFSAR commitments, 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.

- 2MST-AMI27M, Suppression Pool Temp Monitoring Channel Functional on January 1, 2009
- 0PT-12.2D, #4 Diesel Generator Monthly Load Test on January 5, 2009
- 0PT-08.1.6, Suppression Pool Level Indicator Operability on February 3, 2009

b. Findings

No findings of significance were identified.

.2 In service Testing (IST) Surveillance

a. Inspection Scope

The inspectors reviewed the performance of 0PT-06.1, Unit 2 Standby Liquid Control System Operability Test on January 9, 2009, to evaluate the effectiveness of the licensee's American Society of Mechanical Engineers (ASME) Section XI testing program for determining equipment availability and reliability. The inspectors evaluated selected portions of the following areas: 1) testing procedures, 2) acceptance criteria, 3) testing methods, 4) compliance with the licensee's IST program, TS, selected licensee commitments, and code requirements, 5) range and accuracy of test instruments, and 6) required corrective actions.

b. Findings

No findings of significance were identified.

.3 Containment Isolation Valve Testing

a. Inspection Scope

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:

- 0PT-20.3, Local Leakrate Testing, on Unit 2 Tip Ball Valves, 2-C51-J004A, B, C, D-BAL-VLV, and Tip Nitrogen Purge Line Check Valve, 2-C51-TIP-CHV on March 10, 2009.

The inspectors observed in-plant activities and reviewed procedures and associated records to determine whether: any preconditioning occurred; 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 TSs, 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 and dispositioned in the corrective action program. Documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

1EP6 Emergency Planning Drill Evaluation

a. Inspection Scope

The inspectors observed a site emergency preparedness training drill/simulator scenario conducted on February 4, 2009. The inspectors reviewed the drill scenario narrative to identify the timing and location of classifications, notifications, and protective action recommendations development activities. During the drill, the inspectors assessed the adequacy of event classification and notification activities. The inspectors observed portions of the licensee's post-drill. The inspectors verified that the licensee properly evaluated the drill's performance with respect to performance indicators and assessed drill performance with respect to drill objectives.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstones: Occupational Radiation Safety (OS) and Public Radiation Safety (PS)

2OS1 Access Control To Radiologically Significant Areas

a. Inspection Scope

Access Controls The inspectors evaluated licensee performance in controlling worker access to radiologically significant areas and monitoring jobs in-progress associated with the Unit 2 Refueling Cycle 19 (B219R1) outage. The inspectors directly observed implementation of administrative and physical radiological controls; reviewed and discussed general employee radiation worker (radworker) training; evaluated radworker and health physics technician (HPT) knowledge of and proficiency in implementing radiation protection requirements; and assessed worker exposures to radiation and radioactive material.

During facility tours, the inspectors directly observed postings and physical controls for radiation area, high radiation area (HRA), and potential airborne radioactivity area locations established within the radiologically controlled area (RCA) of the Unit 2 (U2) drywell, Unit 1 (U1) and U2 reactor buildings, U1 and U2 turbine buildings and radioactive waste (radwaste) processing and storage locations. The inspectors independently measured radiation dose rates or directly observed conduct of licensee radiation surveys for selected RCA areas or equipment. Results were compared to current licensee surveys and assessed against established postings and Radiation Work Permit (RWP) controls. Licensee key control and access barrier effectiveness were evaluated for selected U1 and U2 Locked High Radiation Area (LHRA) and Very High Radiation Area (VHRA) locations. Changes to procedural guidance for LHRA and VHRA controls were discussed with health physics (HP) supervisors. Controls and their implementation for storage of irradiated materials within the U1 and U2 spent fuel pool (SFP) locations were reviewed and discussed in detail. Established radiological controls were evaluated for selected B219R1 tasks including fuel movement under-vessel control rod drive (CRD) replacement, reactor head lift, feedwater valve repair, and reactor recirculation pump replacement. In addition, licensee controls for areas where dose rates could change significantly as a result of plant shutdown and refueling operations were reviewed and discussed.

For selected tasks, the inspectors attended pre-job briefings and reviewed RWP details to assess communication of radiological control requirements to workers. Occupational worker adherence to selected RWPs and HPT 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 U2 drywell and refueling floor activities observed.

The inspectors evaluated the effectiveness of radiation exposure controls, including air sampling, barrier integrity, engineering controls, and postings through a review of both internal and external exposure results. Worker exposure as measured by ED and by licensee evaluations of skin doses resulting from discrete radioactive particle or dispersed skin contamination events during current B219R1 activities were reviewed.

For HRA tasks involving significant dose rate gradients, e.g. refueling activities, the inspectors evaluated the use and placement of whole body and extremity dosimetry to monitor worker exposure. The inspectors also reviewed and discussed selected whole-body count analyses conducted during the current B219R1 U2 outage.

Radiation protection activities were evaluated against the requirements of Updated Final Safety Analysis Report (UFSAR) Section 12; Technical Specifications (TS) Sections 5.4 and 5.7; 10 Code of Federal Regulations (CFR) Parts 19 and 20; and approved licensee procedures. Records reviewed are listed in Sections 2OS1, 2OS2, 2PS2 and 4OA1 of the report Attachment.

<u>Problem Identification and Resolution</u> Licensee Corrective Action Program (CAP) documents associated with access control to radiologically significant areas were reviewed and assessed. This included review of selected Condition Report Quality Assurance records related to radworker and HPT performance. The inspectors evaluated the licensee's ability to identify, characterize, prioritize, and resolve the

identified issues in accordance with procedure Corrective Action Program (CAP) – NGGC-0200, Corrective Action Program, Revision (Rev.) 27. The inspectors also evaluated the scope of the licensee's internal audit program and reviewed recent assessment results. Licensee CAP documents reviewed are listed in Section 2OS1 of the report Attachment.

The inspectors completed 21 of the required line-item samples described in Inspection Procedure (IP) 71121.01.

b. Findings

No findings of significance were identified.

2OS2 As Low As Reasonably Achievable (ALARA) Planning and Controls

a. Inspection Scope

As Low As Reasonably Achievable (ALARA) The inspectors reviewed ALARA program guidance and its implementation for ongoing 2 R19 job tasks. The inspectors evaluated the accuracy of ALARA work planning and dose budgeting, observed implementation of ALARA initiatives and radiation controls for selected jobs in-progress, assessed the effectiveness of source-term reduction efforts, and reviewed historical dose information.

ALARA planning documents and procedural guidance were reviewed and projected exposure estimates were compared to actual dose expenditures for the following high dose jobs: "B" Recirculation Pump Motor Replacement, CRD Removal/Installation, Mechanical Maintenance (Balance of Plant), Scaffolding Installation/Removal, and Unqualified Coatings Removal. The inspectors reviewed the integration of ALARA work plan requirements into specific job task RWPs. Differences between budgeted dose and actual exposure received were discussed with ALARA staff. Changes to dose budgets relative to changes in radiation source term and/or job scope also were discussed. The inspectors attended pre-job briefings and evaluated the communication of ALARA goals, RWP requirements, and industry lessons-learned to job crew personnel. The inspectors also attended an ALARA Committee meeting and observed the interface between plant management and ALARA planning staff.

