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

July 30, 2007

Carolina Power and Light Company
ATTN: Mr. J. Scarola
Vice President
Brunswick Steam Electric Plant
P. O. Box 10429
Southport, NC 28461

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT - NRC INTEGRATED INSPECTION REPORT NOS. 05000325/2007003 AND 05000324/2007003

Dear Mr. Scarola:

On June 30, 2007, the US Nuclear Regulatory Commission (NRC) completed an inspection at your Brunswick Unit 1 and 2 facilities. The enclosed integrated inspection report documents the inspection findings, which were discussed on July 19, 2007, with Mr. Benjamin Waldrep 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, one self-revealing finding of very low safety significance (Green) was identified. The finding was determined to involve a violation of NRC requirements. Additionally, two licensee-identified violations which were determined to be of very low safety significance are listed in this report. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these findings as non-cited violations (NCVs), in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you contest these non-cited violations, 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 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 by Scott Shaeffer Acting For/

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

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

Enclosure: Inspection Report 05000325, 324/2007003
w/Attachment: Supplemental Information

cc w/encl: (See page 3)

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Letter to James Scarola from Randall A. Musser dated July 30, 2007

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT - NRC INTEGRATED INSPECTION REPORT NOS. 05000325/2007003 AND 05000324/2007003

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos:	50-325, 50-324
License Nos:	DPR-71, DPR-62
Report Nos:	05000325/2007003 and 05000324/2007003
Licensee:	Carolina Power and Light (CP&L)
Facility:	Brunswick Steam Electric Plant, Units 1 & 2
Location:	8470 River Road SE Southport, NC 28461
Dates:	April 1- June 30, 2007
Inspectors:	E. DiPaolo, Senior Resident Inspector J. Austin, Resident Inspector G. Kuzo, Senior Health Physicist (Section 4OA3) A. Nielsen, Health Physicist (Section 4OA3)
Approved by:	Randall A. Musser, Chief Reactor Projects Branch 4 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000325/2007003, 05000324/2007003; 03/01/07 - 06/30/07; Brunswick Steam Electric Plant, Units 1 and 2; Problem Identification and Resolution.

The report covered a 3-month period of inspection by resident inspectors. One self-revealing Green non-cited violation (NCV) was identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). 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.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. A self-revealing non-cited violation of 10CFR50, Appendix B, Criterion XVI, Corrective Action, was identified for failing to incorporate operating experience into appropriate precautions and operating limitations for single recirculation loop operation into plant procedures and training. As a result, Unit 2 experienced an automatic reactor scram on December 25, 2006 due to actuation of the Neutron Monitoring Oscillation Power Range Monitors while in single recirculation loop operation.

  The finding was more than minor because it was associated with equipment performance and affected the Initiating Events Cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety function during power operations. The finding was assessed using the Significance Determination Process for Reactor Inspection Findings for At-Power Situations and determined to be of very low safety significance (Green) because, although the finding contributed to the likelihood of a reactor trip, it did not contribute to the likelihood that mitigation equipment or functions would not be available. This finding has a crosscutting aspect in the area of Problem Identification and Resolution, specifically because the licensee did not implement appropriate changes to plant procedures and training programs to address operating experience that was reviewed (Section 4OA2.2).

B. Licensee Identified Violations

Violations of very low safety significance which were identified by the licensee have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. The violations are listed in Section 4OA7.

REPORT DETAILS

Summary of Plant Status

Unit 1

Unit 1 began the inspection period operating at full power. On April 1, Unit 1 completed a Technical Specification-required shutdown due to emergency diesel generator (EDG) #4 being out of service for the allowed outage time. EDG #4 was removed from service on March 25 for planned maintenance. Multiple equipment issues discovered during post-maintenance testing resulted in the inability to return EDG #4 to an operable status within the seven-day completion time. Following returning the EDG to an operable status, operators commenced a reactor startup on April 5. Full power was achieved on April 10. On April 11, a planned downpower was performed to approximately 70 percent to facilitate a control rod improvement and to perform secondary plant maintenance. Full power was achieved later that day. A planned downpower to approximately 53 percent was performed on May 18 to facilitate reactor feed pump testing, main turbine and main steam valve testing, and to perform secondary plant maintenance. Full power was achieved on May 21. On June 15, a planned downpower to approximately 70 percent was performed to facilitate main turbine and main steam valve testing and to perform a control rod sequence exchange. Power was returned to full on June 17. On June 28, operators reduced power to approximately 73 percent due to rising bearing temperatures on the B circulating water intake pump which necessitated that the pump be removed from service. Following the completion of pump repairs, full power was achieved on June 29 where power remained for the duration of the inspection period.

Unit 2

At the beginning of the inspection period, Unit 2 was in Mode 4 (Cold Shutdown) with preparations being made to perform reactor coolant system hydrostatic testing. On April 16, operators commenced a reactor startup. Full power was achieved on April 21. On May 26, a planned downpower to approximately 53 percent was performed to facilitate temporary leak repairs on the main turbine electro-hydraulic control system. Full power was achieved on May 27. On June 29, operators performed a planned downpower to approximately 70 percent to facilitate reactor feed pump maintenance, main turbine valve testing, and to perform a control rod sequence. Full power was achieved on June 30 where it remained for the duration of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors reviewed the licensee’s preparations for severe weather conditions prior to hurricane season and hot weather. The inspectors reviewed the results of multidiscipline-attended preparation meetings and reviewed the station's procedures for severe weather warnings (i.e., hurricanes). The inspectors toured and reviewed a sampling of design features (e.g., missile shields, severe weather doors, sumps) of the nuclear service water and emergency diesel generator buildings (1 adverse weather sample of 2 systems) to verify that they would remain functional when challenged by adverse weather. Documents reviewed are listed in the Attachment.

To assess the licensee's ability to identify and correct problems, the inspectors reviewed Action Request (AR) 238092 which identified material condition issues associated with a Diesel Generator Building severe weather door.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Partial System Walkdowns

a. Inspection Scope

The inspectors performed three partial walkdowns of the below-listed systems to verify that the systems were correctly aligned while the redundant train or system was inoperable or out-of-service (OOS) or, for single train risk significant systems, while the system was available in a standby condition. The inspectors assessed conditions such as equipment alignment (i.e., valve positions, damper positions, and breaker alignment) and system operational readiness (i.e., control power and permissive status) that could affect operability. The inspectors verified that the licensee identified and resolved equipment alignment problems that could cause initiating events or impact mitigating system availability. The inspectors reviewed Administrative Procedure ADM-NGGC-0106, Configuration Management Program Implementation, to verify that available structures, systems or components (SSCs) met the requirements of the configuration control program. Documents reviewed are listed in the Attachment.

