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

July 29, 2005

Carolina Power and Light Company
ATTN: Mr. C. J. Gannon
Vice President
Brunswick Steam Electric Plant
P. O. Box 10429
Southport, NC 28461-0429

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT - NRC INTEGRATED INSPECTION REPORT NOS. 05000325/2005003 AND 05000324/2005003

Dear Mr. Gannon:

On June 30, 2005, the US Nuclear Regulatory Commission (NRC) completed an inspection at your Brunswick Units 1 and 2 facilities. The enclosed integrated inspection report documents the inspection findings, which were discussed on July 7, 2005, with Mr. Tim Cleary 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, three self-revealing findings of very low safety significance (Green) were identified. These findings were determined to involve violations of NRC requirements. However, because of their very low safety significance and because they are entered into your corrective action program, the NRC is treating these three findings as non-cited violations (NCVs) consistent with Section VI.A.1 of the NRC Enforcement Policy. Additionally, licensee-identified violations which were determined to be of very low safety significance are listed in this report. 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 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/

Paul E. Fredrickson, 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, 324/2005003
w/Attachment: Supplemental Information

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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:	05000325/2005-003 and 05000324/2005-003
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, 2005 - June 30, 2005
Inspectors:	E. DiPaolo, Senior Resident Inspector J. Austin, Resident Inspector
Approved by:	Paul Fredrickson, Chief Reactor Projects Branch 4 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000325/2005003, 05000324/2005003; April 1, 2005 - June 30,2005; Brunswick Steam Electric Plant, Units 1 and 2; Refueling Outage, Permanent Plant Modification, Problem Identification and Resolution.

The report covered a three-month period of inspection by resident inspectors. Three Green self-revealing non-cited violations were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 609, “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 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Barrier Integrity

- Green. A self-revealing non-cited violation of 10CFR50, Appendix B, Criterion III, Design Control, was identified for failure to assure that Technical Specification (TS) requirements for the feedwater and main turbine high water trip function remained operable with the introduction of a filtered time constant for reactor vessel level. As a result, instrumentation associated with TS 3.3.2.2, Feedwater and Main Turbine High Water Level Trip Instrumentation, were inoperable from April 30, 2004 for Unit 1 and April 30, 2003 for Unit 2 until the time constant filters were removed on April 10, 2005

  This finding is greater than minor because it is associated with the design control attribute of the Barrier Integrity Cornerstone and affects the cornerstone objective of providing reasonable assurance that physical design barriers (i.e., fuel cladding) protect the public from radionuclide releases caused by events. This finding is of very low safety significance because it could affect the fuel cladding, but could not effect the integrity of the reactor cooling system. The cause of this finding is identified as a performance aspect of the human performance cross-cutting area, in that the cause was attributed to a lack of sufficient questioning attitude from engineering personnel, related to the impact of a parameter change on all system output responses. (Section 1R17).

Cornerstone: Initiating Events

- Green. A self-revealing non-cited violation of Technical Specification (TS) 3.0.5., which allows some inoperable equipment, declared as such through a TS Action, to be returned to service solely for the purpose of demonstrating operability, was identified for failure to properly utilize this TS when returning a control rod to service following maintenance with Unit 1 in Mode 5 (Refueling). This resulted in the failure to meet the required actions of TS 3.9.2, Refuel Position One-Rod-Out Interlock, and TS 3.9.4, Control Rod Position Indication, with the unit in Mode 5.

The finding is greater than minor because it is associated with the equipment configuration control attribute of the Initiating Events Cornerstone and affects the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions while shutdown. This finding is of very low safety significance because, using Appendix G of the SDP, it did not constitute a finding that required quantitative assessment. The cause of this finding is a performance aspect of the human performance cross-cutting area, in that the cause was attributed to operator knowledge of the requirements of TS 3.0.5 and communication errors between Maintenance and Operations (Section 1R20).

- Green. A self-revealing non-cited violation of Technical Specification (TS) 5.4.1.a. Procedures, was identified for failure to provide adequate condensate system procedural guidance to preclude the reactor feed pumps from tripping on low suction pressure during plant operations. The inadequate procedures contributed to a Unit 2 automatic reactor scram on April 9, 2005, due to low reactor vessel level.