The inspectors made direct field or closed-circuit-video observations of outage job tasks involving recirculation pump motor replacement and scaffolding installation/removal. For the selected tasks, the inspectors evaluated radworker and HPT job performance (including use of low dose waiting areas); individual and collective dose expenditure versus percentage of job completion; surveys of the work areas, appropriateness of RWP requirements; and adequacy of implemented engineering controls. For selected high dose jobs, the inspectors interviewed radworkers and job sponsors regarding understanding of dose reduction initiatives and their current and expected accumulated doses at completion of the job tasks. The inspectors also reviewed in-progress and post-job reviews for selected high dose job tasks.

Implementation and effectiveness of selected program initiatives with respect to source-term reduction were evaluated. Chemistry program ALARA initiatives and their effect on

U2 drywell dose rate trends were reviewed. The inspectors evaluated the licensee's program for reduction of cobalt and reviewed the list of valves identified for cobalt mitigation. The effectiveness of temporary shielding installed for the current outage was assessed through review of shielding request packages and pre-shielding versus post-shielding dose rate data.

Plant exposure history for 2006 through 2008 and data reported to the NRC pursuant to 10 CFR 20.2206 were reviewed, as were established goals for reducing collective exposure during the current B219R1 outage. The inspectors reviewed procedural guidance for dosimetry issuance and exposure tracking. The inspectors also examined dose records of declared pregnant workers to evaluate assignment of gestation dose. In addition, selected individual access records were reviewed for dose received during work in areas with high dose rates.

ALARA program activities and their implementation were reviewed against 10 CFR Part 20, and approved licensee procedures. In addition, licensee performance was evaluated against guidance contained in Regulatory Guide (RG) 8.8, Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low As Reasonably Achievable and RG 8.13, Instruction Concerning Prenatal Radiation Exposure. Procedures and records reviewed within this inspection area are listed in Sections 2OS2 of the report Attachment.

Problem Identification and Resolution The inspectors reviewed selected NCRs in the area of exposure control. The inspectors evaluated the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with procedure CAP-NGGC-0200, Corrective Action Program, Rev. 27. The inspectors also evaluated the scope of the licensee's internal audit program and reviewed recent assessment results. Licensee CAP documents reviewed are listed in Section 2OS2 of the report Attachment.

The inspectors completed 23 of the line-item samples described in IP 71121.02.

b. Findings

No findings of significance were identified.

2PS2 Radioactive Material Processing and Transportation

a. Inspection Scope

Waste Processing and Characterization Selected liquid and solid radioactive waste (radwaste) processing system components were inspected for material condition and for configuration compliance with the UFSAR and Process Control Program (PCP) details. Inspected equipment included the waste hold-up and processing tanks; demineralizer systems; resin transfer piping; radwaste processing equipment; and abandoned waste processing equipment. The inspectors discussed component function, equipment operability, and changes to radwaste processing systems with licensee staff.

Radioactive waste disposal data for calendar year (CY) 2007 and CY 2008 were reviewed and discussed. Radionuclide characterizations from January 1, 2007, through

March 1, 2009, for selected radioactive waste streams were reviewed and discussed with cognizant staff. Licensee guidance and processes for monitoring changes in waste stream isotopic mixtures were discussed with cognizant licensee representatives. For irradiated metal equipment, reactor water clean-up and condensate resins, spent filters, and dry active waste (DAW), the inspectors reviewed radionuclide determination analyses and evaluated determination of hard-to-detect nuclides. The subject reviews included evaluation of gamma spectroscopy data comparisons of licensee waste stream analyses with vendor laboratory data and verification of appropriate use of scaling factors for waste characterization.

Radwaste processing activities were reviewed for compliance with 10 CFR Part 50.59 and consistency with current licensee PCP and UFSAR details. Waste stream characterization analyses and selected shipping records were reviewed against regulations detailed in 10 CFR Part 20, 10 CFR Part 61, 49 CFR Part 173, and guidance provided in the Branch Technical Position (BTP) on Waste Classification and Waste Form. Reviewed documents are listed in Section 2PS2 of the report Attachment.

Transportation During the week of March 23, 2009, the inspectors directly observed preparation activities for a radioactive material shipment of low specific activity material. The inspectors noted package bracing and conveyance placards, evaluated shipping paper documentation for adequacy and completeness, and interviewed the shipping technicians regarding knowledge of Department of Transportation (DOT) regulations. The inspectors also reviewed dose rate and contamination survey data for the shipping package and compared the results to DOT limits.

In addition, five additional shipping records and supporting documents for radioactive material and radioactive waste shipments were reviewed for consistency with licensee procedures and compliance with NRC and DOT regulations. The inspectors reviewed emergency response information, DOT shipping package classification, radiation survey results, and evaluated whether receiving licensees were authorized to accept the packages. Licensee procedures for use of Type B shipping casks were compared to recommended vendor protocols and Certificate of Compliance (CoC) requirements. In addition, training records for individuals currently qualified to ship radioactive material were reviewed.

Transportation program implementation was reviewed against regulations detailed in 10 CFR Parts 20 and 71, 49 CFR Parts 172-178; as well as the guidance provided in NUREG-1608. Training activities were assessed against 49 CFR Part 172 Subpart H. Documents reviewed during the inspection are listed in Section 2PS2 of the report Attachment.

Problem Identification and Resolution. The inspectors reviewed and discussed with HP supervision selected CRs and audits associated with transportation and radioactive waste processing program activities. The inspectors assessed the licensee's ability to characterize, prioritize, and resolve the identified issues in accordance with licensee procedure NMP-GM-002, Corrective Action Program, Ver. 7.0 and NMP-GM-002-001, Corrective Action Program Instruction, Ver 8.

The inspectors completed the six specified line-item samples detailed in IP 71122.02.

b. Findings

No findings of significance 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 Indicator Guideline.

Initiating Events Cornerstone

- Unplanned Scrams per 7000 Critical Hours
- Unplanned Scrams with Complications
- Unplanned Power Changes per 7000 Critical Hours

The inspectors sampled licensee submittals for the performance indicators listed above for the period of January 1, 2008, through December 31, 2008. The inspectors reviewed the licensee's operator narrative logs, issue reports, event reports and NRC Inspection reports for the period to validate the accuracy of the submittals. Specific documents reviewed are listed in the report Attachment.

Occupational Radiation Safety Cornerstone The inspectors reviewed PI data collected from April 1, 2008, through December 31, 2008, 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 documents relative to this PI are listed in Sections 2OS1, 2OS2, and 4OA1 of the report Attachment.

Public Radiation Safety Cornerstone The inspectors reviewed the Radiological Control Effluent Release Occurrences PI results for the period of April 1, 2008, through December 31, 2008. For the assessment period, the inspectors reviewed cumulative and projected doses to the public, and verified compensatory sampling was conducted as required for out-of-service (OOS) effluent radiation monitors. The inspectors also reviewed licensee procedural guidance for collecting and documenting PI data. Documents reviewed are listed in Section 4OA1 of the report Attachment.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

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

a. 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 corrective action program. The review was accomplished by reviewing daily action request reports.

b. Findings

No findings of significance were identified.