- Emergency diesel generator (EDG) #3 when EDG #4 was OOS for unplanned maintenance on April 1, 2007
- Unit 2 high pressure coolant injection (HPCI), a risk significant system, during startup and heatup activities following Refueling Outage B21BR1 on April 16, 2007
- EDGs #1 and #2 when EDG #4 was OOS for planned maintenance on June 26, 2007

To assess the licensee's ability to identify and correct problems, the inspectors reviewed the following Action Requests (ARs):

- AR 228694, Unit 1 HPCI system oil filter differential alarm during weekly testing
- AR 229898, Unit 2 reactor core isolation cooling (RCIC) system barometric condenser vacuum pump failure
- AR 237360, Temporary scaffold placement issues in 480 volt emergency bus rooms
- AR 232906, Unit 2 division 1 backup nitrogen pressure reduction due to system leak
- AR 234482, High starting air pressure on EDG #4

.2 Detailed Equipment Alignment

a. Inspection Scope

The inspectors performed a complete walkdown of the accessible portions of the Unit 2 reactor core isolation cooling system. The inspectors focused on verifying adequate material condition and correct system alignment. The inspectors reviewed the Technical Specifications (TS), operating procedures, and the Updated Final Safety Analysis Report. The inspectors held discussions with the applicable plant personnel to review system status including a review of open system modifications and temporary modifications. The inspectors reviewed open work requests for the system, operator work-arounds, and open adverse conditions or ARs to ensure that the impact on equipment functionality was properly evaluated. The inspectors reviewed the documents listed in the Attachment.

b. Findings

No findings of significance were identified.

1R05 Fire Area Walkdowns

a. Inspection Scope

The inspectors reviewed ARs and work orders (WOs) associated with the fire suppression system to confirm that their disposition was in accordance with Administrative Procedure 0AP-033, Fire Protection Program Manual. The inspectors reviewed the status of ongoing surveillance activities to verify that they were current to support the operability of the fire protection system. In addition, the inspectors observed the fire suppression and detection equipment to determine whether any conditions or deficiencies existed which would impair the operability of that equipment. The inspectors toured the following eight areas important to reactor safety and reviewed the associated prefire plans to verify that the requirements for fire protection design features, fire area boundaries, and combustible loading were met. The inspectors reviewed Plant Operating Manual, Volume XIX, Prefire Plan 0PFP-DG, Diesel Generator Building Prefire Plans in preparing for the inspection.

- Diesel Generator Cells #1, #2, #3, and #4, 23 foot elevation (4 areas)
- 480 Volt Emergency Control Center Room E5, E6, E7, and E8, 23 foot elevation (4 areas)

To assess the licensee's ability to identify and correct problems, the inspectors reviewed the following ARs:

- AR 227614, Appendix R safe shutdown procedure enhancement associated with shutdown cooling isolation valve operation
- AR 229910, Alternate safe shutdown impairment due to condensate storage tank unavailable due to tank recoating activities

b. Findings

No findings of significance were identified.

1R06 Flood Protection

a. Inspection Scope

The inspectors performed a walkdown of the Emergency Diesel Generator Building to verify that internal flood protection features were consistent with the licensee's internal flooding analysis as described in UFSAR Section 3.4.2, Protection From Internal Flooding. The inspectors reviewed the effects of postulated piping failures for the area to verify that analysis assumptions and conclusions were based on the current plant configuration. The internal flooding design features and equipment for coping with internal flooding were also inspected. The walkdown included sources of flooding and drainage, sump pumps, level switches, watertight doors, curbs, pedestals and equipment mounting. The inspectors reviewed the procedures for coping with internal flooding.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors observed licensed operator performance and reviewed the associated training documents during simulator evaluated scenarios for training cycle 2007-02. The simulator observations and review included evaluations of emergency operating procedure and abnormal operating procedure utilization. The inspectors reviewed Procedure 0TPP-200, Licensed Operator Continuing Training Program, to verify that the program ensures safe power plant operation. Simulator sessions were observed on May 16, 2007. The scenarios tested the operators' ability to respond to secondary plant failures, loss of emergency power, and accidents. The inspectors reviewed operator activities to verify consistent clarity and formality of communication, conservative decision-making by the crew, appropriate use of procedures, and proper alarm response. Group dynamics and supervisory oversight, including the ability to properly identify and implement appropriate TS actions, regulatory reports, and notifications, were observed. The inspectors observed instructor critiques and preliminary grading of the operating crews and assessed whether appropriate feedback was planned to be provided to the licensed operators.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

For the two equipment issues described in the ARs listed below, the inspectors reviewed the licensee's implementation of the Maintenance Rule (10 CFR 50.65) with respect to the characterization of failures, the appropriateness of the associated Maintenance Rule a(1) or a(2) classification, and the appropriateness of the associated a(1) goals and corrective actions. The inspectors reviewed the work controls and work practices associated with the degraded performance or condition to verify that they were appropriate and did not contribute to the issue. The inspectors also reviewed operations logs and licensee event reports to verify unavailability times of components and systems, if applicable. Licensee performance was evaluated against the requirements of Procedure ADM-NGGC-0101, Maintenance Rule Program.

- AR 227912, EDG #4 unable to accept full load
- AR 232550, Increased differential pressure/reduced flow on Unit 1 A residual heat removal heat exchanger

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 implementation of 10 CFR 50.65 (a)(4) requirements during scheduled and emergent maintenance activities, using Procedure 0AP-025, BNP Integrated Scheduling and Technical Requirements Manual 5.5.13, Configuration Risk Management Program. The inspectors reviewed the effectiveness of risk assessments performed due to changes in plant configuration for maintenance activities (planned and emergent). The review was conducted to verify that, upon unforeseen situations, the licensee had taken the necessary steps to plan and control the resultant emergent work activities. The inspectors reviewed the applicable plant risk profiles, work week schedules, and maintenance WOs for the following five conditions:

- WO 133560, Elevated risk due to outages on emergency buses E3 and E7 on April 7-8, 2007 (planned)
- AR 230139, Unit 2 RCIC system OOS due to tripping on low suction pressure on April 18, 2007 (emergent)
- WO 1028152, Elevated risk due to cleaning the Unit 1 A RHR heat exchanger on May 21-24, 2007 (planned)
- WO 139621, Unit 2 elevated risk due to EDG #4 OOS on June 25-28, 2007 (planned)
- AR 237957, Unit 1 entered elevated risk condition due to unavailability of HPCI and RCIC systems in the pressure control mode because of disabled condensate storage tank level switches (planned)

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the operability evaluations associated with the five issues documented in the ARs listed below, which affected risk significant systems or components, to assess, as appropriate: 1) the technical adequacy of the evaluations; 2) the justification of continued system operability; 3) any existing degraded conditions used as compensatory measures; 4) the adequacy of any compensatory measures in place, including their intended use and control; and 5) where continued operability was considered unjustified, the impact on any TS limiting condition for operation and the risk significance. In addition to the reviews, discussions were conducted with the applicable system engineer regarding the ability of the system to perform its intended safety function.