  The finding is greater than minor because it is associated with the procedure quality attribute of the Initiating Events Cornerstone and affects the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during power operations. This finding is of very low safety significance because, although it contributes to the likelihood of a reactor trip, it does not contribute to the likelihood that mitigation equipment or functions would be unavailable.

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. These violations and corrective actions are listed in Section 4OA7.

REPORT DETAILS

Summary of Plant Status

Unit 1 began the report period operating at full power. On April 15, the unit completed a plant shutdown to commence mid-cycle outage B115M1. The planned outage was performed to address detected leaking fuel assemblies and elevated drywell unidentified leakage. Following the completion of maintenance activities, the unit entered Mode 2 (Startup) on April 22 and Mode 1 (Power Operation) on April 23. Full power was achieved on April 26. On May 12, while emergency diesel generator (EDG) #1 was out-of-service, power was lost to emergency bus E-1 when the associated feeder breaker tripped unexpectedly. The unit commenced a plant shutdown required by Technical Specifications (TS) due to the resultant loss of function to the reactor coolant system leakage detection system and operability of emergency bus E-1. The shutdown was halted with the unit at approximately 50 percent when the cause of the feeder breaker tripping was determined and corrected. Unit 1 returned to full power on May 15 where it remained for the duration of the inspection period.

Unit 2 began the report period in Mode 4 (Cold Shutdown) having completed refueling activities for refueling outage (RFO) B217R1. Mode 2 (Startup) was entered on April 4 and Mode 1 (Power Operation) was entered on April 6. On April 9, while at 61 percent power, an automatic reactor scram occurred due to low reactor vessel water load. The low level was caused by the operating reactor feed pump tripping due to a condensate system transient during the performance of startup extended power uprate testing. The unit proceeded to Mode 4 (Cold Shutdown) to recover from reactor vessel temperature stratification following the trip and to facilitate maintenance on a leaking main system safety/relief valve. Following recovery activities, the unit entered Mode 2 (Startup) on April 11 and Mode 1 (Power Operation) on April 12. Full power was achieved on April 18. On April 20, the unit performed an unplanned downpower to approximately 65 percent to address high vibrations on the B reactor feed pump. Following replacement of the pump's impeller, the unit returned to full power on May 1. On May 9, Unit 2 performed a dispatch-directed downpower to approximately 82 percent due to an offsite coupling capacitor failure on the Delco West 230 KV Line. The unit returned to full power later that day. On June 3, the unit performed a planned downpower to approximately 53 percent to facilitate fuel leak suppression testing, valve testing, and control rod testing. Full power was achieved on June 9. An unplanned downpower to approximately 75 percent was performed on June 23 in response to the tripping of the B circulating water pump due to a high differential pressure at the intake structure cause by gracilaria (seaweed) buildup on the intake trash racks. Following intake structure cleaning the unit returned to full power later that day 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 inspector's reviewed preparations for severe weather conditions prior to hurricane season and hot weather. The inspectors observed 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 EDG buildings (1 adverse weather sample of 2 systems) to verify that they would remain functional when challenged by adverse weather. For hot weather preparations, the inspectors reviewed the below listed emergent issues associated with plant ventilation systems :

- Action Request (AR) 161542, South RHR room cooler will not start
- AR 162207, Emergency switch gear high temperature alarm
- AR 160663, Failure of spare control room air conditioning unit

Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment-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.

- Unit 1 suppression chamber equipment and systems while in Mode 5 (Refueling) and making preparation for plant startup on April 19, 2005 (risk significant system)
- EDG #2 when EDG #1 was OOS on May 10, 2005 for planned maintenance.
- Nuclear service water intake structure equipment during a period of high gracilaria flow in the intake canal on June 23-24, 2005

To assess the licensee's ability to identify and correct adverse conditions, the inspectors reviewed the licensee's actions in response to AR 156475 which was associated with a 10CFR21 report concerning the EDG governor system and AR 155267 which documented a Unit 2 drywell-to-suppression chamber vacuum breaker that failed to indicate closed.

b. Findings

No findings of significance were identified.