.2 Selected Issue Follow-up Inspection: Unplanned LCO Entry, Control Building Ventilation Isolation

a. Scope

The inspectors selected AR #281950, Unplanned LCO Entry, Control Building Ventilation Isolation for detailed review. This AR was associated with an inadvertent loss of control air in the control building ventilation system due to improper maintenance procedures. The inspectors reviewed this report to verify that the licensee identified the full extent of the issue, performed an appropriate evaluation, and specified and prioritized appropriate corrective actions. The inspectors evaluated the report against the requirements of the licensee's corrective action program as delineated in corporate procedure CAP-NGGC-0200, Corrective Action Program, and 10 CFR 50, Appendix B.

b. Findings

Introduction: A self-revealing Green NCV of Technical Specification (TS) 5.4.1, Procedures, was identified for inadequate maintenance procedures for the control room air conditioning and emergency ventilation system instrument air dryer. As a result, on January 21, 2009, the control room air conditioning and emergency ventilation instrument air system lost air pressure, rendering the control room air conditioning (AC) system and the control room emergency ventilation (CREV) system inoperable.

Description: The control room air conditioning and emergency ventilation instrument air system supplies control air from two air compressors, through a single air dryer to the three control room AC units and the two CREV systems. Since Unit 1 and Unit 2 share a common control room, the control room AC and CREV systems are also shared between the two units. The air dryer uses a refrigerant system to cool moisture in the air tubing and drains the moisture out of the air tubing through a moisture separator installed at a low point in the tubing below the air dryer. On January 21, 2009, at approximately 1:00 p.m. ambient temperature was above freezing, but the air dryer cooled the moisture in the instrument air tubing to below freezing. The moisture froze and blocked the instrument air line. The line blockage prevented the air compressors from maintaining system pressure. The loss of air pressure prevented the three control room AC units from operating, caused all of the isolation dampers in the CREV system to shut, and caused all of the fans in the CREV system to turn off. The control room remained isolated, and the control room AC and the CREV systems remained inoperable until 2:29 p.m. on January 21, 2009, when air pressure was restored by removing the ice from the air tubing and installing temporary heaters in the room to prevent further freezing.

The vendor manual for the air dryer, FP-81539, section 2d, states that the air dryer is factory set to avoid air side freezing down to an ambient temperature of 40ºF and an altitude up to 1500 feet. FP-81539 also states that if ambient temperature drops below 40ºF, the refrigerant pressure must be increased in order to avoid freezing. This guidance was not incorporated into the maintenance procedures for the control room air conditioning and emergency ventilation instrument air system, and heaters did not maintain the temperature in the area where the dryer is located significantly above 40ºF. When ambient temperatures remained consistently low throughout the day and preceding night of January 21, 2009, the air line froze.

Analysis: The failure to implement an adequate maintenance procedure for the control room air conditioning and emergency ventilation instrument air system is a performance deficiency. This performance deficiency is more than minor because it is associated with structure, system, and component (SSC), and barrier performance attribute of the Barrier Integrity Cornerstone. It also adversely affected the cornerstone objective of maintaining a radiological barrier for the control room. The finding was determined to be of very low safety significance because it only affected the radiological barrier function of the control room, and does not represent a degradation of the smoke or toxic atmosphere barrier function of the control room. The cause of the finding is related to the cross-cutting area of human performance, resources component, complete and accurate documentation aspect because the licensee did not incorporate adequate guidance for maintaining the control room AC and CREV instrument air dryer in their maintenance procedures. (H.2(c))

Enforcement: TS 5.4.1, Procedures, requires that written procedures shall be implemented covering applicable procedures recommended in Regulatory Guide 1.33, Appendix A, November 1972 (Safety Guide 33, November 1972). Regulatory Guide 1.33, section I (Safety Guide 33, November 1972) requires written procedures for maintenance that can affect the performance of safety-related equipment. Contrary to the above, maintenance instructions did not contain adequate guidance for maintaining the control room AC and CREV instrument air dryer. Because the finding is of very low safety significance and has been entered into the CAP (NCR #315770), and consistent with the NRC Enforcement Policy, this violation is being treated as a non-cited violation, and is designated as NCV 05000324,325/2009002-05, Inadequate Maintenance Procedure for the Control Room Air Conditioning and Emergency Ventilation Instrument Air System.

4OA3 Follow-up of Events

.1 (Closed) LER 05000324, 325/2008-002: Loss of Two Control Room Air Conditioning (AC) Subsystems. On June 3, 2008, two operating Control Room Air Conditioners tripped during the replacement of a solenoid valve for the Unit 1 Cable Spread Room supply/exhaust fan dampers due to a maintenance error which caused an interruption of control air to the system. Maintenance procedures have been modified to prevent a similar occurrence. This event was identified as a Green non-cited violation of TS 5.4.1 in NRC Inspection Report 05000325,324/2008003 (NCV 05000325,324/2008003-01). This LER is closed.

.2 (Closed) LER 05000324, 325/2008-004: Control Room Emergency Ventilation (CREV) Subsystems Inoperable Due to Failure to Isolate. On June 19, 2008, the Control Room authorized post-maintenance testing following replacement of solenoid valves affecting the CREV subsystem. During performance of the test, the 2D Control Building exhaust fan damper failed to close and the associated Control Building exhaust fan failed to trip as expected. The replaced solenoid valve failed after installation. The failed solenoid valve was replaced and the system was returned to service. This LER was reviewed and no findings of significance were identified and no violation of NRC requirements occurred. This LER is closed.

.3 (Closed) LER 05000324/2008-001: Automatic Reactor Scram Due to Turbine Power Load Unbalance Actuation. On August 30, 2008, the Unit 2 reactor automatically scrammed due to a spurious power load unbalance signal which tripped the main turbine. The spurious signal was determined to be generated from maintenance activities in the switchyard. This maintenance was being performed to install a fault recorder in the generator output circuitry. During testing of the fault recorder, a test device was installed in the circuitry which caused a momentary short circuit which ultimately caused the scram. The licensee determined that the maintenance was not being controlled adequately and has put corrective actions in place, including revising procedures to ensure switchyard activities are properly risk-evaluated. This event was determined to be a licensee-identified violation of 10CFR50.65a(4) as documented in NRC Inspection Report 05000325,324/2008004, section 4OA7. This LER is closed.

.4 (Closed) LER 05000324/2008-002: Manual Reactor Scram Due to Spurious Safety Relief Valve (SRV) Opening. On November 9, 2008, an SRV spuriously opened, resulting in high suppression pool temperature and leading Unit 2 operators to manually scram the Unit 2 reactor. The root cause of the spurious SRV actuation was improper assembly of the relief valve pilot valve by licensee maintenance personnel. The pilot valve was improperly assembled because the pilot spring was placed on the ledge of the pilot spring follower. The spring subsequently slipped off of the ledge, effectively lengthening the spring and lowering the relief valve setpoint. When the spring slipped off of the spring follower ledge, the valve opened at normal operating pressure. The licensee has implemented corrective actions which revised the procedure for pilot valve assembly to inspect the spring after valve assembly to ensure it is seated properly within the spring follower. This event was determined to be a Green non-cited violation of TS 5.4.1, Procedures in NRC Inspection Report 05000325,324/2008005 (NCV 05000325/2008005-01). This LER is closed.

.5 (Closed) LER 05000325/2008-007: Automatic Reactor Scram Due to Electro-Hydraulic Control (EHC) System Failure. On November 26, 2008, the Unit 1 reactor automatically scrammed due to an EHC system failure, which caused control valves to open excessively and cause a low reactor pressure reactor scram. The EHC system failure was caused by a circuit board becoming unseated and losing electrical contact. The circuit board unseated because the circuit board guide slot was slightly rounded, preventing full engagement with the terminal receptacle. The affected EHC circuit board was replaced and the licensee changed maintenance procedures to require a visual inspection to ensure EHC circuit boards are fully engaged after EHC system maintenance. This LER was reviewed and no findings of significance were identified and no violation of NRC requirements occurred. This LER is closed.