- AR 233463, RCIC system operability concerns due to non-safety-related power supply to keepfill pumps
- AR 229788, Unit 2 HPCI main pump-to-speed reducer coupling issues
- AR 229771, Unit 2 RCIC system turbine lubricating oil filter leak
- AR 230139, Unit 2 RCIC system suction pressure/flow fluctuations on system startup.
- AR 232550, Increase in measured differential pressure on Unit 1, a residual heat removal heat exchanger during surveillance testing

To assess the licensee's ability to identify and correct problems, the inspectors reviewed AR 228964 concerning the discovery of high particulates in the EDG #3 saddle fuel oil tank.

b. Findings

No findings of significance were identified.

R19 Post-Maintenance Testing

a. Inspection Scope

For the five maintenance activities listed below, the inspectors reviewed the post-maintenance test procedure and witnessed the testing and/or reviewed test records to confirm that the scope of testing adequately verified that the work performed was correctly completed. The inspectors verified that the test demonstrated that the affected equipment was capable of performing its intended function and was operable in accordance with TS requirements. The inspectors reviewed the licensee’s actions against the requirements in Procedure 0PLP-20, Post Maintenance Testing Program.

- WO 1041507, Replace high fuel oil tank level (HCLR) control relay on EDG #4
- WO 1045570, Replace Unit 2 RCIC system lubricating oil filter
- WO 192495, Replace EDG #4 governor controls
- WO 137350, Repair Unit 2 RCIC system test return valve (2-E51-F022)
- WO 974565, Replace Unit 2 HPCI system main pump outboard seal

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities

Unit 2 Refueling Outage B218R1

a. Inspection Scope

The inspectors evaluated Unit 2 Refueling Outage (RFO) B218R1 activities which commenced on March 2, 2007. At the start of the inspection, Unit 2 was in Mode 4 (Cold Shutdown) with preparations being made for startup. Documents reviewed are listed in the Attachment. The following specific areas were reviewed during the inspection period:

Licensee Control of Outage Activities. The inspectors reviewed configuration changes due to emergent work and unexpected conditions were controlled in accordance with the outage risk control plan. The inspectors reviewed the following specific items, as specified:

- Decay Heat Removal and Reactor Coolant System Instrumentation. The inspectors reviewed decay heat removal procedures and observed decay heat removal systems’ parameters to verify proper removal of decay heat. The inspectors also conducted main control room panel walkdowns and walked down portions of the systems in the plant to verify system availability and to confirm that no work was ongoing that might prevent system use for decay heat removal.

- Reactivity Control. The inspectors observed licensee performance to verify that reactivity control was conducted in accordance with procedures and TS requirements. The inspectors conducted a review of outage activities and risk profiles to verify activities that could cause reactivity control problems were identified.

Monitoring of Heatup and Startup Activities. The inspectors reviewed to verify, on a sampling basis, that TS, license conditions, and other requirements for mode changes were met prior to changing modes or plant configurations. The inspectors performed a walkdown of containment to verify that debris, which could affect performance of the emergency core cooling suction strainers, had been appropriately removed.

Crane and Heavy Lift Inspection. The inspectors reviewed the reactor building crane per the guidance of Operating Experience (OpESS) FY2007-03, Crane and Heavy Lift Inspection, Supplemental Guidance for IP-71111.20. The review consisted of verifying that the design basis for the crane was single-failure proof, verifying that the licensee has a preventive maintenance program in place, to confirm that testing and inspection procedures are completed just prior to crane use, and verify that the reactor vessel head lift procedure conforms to heavy load handling commitments.

Identification and Resolution of Problems. The inspectors reviewed ARs to verify that the licensee was identifying problems related to refueling outage activities at an appropriate threshold and entering them in the corrective action program. The inspectors attended AR review meetings throughout the refueling outage to verify appropriate prioritization of planned resolution of deficiencies discovered during the outage. The inspectors reviewed the following issues identified during the outage to verify that the appropriate corrective actions were implemented:

- AR 226209, Shutdown cooling suction valve (E11-F009) failed to stroke open following maintenance
- AR 231448, Electro-hydraulic control sump level lowering trend following unit startup
- AR 224408, Shutdown cooling suction valve (2E11-F009) failed to open when placing shutdown cooling in service
- AR 237575, As-found setpoints for 4 safety/relief valves high out-of-specification

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

.1 Routine Surveillance Testing

a. Inspection Scope

The inspectors either observed surveillance tests or reviewed test data for the four risk-significant SSC surveillances listed below to verify the tests met TS surveillance requirements, UFSAR commitments, in-service testing (IST) 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.

- Periodic Test 0PT 12.2.D, Number 4 Diesel Generator Monthly Load Test, performed on April 1, 2007
- Periodic Test 0PT 8.1.4a, Residual Heat Removal Service Water Operability Test - Loop A, performed on Unit 1 on April 20, 2007
- Periodic Test 0PT 8.2.2b, Low Pressure Coolant Injection/Residual Heat Removal System Operability Test - Loop B, performed on Unit 2 on May 15, 2007
- Periodic Test 0PT-13.1, Reactor Recirculation Jet Pump Operability Test, performed on Unit 1 on June 11, 2007

To assess the licensee's ability to identify and correct problems, the inspectors reviewed the following ARs:

- AR 229267, EDG test/blocking relay inhibit function not properly tested
- AR 234689, HPCI system pump discharge check valve testing issues
- AR 223697, Thermography was not performed on EDG #3 energized control relays during operability run

b. Findings

No findings of significance were identified.