.6 (Closed) LER 05000325/2009-001: Loss of Control Room Air Conditioning and Emergency Ventilation System. On January 21, 2009, a loss of control building heating, ventilation, and air conditioning (HVAC) control air occurred. As a result, the two control room emergency ventilation subsystems and the three control room air conditioning subsystems became inoperable. The direct cause of the loss of control building HVAC control air was blockage of air flow through the system air dryer due to freezing of condensate within the dryer’s cooling coils. The freezing occurred because ambient temperature in the area of the dryer was allowed to reach the freezing point of the refrigerant in the dryer and maintenance procedures did not require adjustment of refrigerant pressure to preclude the freezing. The licensee restored control air by heating the room with portable heaters. Corrective actions include implementing procedures to prevent ambient temperature from dropping to the refrigerant freezing point. The event is a non-cited violation of TS 5.4.1, Procedures, which is discussed in section 4OA2 of this report. This LER is closed.

4OA5 Other Activities

.1 Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

During the inspection period the inspectors conducted observations of security force personnel and activities to ensure that the activities were consistent with licensee security procedures and regulatory requirements relating to nuclear plant security. These observations took place during both normal and off-normal plant working hours.

These quarterly resident inspector observations of security force personnel and activities did not constitute any additional inspection samples. Rather, they were considered an integral part of the inspectors' normal plant status reviews and inspection activities.

b. Findings

No findings of significance were identified.

.2 Independent Spent Fuel Storage Installation (ISFSI)

a. Inspection Scope

The inspectors examined installation of the reinforcing steel and form work configuration, observed the concrete placement, and reviewed Quality Assurance documents associated with the ISFSI Cask Storage Pad.

The east ISFSI pad was 270 ft long X 45 ft wide and was divided into three separate sections with two construction joints. The inspectors witnessed the placement of the middle section of the east pad to ensure the licensee had measured the reinforcing steel size, spacing, splice length, and the concrete minimum protection coverage in accordance with the requirements of the design drawings and the American Concrete Institute (ACI) 349, Code Requirements for Nuclear Safety-Related Concrete. The inspectors reviewed the concrete pre-placement inspection checklist prior to the concrete pour. The inspectors reviewed the procedures, specifications, and calculations for the concrete pad construction activities. The inspectors also reviewed the licensee cold weather concrete activity protection with the requirements of ACI 306.1, Standard Specification for Cold Weather Concreting.

The inspectors observed placement activities to verify that activities pertaining to concrete delivery time, flow distance, layer thickness and concrete consolidation or vibration conformed to industry standards established by the American Concrete Institute. Concrete batch tickets were examined to verify that the specified concrete mix was being delivered to the site. The inspectors observed that concrete placement activities were continuously monitored by the licensee quality control and qualified contractors. The inspectors witnessed in-process testing and reviewed the results for slump, air content, temperature, unit weight, and molding of the concrete cylinders for the compressive strength testing, and also witnessed sample points and portion of truck loads to verify that concrete samples for the field testing and cylinders for the testing were obtained at the point of placement (end of pump line) and the middle portion of the truck loads. The inspectors reviewed that the cylinders were molded in accordance with applicable American Society for Testing and Materials (ASTM) requirements of ASTM C 172, Standard Method of Sampling Freshly Mixed Concrete, and that concrete field testing was performed by qualified inspectors from an independent testing company.

The inspectors also reviewed records documenting inspection of the concrete batch plant and concrete truck mixers performed by an independent engineering and consulting company. The consulting company verified that the batch plant and trucks met the requirements of National Ready Mixed Concrete Association (NRMCA).

Activities were reviewed to determine if the consulting company's inspection or verification of the trucks and batch plant were performed in accordance with engineering requirements. The inspectors reviewed the concrete mix data to ensure that mix proportions for delivered concrete were selected based on trial concrete mix results, and that the trial mix met concrete strength requirements.

b. Findings

No findings of significance were identified.

.3 (Closed) Temporary Instruction (TI) 2515/176, EDG TS Surveillance Requirements Regarding Endurance and Margin Testing

Inspection activities for TI 2515/176 were previously completed and documented in inspection report 05000325/2008004 and 05000324/2008004, and this TI is considered closed at Brunswick Steam Electric Plant; however, TI 2515/176 will not expire until August 31, 2009. The information gathered while completing this temporary instruction was forwarded to the Office of Nuclear Reactor Regulation for review and evaluation.

.4 Institute of Nuclear Power Operations (INPO) Plant Assessment Report Review

The inspectors and DRP Branch Chief reviewed the preliminary report dated March 26, 2009 for the INPO plant assessment conducted in January, 2009. The report was reviewed to ensure that issues identified were consistent with the NRC perspectives of licensee performance and to verify if any significant safety issues were identified that required further NRC follow-up.

4OA6 Management Meetings

.1 Exit Meeting Summary

On March 13, 2009, the inspectors presented the results in section 1R08 of this report to licensee management.

On March 27, 2009, the inspectors discussed results of the onsite radiation protection inspection with Mr. Ben Waldrep, Plant Vice President, and other responsible staff. The inspectors noted that proprietary information was reviewed during the course of the inspection but would not be included in the documented report.

On April 17, 2009 the inspectors presented the inspection results to Mr. Ben Waldrep, and other members of the licensee staff. The inspectors confirmed that proprietary information was not provided or examined during the inspection period.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

M. Annacone, Director Site Operations
G. Atkinson, Supervisor – Licensing and Regulatory Affairs
C. Barnhill, Dosimetry Supervisor
L. Beller, Superintendent, Operations Training
M. Blew, Engineering
A. Brittain, Manager – Security
J. Crider, ISFSI Project Supervisor
B. Davis, Manager – Engineering
P. Dubrouillet, Supervisor - Operations Support
L. Grzeck, Lead Engineer - Technical Support
B. Harlee, Health Physics Supervisor
S. Howard, Manager - Operations
R. Ivey, Manager – Station Recovery
J. Johnson, Manager – Environmental and Radiological Controls
S. Larson, ISI Coordinator
B. Mclendon, Health Physics Supervisor
P. Mentel, Manager - Support Services
E. Morris, Instrument Supervisor
W. Murray, Licensing Specialist
J. Piepmeyer, Environmental and Chemistry Superintendent
A. Pope, Manager - Maintenance
E. Rochelle, Health Physics Supervisor
T. Sherrill, Engineer - Technical Support
G. Simmons, Environmental and Radiation Control Manager
G. Spry, Welding Engineer
J. Titrington, Manger - Nuclear Oversight Section
R. Tripp, ISFSI Project Manager
M. Turkal, Lead Engineer - Technical Support
J. Vincelli, Superintendent - Environmental and Radiological Controls
B. Waldrep, Site Vice President
M. Williams, Manager - Training Manager
E. Wills, Plant General Manager
B. Wilton, Engineering

NRC Personnel

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

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000325/2009002-01	NCV	Failure to Identify and Correct a Condition Adverse to Quality Affecting the Operability of the Standby Gas Treatment Train B (Section 1R06)
05000325,324/2009002-02	NCV	Failure to Perform a 10 CFR 50.59 Evaluation for a Plant Modification (Section 1R18)
05000324/2009002-03	NCV	Failure to Follow Procedures During Reactor Head Disassembly (Section 1R20)
05000324/2009002-04	NCV	Unauthorized Maintenance Results in Loss of Shutdown Cooling (Section 1R20)
05000324,325/2009002-05	NCV	Inadequate Maintenance Procedure for the Control Room Air Conditioning and Emergency Ventilation Instrument Air System (Section 4OA2)