.2 In-service Surveillance Testing

a. Inspection Scope

The inspectors reviewed the performance of Periodic Test 0PT-10.1.1, Reactor Core Isolation Cooling System Operability Test, performed on Unit 2 on May 14, 2007. The inspectors evaluated the effectiveness of the licensee's American Society of Mechanical Engineers (ASME) Section XI testing program to determine 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. The inspectors also assessed any applicable corrective actions taken.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed Operating Manual 0PLP-22, Temporary Changes, to assess the implementation of Engineering Change (EC) 66766, Install Instrument Snubber on Unit 2 RCIC Instrument 2E52-PSL-N006 (pump section low pressure switch). The inspectors reviewed the EC to verify that the modification did not affect the functional capability of the EDG, that the modification was properly installed, and appropriate post-installation testing was performed.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation

a. Inspection Scope

The inspectors observed a site emergency preparedness training drill/simulator scenario conducted on May 31, 2007. The inspectors reviewed the drill scenario narrative to identify the timing and location of classifications, notifications, and protective action recommendations development activities. The inspectors evaluated the drill conduct from the control room simulator, technical support center, and the emergency operations facility. During the drill, the inspectors assessed the adequacy of event classification and notification activities. The inspectors observed portions of the licensee's post-drill critiques at the technical support center and emergency operating facility. 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. To assess the ability of the licensee to identify and correct problems, the inspectors reviewed the following corrective action documents that were generated as a result of the drill:

- AR 234914, Emergency response organization Team 3 did not properly demonstrate personnel accountability and evacuation of non-essential personnel
- AR 234932, Technical support center Team 3 weaknesses
- AR 238807, Nuclear Assurance identified the weaknesses associated with emergency preparedness drill/exercise critique results

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

a. Inspection Scope

The inspectors sampled licensee data for the performance indicators (PIs) listed below. To verify the accuracy of the PI data reported during the period reviewed, PI definitions and guidance contained in NEI 99-02, Regulatory Assessment Indicator Guideline, Rev. 4 were used to verify the basis for each data element.

Reactor Safety Cornerstone

The inspectors sampled licensee submittals for the Units 1 and 2 PIs listed below for the period April 2006 through March 2007.

- Safety System Functional Failures
- Mitigating System Performance Index-Cooling Water Support System
- Mitigating System Performance Index-Emergency AC Power System

A sample of plant records and data was reviewed and compared to the reported data to verify the accuracy of the PIs. The licensee's corrective action program records were also reviewed to determine if any problems with the collection of PI data had occurred. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

.1 Routine Review of ARs

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

.2 Annual Sample Review

a. Inspection Scope

The inspectors performed an in-depth annual sample review of the below-listed issues as documented in licensee corrective action documents to verify that conditions adverse to quality were addressed in a manner that was commensurate with the safety significance of the issue. The inspectors reviewed the actions taken to verify that the licensee had adequately addressed the following attributes:

- Complete, accurate, and timely identification of the problem
- Evaluation and disposition of operability and reportability issues
- Consideration of previous failures, extent of condition, generic or common cause implications
- Prioritization and resolution of the issue commensurate with the safety significance
- Identification of the root cause and contributing causes of the problem
- Identification and implementation of corrective actions commensurate with the safety significance of the issue

The inspectors reviewed the following issues:

- AR 217345, Unit 2 automatic reactor scram on Neutron Monitoring System Oscillation Power Range Monitors on December 25, 2006
- AR 224933, Unit 2 HPCI system main pump outboard radial bearing found damaged

b. Findings

Introduction

A self-revealing Green non-cited (NCV) was identified for failure to incorporate appropriate precautions and operating limitations for single loop operation into plant procedures and training based on the review of applicable operating experience.

Description

On December 25, 2006, while operating at approximately 64 percent power in single recirculation loop operation, Unit 2 experienced an automatic reactor scram due to tripping channels 2 and 4 of the Neutron Monitoring System Oscillation Power Range Monitors (OPRMs). Single loop operation was entered on December 24, 2006 due to a degraded condition of the A recirculation pump seal which necessitated the removal of the pump from service. The inspectors reviewed the cause of the scram and the operating experience associated with the OPRMs.

The OPRMs were installed at Brunswick to address the generic boiling water reactor issue of thermal-hydraulic instability power oscillations. The system is designed to detect unstable power oscillations that have the potential to occur while operating at relatively high power-to-core flow ratios. The system monitors 24 OPRM core cells which receive input from the neutron monitoring local power range monitors. The local power range monitor signals are combined to produce a cell reference value. The reference values are processed by three separate algorithms that test for neutron flux oscillations; Period Based Detection Algorithm, Amplitude Based Algorithm, and Growth

Rate Algorithm. The actuation of the OPRMs which caused the scram on December 25 was due to exceeding the setpoint of the instruments' Growth Rate Algorithm.

After entering single loop operation on December 24, the decision was made to raise reactor power to the maximum allowed in single loop operation. Control room operators received approximately 25 OPRM alarms (OPRM Period Based Algorithm/Confirmation Density Algorithm and OPRM Upscale Trip) randomly but the alarms did not lock in. Past experience has been that these alarms periodically occur when at reduced power so operators did not consider them unusual. Additionally, numerous process computer event log alarms were received OPRM trips, including Cell Growth Rate warnings. These alarms are not audible and are not routinely monitored. These alarms were later determined to be caused by "reactor power noise". In a boiling water reactor neutron flux is constantly changing due to the boiling regime in the core. These changes are indicated on the average power range monitors and local power range monitors as fluctuations. These fluctuations in power are known as "reactor power noise" and are more pronounced in single loop operation.

Based on consultation with the license' s fuel vendor, the licensee concluded that the trip was not caused by thermal-hydraulic instability. A review of plant data indicated that recirculation flow began to oscillate causing changes in reactor power just prior to the scram. The timing of flow fluctuations coupled with "reactor power noise" was such that the trip setpoint of the Growth Rate Algorithm was exceeded which resulted in the reactor scram.

The licensee determined that the cause of the event was due to inadequate incorporation of operating experience into plant procedures and training. Two prior operating experience documents were received and reviewed by the licensee concerning single loop operation. The first was a Hatch plant event. Operators noted fluctuations in recirculation jet pump flows, reactor level, and average power range noise when increasing power during single loop operation. It was determined that variations in operating parameters during single loop operation can be expected to be as high as twice those observed during normal two loop operation. This licensee determined that no action was needed since the concern was with Period Based Detection Algorithm and the Hatch setpoints were more conservative than Brunswick's.

The second event was forwarded to the licensee by the Boiling Water Reactor Owner's Group (BWROG) in a letter (BWROG-05011), dated April 14, 2005. This letter cautioned plants, that have implemented OPRM systems for stability protection, to the increased potential for OPRM trip signal generation during single loop operation. The letter discussed a Peach Bottom event when an OPRM trip signal was received. The OPRMs trips at that plant were not activated so no reactor scram occurred. The OPRM trip was determined to be due to the increased "reactor power noise" that is experienced in single loop operation. This operating experience was handled informally outside of the Brunswick Operating Experience system. A briefing paper outlining the event was distributed to Control Room Operators and Reactor Engineers. The paper recognized that Brunswick may be vulnerable to a reactor trip from the OPRM during operation in single loop and recommended interim actions to minimize time spent in single loop and to maximize operator awareness of the issue. The operating experience was incorporated into the Reactor Engineers Shift Turnover, however, the information was incorporated neither in the Operations procedures nor the lessons learned database.