Closed

05000324, 325/2008-002	LER	Loss of Two Control Room Air Conditioning (AC) Subsystems. (Section 4OA3)
05000324, 325/2008-004	LER	Control Room Emergency Ventilation (CREV) Subsystems Inoperable Due to Failure to Isolate (Section 4OA3)
05000324/2008-001	LER	Automatic Reactor Scram Due to Turbine Power Load Unbalance Actuation (Section 4OA3)
05000324/2008-002	LER	Manual Reactor Scram Due to Spurious Safety Relief Valve (SRV) Opening (Section 4OA3)
05000325/2008-007	LER	Automatic Reactor Scram Due to Electro-Hydraulic Control (EHC) System Failure (Section 4OA3)
05000325/2009-001	LER	Loss of Control Room Air Conditioning and Emergency Ventilation System (Section 4OA3)
2515/176	TI	EDG TS Surveillance Requirements Regarding Endurance and Margin Testing (Section 4OA5)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

0AOP-13.0, Operation during Hurricane, Flood Conditions, Tornado, or Earthquake
0A1-68, Brunswick Nuclear Plant Response to Severe Weather Warnings
0PEP-02.1, Initial Emergency Actions
0PEP-02.6, Severe Weather
0O1-01.03, Non-Routine Activities
0PM-HT001, Preventative Maintenance on Plant Freeze Protection and Heat Tracing System

Section 1R04: Equipment Alignment

1OP-17, Residual Heat Removal System Operating Procedure
2OP-17, Residual Heat Removal System Operating Procedure
Drawing D-25026, Reactor Building Residual Heat Removal System Piping Diagram
2OP-05, Standby Liquid Control System
Drawing D-02547, Standby Liquid Control System Piping Diagram

Work orders:

01482320	POST-MAINTENANCE UT INSPECTION OF 2B RHR
01482323	POST-MAINTENANCE UT INSPECTION OF 2A RHR
01483931	ULTRASONIC INSP. LINE NO 2-E11-117-24-903
01483931	ULTRASONIC INSPECTION OF LINE NO. 2-E11-117-24-903
01483932	ULTRASONIC INSP. LINE NO 2-E11-118-24-903
01483932	ULTRASONIC INSPECTION OF LINE NO. 2-E11-118-24-903
01483934	UT INSPECTION OF LINE NO. 2-E11-89-4-300
01483934	SI-POST ULTRASONIC INSP. OF LINE NO. 2-E11-89-4-300
01484688	1-E11-F024B REPACK.
01488640	1-E11-C002B-HX: RETIGHTEN OR REPLACE FITTING
01489566	2-E11-F046B: REPLACE VALVE DURING B220R1
01489696	2-E11-F046D: REPLACE VALVE DURING B220R1

Nuclear Condition Reports:
260657, Operability Evaluation Not Performed for Pipe Support
264459, RHR Loop Continuous Vent Discrepancies
270644, Slow Stroke Time on 1-SW-V136
271849, Failed IST Visual Inspection of Valves
272371, Pipe Wall Thinning
279746, Slow Stroke Time on 1-E11-F040
283428, 2B RHRSW Booster Pump Increased Vibrations
298826, 2B RHRSW Booster Pump LCO Entry for High Vibrations

For easy reference, here are some commonly used doc's:
0OP-50.1, Diesel Generator Emergency Power System Operating Procedure
Drawing D-02265, sheets 1A and 1B, drawing D-02266, sheets 2A and 2B, Piping Diagram for Diesel Generators Starting Air System Units 1 and 2

Drawing D-02268, sheets 1A and 1B, drawing D-02269, sheets 2A and 2B, Piping Diagram for Diesel Generators Fuel Oil System Units 1 and 2
Drawing D-02270, sheets 1A and 1B, drawing D-02271, sheets 2A and 2B, Piping Diagram for Diesel Generators Lube Oil to Lube Oil System Units 1 and 2
Drawing D-02272, sheets 1A and 1B, drawing D-02273, sheets 2A and 2B, Piping Diagram for Diesel Generators Jacket Water System Units 1 and 2
Drawing D-02272, sheets 1A and 1B, drawing D-02273, sheets 2A and 2B, Piping Diagram for Diesel Generators Jacket Water System Units 1 and 2
Drawing D-02274, sheets 1 and 2, Piping Diagram for Diesel Generators Service and Demineralized Water System Units 1 and 2
1OP-16, Reactor Core Isolation Cooling System Operating Procedure
2OP-16, Reactor Core Isolation Cooling System Operating Procedure
1OP-19, High Pressure Cooling Injection System Operating Procedure
2OP-19, High Pressure Cooling Injection System Operating Procedure
1OP-17, Residual Heat Removal System Operating Procedure
2OP-17, Residual Heat Removal System Operating Procedure
Drawing D-25026, Reactor Building Residual Heat Removal System Piping Diagram
2OP-05, Standby Liquid Control System
Drawing D-02547, Standby Liquid Control System Piping Diagram

Section 1R05: Fire Protection

1PFP-RB, Reactor Building Prefire Plans Unit 1
2PFP-RB, Reactor Building Prefire Plans Unit 2

For easy reference, here are some doc numbers:
0PFP-CB, Control Building Prefire Plans
0PFP-DG, Diesel Generator Building Prefire Plans
0PFP-PBAA, Power Block Auxiliary Areas Prefire Plans SW, RW, AOG, TY, EY
0PFP-013, General Fire Plan
1PFP-RB, Reactor Building Prefire Plans Unit 1
1PFP-TB, Turbine Building Prefire Plans Unit 1
2PFP-RB, Reactor Building Prefire Plans Unit 2
2PFP-TB, Turbine Building Prefire Plans Unit 2
0OP-41, Fire Protection and Well Water System
0PFP-MBPA, Miscellaneous Buildings Pre-Fire Plans – Protected Area
0PT-34.11.2.0, Portable Fire Extinguisher Inspection

Section 1R07: Heat Sink Performance

0ENP-2704, Administrative Control of NRC Generic Letter 89-13 Requirements
NLS-90-005, CP&L Response To NRC Generic Letter 89-13
0PM-STU500, Service Water Intake Structure Inspection and Cleaning
0MST-DG500R, Emergency Diesel Generators 24 Month Inspection
0PM-HX503, RHR Service Water Booster Pump Motor Heat Exchanger Inspection
0PM-ACU500, Inspection and Cleaning of the RHR/Core Spray Room Aerofin Cooler Air Filters and Coolers
0ENP-2705, Performance Trending of RHR Heat Exchangers

Calculation 0SW-0096, Calculation for Tube Plugging and Fouling of Service Water Safety Related Heat Exchangers

Calculation 0SW-0097, RHR and Core Spray Room Cooler Performance
Calculation G0050A-16, Service Water Single Failure Analysis

Section 1R08: Inservice Inspection Activities

Procedures:
Welding Material Issue Procedure for Brunswick Nuclear
NDEP-0437 Ultrasonic Testing Calibration/Examination for Component 2E1163-21-SWC Rev. 03
NDEP-0425 Ultrasonic Testing Calibration/Examination for Component 2B32RECIRC-28-B-8 Rev. 08
NDEP-0613 Visual Examination (VT-3) of Component Supports and Snubbers for component 2-B32-PS7505A Rev. 20