The inspector concluded that the industry operating experience and BWROG operating experience did not result in appropriate precautions and operating limitations for single loop operation being incorporated into plant procedures and training. Review of the operating experience identified the need for action by Operations and Reactor Engineering, however, the followup was handled informally and outside the formal operating experience review process.

Analysis

The failure to incorporate operating experience into appropriate precautions and operating limitations for single loop operation into plant procedures and training is a performance deficiency. As a result, Unit 2 experienced an automatic reactor scram on December 25, 2006 due to actuation of the Neutron Monitoring Oscillation Power Range Monitors while in single loop operation. The finding was more than minor because it was associated with equipment performance and affected the Initiating Events Cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety function during power operations. The finding was assessed using the Significance Determination Process for Reactor Inspection Findings for At-Power Situations and determined to be of very low safety significance (Green) because, although the finding contributed to the likelihood of a reactor trip, it did not contribute to the likelihood that mitigation equipment or functions would not be available. This finding has a crosscutting aspect in the area of Problem Identification and Resolution, specifically because the licensee did not implement appropriate changes to plant procedures and training programs to address operating experience that was reviewed. This finding is in the licensee's corrective action program (CAP) as AR 217345.

Enforcement

10CFR50, Appendix B, Criterion XVI, Corrective Action, requires, in part, that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. Contrary to Criterion XVI, the licensee identified the need for actions to address operational concerns during single loop operation based on review of BWROG Letter 05011, Increased Potential for OPRM Trip in Single Loop Operation, dated April 14, 2005, but failed to incorporate appropriate corrective actions. As a result, Unit 2 experienced an automatic reactor scram on December 25, 2006 due to actuation of the Neutron Monitoring Oscillation Power Range Monitors while in single loop operation. Because the finding is of very low safety significance and has been entered into the CAP (AR 217345) this finding is being treated as an NCV, consistent with Section VI.A of the Enforcement Policy: NCV 05000324/2007003-01, Failure to Incorporate Operating Experience into Plant Procedures and Training.

.3 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a review of the licensee's CAP and associated documents to identify trends that could indicate the existence of a more significant safety issue. The review was focused on repetitive equipment issues but also considered the results of frequent inspector CAP item screening, licensee trending efforts, and licensee human performance results. The review considered the period of January through June 2007. The review further included issues documented outside the normal CAP in major equipment lists, repetitive and/or rework maintenance lists, operational focus list, control room deficiency list, outstanding work order list, quality assurance audit/surveillance reports, key performance indicators, and self-assessment reports. The inspectors compared and contrasted their results with the results contained in the Brunswick Plant CAP Rollup and Trend Analysis report for the 1st quarter 2007. Corrective actions associated with a sample of the issues identified in the licensee's trend reports were reviewed for adequacy. The inspectors also evaluated the reports against the requirements of the licensee's CAP as specified in Nuclear Generation Group Standard Procedure CAP-NGGC-0200, Corrective Action Program, and 10 CFR 50, Appendix B.

b. Assessment and Observations

The inspectors performed a follow-up of the declining trend identified in NRC Inspection Report 05000325,324/2006005 associated with EDG reliability. With the reporting of Mitigating System Performance Index (MSPI) data following the second quarter 2006, the MSPI for emergency AC power was White primarily due to unreliability. The inspector noted that, although the value for the indicator has improved due to historical failures no longer affecting the MSPI, EDG #2 experienced a failure on February 19, 2007. This was due to the failure of a control system relay. The inspectors concluded that EDG reliability continues to be an area of challenge for the licensee. The inspector noted that EDG reliability will be an area of focus for a team that will conduct an inspection in accordance with Inspection Procedure 95002 due to Unit 1 being in the Degraded Cornerstone column of the NRC's Action Matrix as of the first quarter of 2007.

4OA3 Event Follow-up

.1 Personnel Performance during Plant Evolutions

a. Inspection Scope

The inspectors reviewed operator actions to perform a TS required shutdown of Unit 1 on April 1, 2007 due to EDG #4 being inoperable for 7 days. To assess operator performance during the transient, the inspectors reviewed operator logs, plant computer data, and observed operator actions.

b. Findings

No findings of significance were identified.

.2 (Closed) Licensee Event Report 05000324/2007001: Operation Prohibited by Technical Specification 3.3.1.2, “Source Range Monitor Instrumentation”.

On March 26, 2007, while in Mode 5 (Refueling) operators were performing post-maintenance testing of control rod position indicators which necessitated withdrawing and re-inserting control rods. Technical Specification 3.3.1.2 requires that there be an operable sources range monitor in the core quadrant where core alterations are being performed. Contrary to this requirement, control rod 10-35 was withdrawn a single notch in a fueled quadrant of the core where there was not an operable source range monitor. This condition was identified by a licensed operator during the performance of surveillance activities. The licensee entered the issue into the CAP as AR 227261.

This issue is greater than minor because it is associated with the human performance attribute of the Initiating Events Cornerstone and affected the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions while shutdown. This finding was considered to have very low safety significance (Green) because, using Appendix G of the Significance Determination Process, it did not constitute a finding that required quantitative assessment. The enforcement aspects of this issue are discussed in Section 4OA7. This LER is closed.

.3 (Closed) LER 05000324/2006003: Automatic Reactor Scram due to Trip of Neutron Monitoring System.

On December 25, 2006, Unit 2 experienced an automatic reactor scram while in single loop operation due satisfying the Neutron Monitoring System Oscillation Power Range Monitors Growth Rate Algorithm trip setpoint. Review of the cause of the event is documented in 4OA3.2 and resulted in a Green NCV. No other issues were identified by review of the LER. This LER is closed.

.4 (Closed) LER 05000325,324/2007002: Technical Specification Required Shutdown Due To Emergency Diesel Generator 4 Inoperability.

An April 1, 2007, Unit 1 was shutdown in accordance with TS LCO 3.8.1 due to the allowed outage time with one EDG being inoperable expiring. This requirement did not apply to Unit 2 as that unit was in Mode 4 (Cold Shutdown) in a refueling outage. The cause of the extended out-of-service time was due to multiple equipment issues discovered during the planned EDG outage. These included two failed refurbished engine governors and design issues with replacement control relays discovered only after extensive post-maintenance testing. EDG 4 was restored to an operable status on April 4, 2007. The inspector determined that no violation of regulatory requirements occurred. This LER is closed.

.5 (Closed) Unresolved Item (URI) 05000325,324/2007007-02: Repetitive Failures of EDG Allen Bradley 700DC Series Relay.