Corrective Action Documents:
NCR 322187 2E11-F048B Body Drain Line Weld Leak dated 02/28/2009
NCR 216783 Self-Assessment - BNP and CR3 Inservice Inspection Program dated 11/17/08
NCR 324973 Flaws Found in Jet Pump Riser Elbow JPCRS-1 dated 3/12/2009
NCR 324976 Flaws Found in 90 Degree Core Spray T-box dated 3/12/2009
NCR 323729 Small Flaw in Core Shroud V-2 Weld dated 3/07/2009
NCR 323799 Flaws in Core Shroud V-6 Weld dated 3/08/2009

Other Documents:
VT-3 of Component 2-B32-PS7505A Visual Examination of Component Supports and Snubbers Record
UT of Component 2E1163-21-SWC Calibration Examination Report
UT of Component 2B32RECIRC-28-B-8 Calibration Examination Report
BWRVIP-76-A: BWR Vessel and Internals Project BWR Core Shroud Inspection and Flaw Evaluation Guidelines
Performance Quality Test Records for Welders and Welding Operators for Ryan L. Bothe
Performance Quality Test Records for Welders and Welding Operators for Joshua A. White
Performance Quality Test Records for Welders and Welding Operators for Ricky N. Jones
NDE Certification Review Report Form 7 for Paul S. Blecha

Section 1R11: Licensed Operator Requalification

0TPP, Licensed Operator Continuing Training Program
1EOP-01-LPC, Level/Power Control
0PEP-2.1.1, Emergency Control – Notification of Unusual Event, Alert, Site Area Emergency, or General Emergency
0PEP-02.1, Initial Emergency Actions
EOP-01-LEP-02, Alternate Control Rod Insertion
AOP-37.0, Low Condenser Vacuum
AOP-04.0, Low Core Flow

Section 1R12: Maintenance Effectiveness

ADM-NGGC-0101, Maintenance Rule Program
NUMARC 93-01, Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants
ADM-NGGC-0203, Preventive Maintenance and Surveillance Testing Administration
EGR-NGGC-0351, Condition Monitoring of Structures
ADM-NGGC-0203, Preventive Maintenance and Surveillance test Administration
0AP-022, BNP Outage Risk Management
AR# 272041, Operational Concerns with Supplemental Spent Fuel Pool Cooling (SSFPC)
AR# 272580, SSFPC System Vulnerability
AR# 273757, AOP-38.1 Entry Due to Loss of SSFPC
AR# 274228, SSFPC B Abnormal Noise

Section 1R13: Maintenance Risk Assessment and Emergent Work Control

0AP-022, BNP Outage Risk Management
ADM-NGCC-0104, Work Management Process
0AI-144, Risk Management
ADM-NGGC-0006, Online EOOS Model

Section 1R15: Operability Evaluations

OPS-NGGC-1305, Operability Determinations
OPS-NGGC-1307, Operational Decision making

Section 1R18: Plant Modifications

EGR-NGGC-0005, Engineering Change
EGR-NGGC-0011, Engineering Product Quality
AR# 75947, Isolation of Reactor Building Air Filters and Dryers
AR# 316630, Temporary Condition on Unit 1 and 2 RB Air Compressors Excessive
AOP-20, Loss of Service Air
1-EC-05-145, Unit 1 Equipment Control Tags for RB Standby Air Compressors
2-EC-05-257, Unit 2 Equipment Control Tags for RB Standby Air Compressors

Section 1R19: Post Maintenance Testing

0PLP-20, Post Maintenance Testing Program

Section 1R20: Outage Activities

1OP17, Residual Heat Removal System Operating Procedure
0GP-01, Prestartup Checklist
0GP-02, Approach to Criticality and Pressurization of the Reactor
0GP-03, Unit Startup and Synchronization
0GP-12, Power Changes

0FH-11, Refueling
0FH-11A, Refueling Platform Operations
0FH11N, Control Rod Shuffle
0CM-FH504, Control Rod Removal/Installation
0SMP-FH506, Fuel Preparation
0SMP-RPV501, Reactor Vessel Disassembly
0SMP-RPV502, Reactor Vessel Reassembly
0MMM-015, Operation and Inspection of Cranes and Material Handing Equipment

Section 2OS1: Access Controls to Radiologically Significant Areas

Procedures, Manuals, and Guidance Documents
0AI-112, Control Of Material In Spent Fuel Pool, Revision (Rev.) 19
HPS-NGGC-0024, Alpha Monitoring Guidelines, Rev. 0
REG-NGGC-0009, NRC Performance Indicators And Monthly Operating Report Data, Rev. 9
DOS-NGGC-0002, Dosimetry Issuance, Rev. 26
HPS-NGGC-0016, Access Controls, Rev. 4
OE&RC-0040, Administrative Controls For High Radiation Areas, Locked High radiation Areas, And Very High Radiation Areas, Rev. 30
OE&RC-0100, Radiation Surveys Methods, Rev. 33
OE&RC-0111, Survey Methods for Removable Surface Contamination, Rev. 32
OE&RC-0175, Radiological Controls for Diving Operations, Rev. 4
OE&RC-0241, Health Physics Coverage in the Drywells During Fuel Irradiated Component Movement, Rev. 14
HPS-NGGC-0003, Radiological Posting, Labeling and Surveys, Rev. 13
OE&RC-0020, Radiological Pre-Job Briefing, Rev. 5
OE&RC-0120, Routine/Special Airborne Radioactivity Survey, Rev. 22
HPS-NGGC-0014, Radiation Work Permits, Rev. 4
TRN-NGGC-0010, Plant access, radiation worker and respiratory protection training, Rev. 9
Corrective Action Program (CAP) – NGGC-0200, Corrective Action Program, Rev. 27.

Records and Data Reviewed
Unit 1 Fuel Pool Inventory WO 1331746-01, 02/19/09
Unit 2 Fuel Pool Inventories WO 1384445-01, 02/24/09
2008 List Of Personnel Contamination Reports, 01/01/08 – 02/03/08
Assessment Number 259364, Self Assessment of High radiation And Contamination Controls, 09/08/08 – 09/12/08
BNAS 0703, Radiation Protection Assessment, 02/20/07
BNAS 08-00603, Radiation Protection Assessment, 02/20/08
Occupational Cornerstone Performance Indicator Data for 2008 and 2009
U1 and U2 Locked High Radiation Area, Very High Radiation Area Surveillances, 03/10/09
Spent Fuel Pool Locked Item Surveillance, 03/10/09
Personnel Contamination Event 09-009, 03/10/09
Personnel Contamination Event 09-010, 03/10/09
Radiation Work Permit (RWP) Number (No.) 4654, Metallic Penetration and Surveillance
RWP No. 4220, Drywell Feedwater Valves
RWP No. 4613, Area Walk-downs