In NRC Inspection Report 05000325,324/2007007, dated March 23, 2007, an NRC inspection team identified an unresolved item (URI 05000325,324/2007007-02) to further review issues involving the licensee's corrective actions associated with EDG Allen Bradley 700DC Series Relay failures. The inspectors reviewed the licensee's priority one investigation (AR 223012) and concluded that there have been multiple Allen Bradley 700DC Series Relay failures identified over the prior four years. However, a review of licensee actions taken to address the failures did not identify any specific performance deficiencies. No violations of regulatory requirements were identified. This URI is closed.

.6 Onsite Groundwater Tritium Concentrations Exceeding NEI Voluntary Reporting Criteria.

a. Inspection Scope

On June 13, 2007, the licensee reported (Event Notification 43420) that local, county, and State of North Carolina authorities were notified regarding the identification of groundwater tritium (H-3) concentrations in shallow onsite monitoring wells which exceeded recently established Nuclear Energy Institute voluntary reporting criteria. The subject wells were established to evaluate the potential movement of H-3 from the licensee's onsite storm drain stabilization pond (SDSP) to the surrounding groundwater environs and/or to onsite structures including equipment manholes. On June 18, 2007, two Region II inspectors were dispatched to the site to review the licensee's preliminary investigation into the cause, extent and mitigation of the H-3 groundwater contamination; its potential impact on both onsite and offsite surface and groundwater environments; and preliminary dose estimates to general members of the public.

The SDSP receives liquid input from the storm drain collector basin (SDCB) which in turn receives potentially contaminated liquids from roof and surface drains located within the radiologically controlled area (RCA), and from the turbine building (TB) sump. Previous licensee investigations have determined the most significant source of H-3 to the SDCB is TB steam leakage and its subsequent condensation by the building ventilation swamp cooler and discharge via TB building sumps. The inspectors reviewed and discussed historical radionuclide sampling frequencies and H-3 analysis results for the SDSP and SDCB locations. In both systems, H-3 concentrations from routine operations have exceeded 1 E+6 picocuries per liter (pCi/l). Liquid effluent from the SDSP is monitored and batch-released in accordance with established procedures and offsite dose calculation manual (ODCM) requirements.

Groundwater Contamination Event. The inspectors reviewed and discussed selected corrective action program documents regarding events leading to the identification and reporting of onsite groundwater H-3 levels exceeding the NEI reporting criteria. On May 7, 2007, manhole (MW)-5 and MW-6 located in the vicinity of the meteorological tower and approximately 50 yards from the edge of the SDSP were sampled and radioisotopic analyses conducted in accordance with established maintenance procedures. Results indicated H-3 concentrations ranging from approximately 75,000 to 150,000 pCi/l. Additional sampling and analysis of groundwater seeping into the manholes between May 10 and May 11, 2007, identified H-3 concentrations ranging from 13,000 pCi/l to approximately 235,000 pCi/l. In addition, standing surface water located in drainage ditches adjacent to the SDSP were collected and analyses results documented H-3 concentrations ranging from 47,000 to 880,000 pCi/l.

Between May 11 and May 17, 2007, two instances of water having H-3 concentration of approximately 300,000 pCi/l in drainage ditches located immediately outside of the SDSP and flowing to the site's intake canal were reported. Between May 18, 2007 and May 25, 2007, the licensee collected and analyzed numerous samples including standing water from onsite marsh areas adjacent to the SDSP, offsite surface water from several creeks nearest the pond, and from selected areas of the Cape Fear River. In addition, water samples were collected and analyzed from a marshy area located between the SDSP and Nancy Creek. Elevated concentrations of H-3 (1494 to 432,700 pCi/L) were identified for samples collected from within the onsite marsh area during low tide conditions but were less than detection limits during high tide. No elevated H-3 concentrations were identified in water samples collected from offsite creek areas and the Cape Fear River with reported values less than analytical detection limits for tritium (~ 300 pCi/L) or slightly above background levels (~ 300- 450 pCi/L). Based on selected analyses, no gamma emitting radionculides were detected within any of the onsite and offsite water samples collected.

On June 1, 2007, the licensee informally notified responsible local, county and State officials of the onsite groundwater contamination issues but did not issue 10 CFR 50.72 report to the NRC. On June 4, 2007, the licensee initiated construction of new shallow (5-20 foot depth), intermediate (32-45 foot depth), and deep wells (approximatley 150 foot depth) adjacent to the SDSP. Preliminary results of samples for four shallow wells located between the SDSP and Nancy Creek indicated maximum H-3 concentrations exceeding approximately 500,000 pCi/l. On June 13, 2007, following the identification of detectable H-3 concentrations within samples collected from both the shallow and intermediate depth monitoring wells, the licensee notified nearby residents, and responsible local and State officials of the onsite contamination in accordance with their Communication Plan guidance. On June 14, 2007, Preliminary Notification, PNO-II-07-005 was issued by the NRC regarding the licensee's notifications.

<u>Onsite and Offsite Surface Water and Groundwater Monitoring</u>. During the week of June 18, 2007, the inspectors directly observed locations and reviewed radionuclide sample results for select shallow (5-20 foot depth) and intermediate (32-45 foot depth) wells, from standing water collected from onsite marsh areas adjacent to the SDSP, and for surface water samples collected from offsite locations. Preliminary analyses of water samples from the shallow wells indicated maximum H-3 concentrations exceeding 500,000 pCi/l . Preliminary results for wells constructed to sample the intermediate aquifer indicated H-3 concentrations ranging from detection limits to approximately 5000 pCi/l but which were less than the current EPA drinking water limit of 20,000 pCi/l. Based on discussions and review of select well hydrological characteristics, the inspectors noted that the general water flow within the intermediate and the deep

aquifers was toward the intake canal, i.e., the normal discharge point for the SDSP batch releases. From review of limited radiological analysis data from previously constructed deep wells, the inspectors noted that all radioanalytical results were less than the licensee's analytical detection limits. Radionuclide analysis for ground water samples collected from a marshy area located on licensee property between the SDSP and Gum Log Creek identified detectable H-3 concentrations ranging from 431 to 19,770 pCi/L. Results for offsite samples collected from Nancy Creek, Gum Log Branch, and select Cape Fear River locations continued to document values less than the licensee's analytical detection limit or near background levels.

The inspectors toured the SDSP and well locations, and directly observed sampling of groundwater from previously established deep wells. In addition, the inspectors toured offsite surface water sample locations on Nancy Creek, Gum Log Branch, Walden Creek, and the Cape Fear River, and observed locations of nearby residences. Approximately, eight water samples from monitoring wells, and surface water sample locations were collected and prepared for split sample analysis, if needed, among the NRC, State of North Carolina Department of Environment and Natural Resources, and licensee representatives.

<u>SDSP Source-Term Reduction and Leak Mitigation</u>. Licensee actions to identify and mitigate the primary source of radionuclide inputs into the the SDCB, and ultimately the SDSP were reviewed and discussed. In addition, the inspectors discussed and directly reviewed preliminary corrective actions to mitigate any inadvertent unmonitored liquid releases from the SDSP into adjacent drainage areas.

Preliminary licensee investigations have determined the most significant source of H-3 in the SDCB to be from TB steam leakage and its subsequent condensation by the building ventilation swamp cooler and discharge *via* a building sump. Review of selected radionuclide concentration data for liquid samples collected from June 2006 through June 2007, indicated H-3 concentrations ranging from 1 E+ 6 to 7 E + 7 pCi/l within the SDCB and from 1 E+5 to 1 E+ 6 pCi/l within the SDSP. Using TB airborne release permit radionuclide concentration data, the inspectors calculated that H-3 concentrations in condensate from building swamp chiller operations could readily exceed 1 E 6 pCi/l for input into the SDCB. The inspectors noted and verified that as of June 10, 2007, all condensate discharges from the TB sump were re-routed to the site's radioactive waste system for processing and release. Review of liquid effluent grab and composite sample data indicated that, in general, this action resulted in a decrease in SDCB H-3 concentrations from 1.4 E-3 to 8.0 E-5 microcuries per milliliter (µCi/ml) between June 4, 2007, and June 19, 2007. From discussions with licensee representatives and radioactive waste system engineers, the inspectors noted that the licensee was continuing to evaluate potential H-3 inputs into SDCB including roof and surface drains within the RCA, leaching of residual contamination from the basin concrete, previous spills within the area, or leaks from other contaminated piping or structures. The inspectors reviewed leak test surveillance records for the Unit 1 and Unit 2 radioactive waste discharge lines for the previous four years and noted that no leaks have been detected. The licensee is continuing to monitor concentrations of H-3 in the SDCB and SDSP facilities to evaluate the effectiveness of current and future actions to mitigate source term input into the systems.

The inspectors also reviewed and discussed their evaluation of potential unmonitored releases and corrective actions regarding the observed flow in drainage ditch areas adjacent to the SDSP and its subsequent discharge to the intake canal. Immediate actions included implementation of ODCM liquid release permit methods to assess offsite doses. Subsequently, the licensee constructed containment catch basins to collect any inadvertent unmonitored leakage from the SDSP into in drainage ditches on the western and southern side of the SDSP. All liquids collected within these basins are currently returned to the pond for proper release to the intake canal in accordance with ODCM criteria. Licensee representatives were continuing their evaluation of the potential leakage from the SDSP and its impact on the surrounding onsite and offsite environs.

Occupational and Public Dose Evaluations. Based on review of CAP documents, calculations, and discussions with responsible staff, the inspectors evaluated initial and ongoing actions for monitoring and evaluating occupational doses to workers and offsite individuals potentially impacted by the contaminated groundwater. To date, occupational doses were negligible. To calculate offsite doses as a result of the ingestion pathway, the inspectors noted that the licensee used H-3 concentrations data for water sampled in the marsh area surrounding the SDSP, tidal flushing of the contaminated water into nearby creeks, and the subsequent ingestion of fish and shellfish by members of the general public. Based on Regulatory Guide 1.109 methodology and assumptions associated with eating fish, the licensee calculated a maximum dose of 5. 9 E-2 millirem per year (mrem/y) to an offsite individual consuming fish. At the time of the inspection, licensee actions to evaluate offsite doses as a result of evaporation from the pond were not complete.

b. Findings

An unresolved item (URI) was identified regarding the adequacy of licensee surveys (evaluations) required by 10 CFR 20.1501(a) necessary to implement Technical Specification (TS) 5.5.4 Offsite Dose Calculation Manual (ODCM) controls used to demonstrate doses to members of the public from the abnormal releases of effluents from the SDSP are ALARA in accordance with 10 CFR 50.36a. This item is unresolved pending NRC review and evaluation of the licensee's root cause analysis regarding abnormal effluent releases from the SDSP to the onsite environment and their subsequent impact on offsite doses: URI: 05000325, 324/ 2007003-002: Review Adequacy of Licensee Monitoring and Control of SDSP Releases to Ensure Doses to Members of the Public Are Maintained ALARA.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On July 19, 2007, the resident inspectors presented the inspection results to Mr. Benjamin Waldrep and other members of his staff. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

On July 25, 2007, a regional based senior health physics inspector presented the inspection results for ground water tritium event followup to Mr. Benjamin Waldrep with an unresolved item related to the adequacy of licensee surveys (evaluations).

4OA7 Licensee Identified Violations

The following violations of very low safety significance (Green) were identified by the licensee and are violations of NRC requirements which met the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for disposition as non-cited violations (NCVs).

- Technical Specification (TS) Limiting Condition for Operation (LCO) 3.3.1.2, Source Range Monitor Instrumentation, requires suspending core alterations except for control rod insertion and initiating action to fully insert all insertable control rods in core cells containing one or more fuel assemblies when one or more required source range monitors is inoperable in Mode 5 (Refueling). Technical Specification Surveillance Requirement 3.3.1.2.2 requires that during core alterations an operable source range monitor be verified to be operable in the core quadrant where core alterations are being performed, when the associated source range monitor is included in the fueled region. Contrary to TS LCO 3.3.1.2, on March 26, 2007, control rod 10-35 was withdrawn a single notch in a fueled quadrant of the Unit 2 core without the source range monitor in the associated core quadrant operable while in Mode 5. This was identified in the licensee's CAP as AR 227261. This finding was considered to have very low safety significance (Green) because, using Appendix G of the Significance Determination Process, it did not constitute a finding that required quantitative assessment.

- 10CFR50.65(a)(4) requires that, before performing maintenance activities, the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities. Contrary to 10CFR50.65(a)(4), on June 5, 2007, the licensee removed the Unit 1 condensate storage tank level switches from service, which rendered the reactor pressure control mode of operation for the reactor core isolation cooling and high pressure coolant injection systems unavailable, and failed to assess the increase in risk from the activity. This condition existed until it was identified and corrected by the licensee on June 28, 2007. The issue was identified in the licensee's CAP as AR 237957. This finding was considered to have very low safety significance (Green) because, using Appendix K of the Significance Determination Process, the incremental core damage probability deficit (ICDPD) as a result of this issue was less than 1.0 e-6.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