RWP No. 4051, Drywell Radiography
RWP No. 4658, Insulation Removal/Installation
RWP No. 4660, Drywell Snubbers Remove/Replace
RWP No. 4657, Drywell/Cavity Shielding
RWP No. 4660, Recirculation Pump Activities
RWP No. 4662, Decon Support
RWP No. 4669, CRD Replacements
RWP No. 4677, 2B Recirculation Pump Motor Replacement
RWP No. 4903, High Risk Task
RWP No. 4630, Reactor Vessel Disassembly/Reassembly - Cavity
Survey No. 031009-008, Drywell Unit 2
Survey Nos. 020209-004, 021109-008, Reactor Building Unit 1
Survey Nos. 022309-005, 021709-006, Reactor Building Unit 2
Survey No. 030409-007, Unit 2 Drywell
Survey No. 022809-013, Drywell Post Shielding
Survey Nos. 031009-036, 03089-052, Unit 2 Refueling Floor
Survey No. Outside Diesel Generator Building
Survey No. 121208-006, East/West Pipe Tunnel
Survey No. 121208-009, North/South Pipe Tunnel
Survey No. 121708-008, Drumming Room
Survey No. 020509-015, Radwaste South End
Survey No. 020209-005, Radwaste North End

Corrective Action Program (CAP) Documents
Nuclear Condition Report (NCR) 00317304, Contaminated tool in hot tool room
NCR 00316215, Dosimeter alarm,
NCR 00306052, Source room door unattended
NCR 00306090, Posting of area
NCR 00286099, Clean area contamination
NCR 00324186, Refueling floor contamination event
NCR 00324209, Incorrect RWP used
NCR 00271573, Refueling floor contamination event
NCR 00327512, Failure to implement management expectations for briefing documentation for dry well entries

Section 2OS2: As Low As Reasonably Achievable (ALARA)

Procedures, Manuals, and Guidance Documents
Brunswick Nuclear Plant ALARA Strategic Plan, June 2008
0E&RC-0020, Radiological Pre-job Briefing, Rev. 5
0E&RC-4270, Elemental Cobalt Sampling, Rev. 0
ADM-NGGC-0105, ALARA Planning, Rev. 8
DOS-NGGC-0002, Dosimetry Issuance, Rev. 26
MNT-NGGC-0003, Radiation Shielding Use, Rev. 11
CAP-NGGC-0200, Corrective Action Program, Rev. 27

Records and Data Reviewed

U2 Recirculation Piping Dose Rate Trending (BRAC Point) Data, B213R1 – B219R1
U1 and U2 Hot Spot Tracking Database
Declared Pregnant Worker Dosimetry Records, February 2007 – March 2009
B219R1 ALARA Work Plan 08-004, CRD Removal/Installation
B219R1 ALARA Work Plan 08-009, Scaffolding Installation/Removal
B219R1 ALARA Work Plan 08-017, "B" Recirc Pump Motor Replacement
B219R1 ALARA Work Plan 08-025, Refuel Floor
B219R1 ALARA Work Plan 08-033, Unqualified Coatings Removal
B219R1 ALARA Work Plan 08-034, Mechanical Maintenance (Balance of Plant)
In-Progress ALARA Evaluation 08-025, Refuel Floor Activities, 3/4/09
In-Progress ALARA Evaluation 08-034, Mechanical Maintenance (Balance of Plant), 3/10/09, 3/17/09, and 3/24/09
Post-Job ALARA Critique, 2677, 2A Recirc Pump Motor Replacement, 5/8/03
Temporary Shielding Request 2-RF-487, U2 Recirc Piping
Temporary Shielding Request 2-RF-498, U2 Recirc Piping Ringheader
RWP 4220, U2 Drywell/ Emergent/ Contingency – High Risk Activities
RWP 4669, CRD Exchange Requiring Multibadging
RWP 4677, U2 "B" Recirc Pump Motor Replacement
RWP 4678, U2 Drywell Unqualified Coatings Removal
Survey No. 032609-021, U2 Drywell FO31A Drain Valves
Survey No. 032209-032, U2 Drywell Personnel Hatch
Survey No. 022809-051, U2 Under vessel
Survey No. 030609-066, U2 Drywell "B" Recirc Pump Motor
Survey No. 037009-065, U2 F067 Valve
Survey No. 031807-012, Recirc Pump "A" Suction and Discharge Lines – Post Shielding
Survey No. 022809-013, U2 Drywell – Initial Downpost Survey
B219R1 Outage Daily Radiological Status, 3/9/09 – 3/13/09 and 3/23/09 – 3/27/09

CAP Documents

B-RP-08-01, BNP Radiation Protection Assessment
NCR 00325477, ALARA work plan 08-034 on track to exceed dose goal
NCR 00324397, Mechanical Maintenance (BOP) RWP being used instead of the One-Time Inspection RWP
NCR 00325474, ALARA work plan 08-004 appears on track to exceed outage dose goal
NCR 00315595, Brunswick exceeded the revised 2008 annual on-line dose goal
NCR 00220235, U2 pre-outage shielding exceeded dose goal
NCR 00315927, Activities were allowed to work in U2 south RHR room without shielding
NCR 00287234, NAS assessment of B117R1 identified ALARA planning weaknesses
NCR 00227051, Critique of ALARA Committee meeting identified several weaknesses
NCR 00315652, Challenges identified in cobalt valve cleanliness program

Section: 2PS2 Radioactive Material Processing and Transportation

Procedures, Manuals, and Guidance Documents

HPPS-NGCC-0001, Radioactive Material Receipt and Shipping Procedure, Rev. 26
HPPS-NGCC-0002, Vendor Cask Utilization Procedure, Rev. 15

Operations Special Procedure (OSP) 02-003, Handling and Use of the Energy Solutions CNS-3-55(B) Cask and CNS-8-120(B) Cask at Brunswick Nuclear Plant, Revision (Rev.) 9
T-Bolt Lid Closure Sequence, Rev. B 09/08/08
Certificate of Compliance (CoC) No. 5805 for the model No. CNS 3-55, Rev. 26
CoC No. 9168 for the model No. CNS 8-120B Package, Rev. 16

Records and Data Reviewed

Shipment No. 07-009, Radioactive material, surface contaminated objects (SCO-II), 7, UN2913, Fissile Excepted; Control Rod Drive (CRD) Assemblies, 01/18/2007
Shipment No. 08-076, Radioactive material, Type B(U) package, 7, UN2916, Fissile Excepted, RQ – Radionuclides; 1 Liner of RWCU Resin, 05/07/2008
Shipment No. 08-077, Radioactive material, Type B(U) package, 7, UN2916, Fissile Excepted, RQ – Radionuclides, 1 Steel Type B Cask; Irradiated LPRMs with fission chamber and CRBs with Stellite & VLs, 06/04/2008
Shipment No. 08-078, Radioactive material, Type B(U) package, 7, UN2916, Fissile Excepted, RQ – Radionuclides, (1) Steel Cask with Filters, VLs and DAW, 06/23/2008
Shipment No. 08-079, Radioactive material,Type B(U) package, 7, UN2916, Fissile Excepted, RQ – Radionuclides, 1 Metal Type B Cask; Irradiated CRBs, VLs, IRM & DAW, 06/26/08
Shipment No. 09-040, Radioactive material, low specific activity (LSA-I) UN2912; Laundry – 20 foot cargo van, 03/24/09
Composite Control Rod Drive Part 61 Data Results, 01/16/2007
Condensate Powdered Resin, Analysis of Multiple Sample Scaling Factor and Percent Abundance Data Comparisons, 07/31/2007 versus 04/18/2008
Reactor Water Clean-up Primary Resin, Analysis of Multiple Sample Scaling Factor and Percent Abundance Data Comparisons, 02/24/2006 versus 03/22/2007 Reactor Water Clean-up
Primary Resin, Analysis of Multiple Sample Scaling Factor and Percent Abundance Data Comparisons, 05/02/2008 versus 12/31/2007
Dry Active Waste, Analysis of Multiple Sample Scaling Factor and Percent Abundance Data Comparisons, 09/14/2006 versus 09/17/2008
CNS 8-120B Cask User Signoff-Sheet for Shipment Number 08-078, 06/09/2008
CNS 3-55 Cask 3-55 Cask User Signoff Sheet for Shipment Number 08-079 06/23/08
Survey Nos. 090408-02 (09/04/08) and 030709-44 (03/07/09, Loading Dock Area
Survey Nos. 030509-026, Shield No. 7 (03/05/09); 022108-016, Shield No. 16, (02/28/08); and 022108-016, Shield No. 16, (07/16/07).
Screening Package No. 232428, Operation of LWP-5 Equipment in the Radwaste Building
Screening Package No. 251219, Changes to Resin Processing Equipment in Resin Processing Area
Engineering Change (EC) Request No. 0000011820, Assess more modern and accurate method for determining actual bead resin level in radwaste spent resin tank, 10/08/2008