G. Atkinson, Supervisor - Emergency Preparedness
L. Beller, Superintendent Operations Training
A. Brittain, Manager - Security
D. Griffith, Manager - Training Manager
J. Fergusen, Manager - ER&C
L. Grzeck, Lead Engineer - Technical Support
T. Hobbs, Plant General Manager
S. Howard, Manager - Operations
R. Ivey, Manager - Site Support Services
W. Murray, Licensing Specialist
T. Pearson, Supervisor - Operations Training
A. Pope, Supervisor - Licensing/Regulatory Programs
S. Rogers, Manager - Maintenance
J. Scarola, Site Vice President
T. Sherrill, Engineer - Technical Support
T. Trask, Manager - Engineering
J. Titrington, Manger - Nuclear Assessment Services
M. Turkal, Lead Engineer - Technical Support
B. Waldrep, Director - Site Operations
M. Williams, Manager - Operations Support
W. Murray, Licensing Specialist

NRC Personnel

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

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Open

05000325,324/ 2007003-02	URI	Review Adequacy of Licensee Monitoring and Control of SDSP Releases to Ensure Doses to Members of the Public Are Maintained ALARA. (Section 4OA3.6)

Closed

05000324/2007001	LER	Operation Prohibited by Technical Specification 3.3.1.2, “Source Range Monitor Instrumentation” (Section 4OA3.2)
05000324/2006003	LER	Automatic Reactor Scram due to Trip of Neutron Monitoring System (Section 4OA3.3)
05000325,324/2007002	LER	Technical Specification Required Shutdown Due To Emergency Diesel Generator 4 Inoperability (Section 4OA3.4)
05000325,324/2007007-02	URI	Repetitive Failures of EDG Allen Bradley 700DC Series Relay (Section 4OA3.5)

Opened and Closed

05000324/2007003-01	NCV	Failure to Incorporate Operating Experience into Plant Procedures and Training (Section 4OA2.2)

LIST OF DOCUMENTS REVIEWED

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

Plant Operating Manual (POM), Volume XXI, Abnormal Operating Procedure, 0AOP-13.0, Operation during Hurricane, Flood Conditions, Tornado, or Earthquake, Rev. 37
POM, Volume I, Administrative Instruction, 0AI-68, Brunswick Nuclear Plant Response to Severe Weather Warnings, Rev. 26
POM, Volume XIII, Plant Emergency Procedure 0PEP-02.6, Severe Weather, Rev. 10
Plant Operating Manual (POM), Volume XIII, Plant Emergency Procedure 0PEP-02.1, Initial Emergency Actions, Rev. 50

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

POM, Volume III, Operating Procedure 0OP-50.1, Diesel Generator Emergency Power System Operating Procedure, Rev. 67
POM, Volume III, Operating Procedure 2OP-16, Reactor Core Isolation Cooling System Operating Procedure, Rev. 99
POM Volume III, Operating Procedure 1OP-18, Core Spray System Operating Procedure, Rev. 44

<u>Section 1R20: Refueling and Other Outage Activities</u>

POM, Volume III, Operating Procedure 2OP17, Residual Heat Removal System Operating Procedure, Rev. 144
POM, Volume IV, General Plant Operating Procedure 0GP-01, Prestartup Checklist, Rev. 173
POM, Volume IV, General Plant Operating Procedure 0GP-02, Approach to Criticality and Pressurization of the Reactor, Rev. 86
POM, Volume IV, General Plant Operating Procedure 0GP-03, Unit Startup and Synchronization, Rev. 69
POM, Volume IV, General Plant Operating Procedure 0GP-12, Power Changes, Rev. 49
POM, Volume XII, Preventive Maintenance 0PM-CRN504, Inspection of Monorails (Single Girder), and Monorail Mounted Trolleys and Fixed Underhung Hoist, Rev. 30
POM, Volume XII, Preventive Maintenance 0PM-CRN002, Overhead Crane Checkout, Rev. 3
POM, Volume XII, Preventive Maintenance 0PM-CRN501, PM for the Fixed Gantry and Track Cranes, Rev. 24
POM, Volume XII, Special Maintenance Procedure 0SMP-RPV501, Reactor Vessel Disassembly, Rev. 7
POM, Volume XII, Maintenance Management Manual 0MMM-015, Operation and Inspection of Cranes and Material Handling Equipment, Rev. 40

Section 4OA3: Event Followup

<u>Procedures, Guidance Documents, and Manuals</u>

Offsite Dose Calculation Manual, Revision (Rev.). 25
Regulatory Compliance Instruction (ORCI) 6.1, Reportable Event Evaluation Criteria and

Processing, Rev. 21.
0E&RC-3250, Groundwater Monitoring Program, Rev. 26
0E&RC-3101, Radiological Environmental Monitoring Program, Rev. 26

Records and Data

Storm Drain Collection Basin Tritium Concentration Data (µCi/ml), July 2005 through June 19, 2007
Storm Drain Stabilization Pond Tritium Concentrations (µCi/ml), July 2005 through June 19, 2007

Non-Unit Specific Radionuclide Trending Information: June 2005 - June 2007 for the following Manhole Locations: 1 SW, 1 NW, 2MH-2SW, 2-MH-2SE, 2-MH-2NW, MW-5, MW-6. 6 SW, 2MH3, 2-MH-3SW, 2-MH-MW3, 9SW, CB-2, CB-4 East, CB-4 West, MHC4, MH-CB7 East, MH-CB7 West, MH-SY6, MH-SY7, MH-WT2, MW-2, SY-4, TS6, TSC3, W 51,

Unit 1 (U1) Radwaste discharge line leak tests, 8/22/03 and 9/6/05
Unit 2 (U2) Radwaste discharge line leak tests, 9/29/04 and 9/26/06
Dose Summation and Projection Report, Gaseous Effluents, 5/1/07 - 5/31/07

Corrective Action Program (CAP) Documents

Action Request (AR) Number (No.) 232383, 233053, and 233865; Brunswick Nuclear Plant (BNP) Action Plan for tritium in groundwater, Rev. 6
Nuclear Condition Report (NCR) 184552, Tritium groundwater monitoring program, 2/16/06
NCR 188784, Electrical Manhole 2-MH-5NW water sample contained cesium-137
NCR 204962, Tritium activity in manhole 6 SW, 08/30/06
NCR 209023, Tritium identified in well ESS-3B, 10/11/06
NCR 229054, Low level tritium identified in March environmental sample, 04/10/07
NCR 232383, Tritium in manholes MW-5 and MW-6, 05/08/07
NCR 232440, E&RC requests for manhole samples, 05/09/07
NCR 233053, Tritium release to intake canal, 05/14/07
NCR 233427, Tritium release to intake canal, 05/17/07
NCR 233857, Airborne tritium pathway, 05/22/07
NCR 233865, Storm drain stabilization pond, 05/22/07
NCR 236175, SDSP Ditch release into the intake canal, 06/12/07
NCR 236389, Tritium identified in manhole 2-MH-2SE, 6/13/07