CAP Documents

NCR 00211471, Radiation levels in radwaste “A” centrifuge increasing
NCR 00261354, Unanticipated removable contamination in Resin Processing Area (RPA)
NCR 00268618, Contamination found on liners in the RPA
NCR 00262841, Susceptibility involving implementation of Brunswick radioactive waste management plant for class B and C radioactive waste post cessation of disposal options
NCR 00281495, Shipping cask O-ring failure

NCR 00282943, Suspension of hardware shipments
NCR 00287415, Unanticipated loss of suction during resin dewatering
NCR 00297755, C-vans not maintained on cribbing – NAS Assessment Issue
NCR 00299829, RCA clean trash that does not fit in small article monitors is routinely placed in radioactive waste containers
NCR 00309614, Vendor software error notification
NCR 00313912, Elevated dose rates in radwaste area
NCR 00321770, Radioactive spills, resin spill on -3 foot elevation radioactive waste area
NCR 00323031, Clean radwaste floor drain lines
NRC 00327121, Non-legible transportation records in QA records

Section 4OA1: Performance Indicator Verification

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

Records and Data
Monthly PI Reports, September 2007 – August 2008
Occupational Exposure Control Effectiveness Data, April – December 2008
Technical Requirements Manual Effluent Monitoring - Inoperability Data, April – December 2008
U1 Archived Operator Logs regarding Inoperable U1 Turbine Building (TB) Ventilation Flow, 04/15 – 04/19/08
U2 Archived Operator Logs regarding Inoperable U2 TB Wide Range Gas Monitoring Equipment Ventilation Flow, 08/14 – 08/15/08
U2 Archived Operator Logs regarding Inoperable U2 TB Ventilation Flow, 11/03 – 11/06/08

Section 4OA3: Followup of Events

1OP17, Residual Heat Removal System Operating Procedure
2OP17, Residual Heat Removal System Operating Procedure
0GP-01, Prestartup Checklist
0GP-02, Approach to Criticality and Pressurization of the Reactor
0GP-03, Unit Startup and Synchronization
0GP-12, Power Changes
0MMM-015, Operation and Inspection of Cranes and Material Handling Equipment

Section 4OA5: Other Activities

Progress Energy Specification 0BNP-C-0014, Rev. 0, Specification for Ready-Mixed Concrete
Procedure OSPP-CEM 500, Rev. 8, Installation of Concrete and Grout
Specification 9527-01-5-5, Rev. 1, Specification for Design, Testing, and Inspection of Concrete Mixes and Concrete Materials
American Concrete Institute (ACI) Manual of Concrete Practice 2006, Part 1
ACI 301-99, Specification for Structural Concrete
ACI 306.1 Standard Specification for Cold Weather Concreting
ACI 349, Code Requirements for Nuclear Safety-Related Concrete
American Society for Testing and Materials (ASTM) C 94, Standard Specification for ready-Mixed Concrete

ASTM C 172, Standard Method of Sampling Freshly Mixed Concrete
Areva Specification NUH-03-0219, Rev. 2, Design of ISFSI Basemat and Approach Slab for NUHOMS Horizontal Storage Module (HSM)
Work Procedure for Concrete Cold Joints
Engineering Change (EC) 0000067537, Rev. 2, ISFSI Pad and Apron Design
Calculation 11171-0201, Rev. 0, Reconciliation of HSM-H Seismic Soil-Structure Interaction (SSI) Acceleration for the Brunswick ISFSI
Matec Project Report 6472-08-2313, Report of Standard Test Method for Specific Gravity and Absorption of Fine Aggregate for Brunswick ISFSI Pad and Aprons
S & W Ready Mix Concrete Company, Laboratory Test Data, Field Test Data, and Material Certifications of 4000 PSI Concrete Mix Designs dated December 28, 2008
Matec Concrete Test Reports for Initial Confirmation Batch, January & February 2009
Matec Reports for Inspection of S & W Ready Mixed Concrete Production Facilities Including Fleet Inspection and Concrete Uniformity Test Report for Trucks
Concrete Pre-Placement Report for ISFSI Pad for Work Order 1360040-04, February 10, 2009
Equipment and Test Personnel Qualifications and Certifications
Material Testing and Certification Records such as Certified Mill Test Report (CMTR)
Drawings SK-67537-C-1001 to –C-1012, ISFSI Project Pad & Apron Plan Including Rebar Details

List of Acronyms and Abbreviations

ALARA	As Low As Reasonably Achievable
AR	Action requests
B117R1	Brunswick Unit 1 Cycle 17 Refueling Outage
B218R1	Brunswick Unit 2 Cycle 18 Refueling Outage
B219R1	Brunswick Unit 2 Cycle 19 Refueling Outage
BTP	Branch Technical Position
CAP	Corrective Action Program
CFR	Code of Federal Regulations
CoC	Certificate of Compliance
CRD	Control Rod Drive
CY	calendar year
DAW	dry active waste
DOT	Department of Transportation
DRD	Direct Reading Dosimeter
ED	Electronic Dosimeter
ESS	Environmental Sampling Station
H-3	tritium
HP	Health Physics
HPGe	high purity germanium
HPT	Health Physics Technician
HRA	High Radiation Area
IP	Inspection Procedure
LHRA	Locked High Radiation Area
LSA	low specific activity
LSC	liquid scintillation counter
NCR	Nuclear Condition Report
NEI	Nuclear Energy Institute
OA	Other Activities
ODCM	Offsite Dose Calculation Manual
OS	Occupational Radiation Safety
PCE	personnel contamination event
pCi/L	picocuries per liter
PCP	Process Control Program
PI	Performance Indicator
PS	Public Radiation Safety
QC	quality control
radwaste	radioactive waste
radworker	radiation worker
RCA	radiologically controlled area
REMP	Radiological Environmental Monitoring Program
Rev	Revision
RG	Regulatory Guide
RP	Radiation Protection
RWP	Radiation Work Permit
SCBA	self-contained breathing apparatus
SDCB	storm drain collector basin

SDSP	Storm Drain Stabilization Pond
SFP	Spent Fuel Pool
SSC	systems, structures, and components
TB	turbine building
TS	Technical Specification
U1	Unit 1
U2	Unit 2
UFSAR	Updated Final Safety Analysis Report
VHRA	Very High Radiation Area