1R05 Fire Protection-Fire Area Walkdowns

a. Inspection Scope

The inspectors reviewed current ARs and work orders (WOs) associated with the fire suppression system to confirm that their disposition was in accordance with Procedure OAP-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. Documents reviewed are listed in the Attachment:

- Service Water Building -13' 4", 4', and 20' elevations (2 areas)
- Units 1 and 2 Cable Access Way 23' elevation (2 areas)
- Unit 1 Battery Rooms A and B 23' elevation (2 areas)
- Unit 2 Battery Rooms A and B 23' elevation (2 areas)

To assess the licensee's ability to identify and correct adverse conditions, the inspectors reviewed the licensee's actions in response to AR 159621 which documented incorrect seal material installed on the high pressure coolant injection room plugs.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

a. Inspection Scope

The inspectors observed test instrumentation setup and reviewed the test results of EDG #2 jacket water cooling heat exchanger performance testing. In performing the review, the inspectors reviewed Engineering Procedure 0ENP-2705, Service Water Heat Exchanger Thermal Performance Testing, Rev. 2. The inspectors verified acceptable heat exchanger performance by reviewing the following test attributes:

- Test acceptance criteria and results appropriately considered differences between test conditions and design conditions
- Test results have considered test instrument inaccuracies and differences
- Frequency of testing is sufficient to detect degradation prior to loss of heat removal capabilities below design basis values

b. Findings

No findings of significance were identified.

1R11 Quarterly Licensed Operator Requalification

a. Inspection Scope

The inspectors observed licensed operator performance and reviewed the associated training documents during simulator training sessions for cycle 2005-02. The simulator observation and review included an evaluation of emergency operating procedure and abnormal operating procedure utilization. The inspectors reviewed Procedure OTPP-200, Licensed Operator Continuing Training (LOCT) Program, to verify that the program ensures safe power plant operation. The scenarios tested the operators' ability to respond to reactor water cleanup leak, scram, anticipated emergency depressurization, and inventory control. The inspectors reviewed the operators 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 assessed whether appropriate feedback was planned to be provided to the licensed operators.

b. Findings

No findings of significance were identified.

1R12 Routine Maintenance Effectiveness

a. Inspection Scope

For the equipment issues described in the work documents listed below, the inspectors reviewed 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 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-NGG-0101, Maintenance Rule Program. The inspectors also reviewed deficiencies related to the work activities listed below to verify that the licensee had identified and resolved deficiencies in accordance with Procedure CAP-NGGC-0200, Corrective Action.

- AR 159241, Motor driven fire pump alternate breaker found tripped
- AR 158454, Unit 2 service water radiation monitor spike

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

The inspectors reviewed the implementation of 10 CFR 50.65 (a)(4) requirements during scheduled and emergent maintenance activities, using Procedure OAP-025, BNP Integrated Scheduling and Technical Requirements Manual (TRM) 5.5.13, Configuration Risk Management Program. The inspectors reviewed the effectiveness of risk assessments performed prior to changes in plant configuration for maintenance activities (planned and emergent). The review was conducted to verify that, upon unforseen 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 WOs for the following OOS equipment or conditions, and the documents listed in the Attachment:

- AR 158390, Unit 2 Delco West 230 KV line fault when EDG #1 was OOS on May 9, 2005 (emergent)
- AR 159108, Motor-driven fire pump OOS due to power feeder cable issues from May 12-24, 2005 (emergent)
- AR 158668, Unit 1 in Red online risk profile with emergency bus E-1 and EDG #1 OOS on May 12, 2005 (emergent)
- Engineering Change (EC) 60789, Implement Units 1 and 2 digital feedwater control system modification on June 9 and June 7, 2005, respectively (planned)

- AR 161964, Unit 2 power reduction and recovery as a result of the trip of B circulating water intake pump due to a intake structure blockage cause by gracilaria buildup on trash racks on June 23, 3005 (emergent)

b. Findings

No findings of significance were identified.

1R14 Operator Performance During Non-Routine Plant Evolutions and Events

.1 Routine Review

a. Inspection Scope

The inspectors reviewed or observed the following three unplanned non-routine events and transient operations. The inspectors reviewed operator actions to verify the response to the event or transient operations was in accordance with procedures and training. Operator logs, plant computer data, and associated operator actions were reviewed as well as the procedures listed in the Attachment.

- AR 155447, Overheating of motor control center 2XDA and entry into Abnormal Operating Procedure (AOP) 39, Loss of DC Ppower, on April 3, 2005
- AR 156020, Unit 2 reactor scram due to low reactor vessel water level on April 19, 2005
- AR 162471, Entry into AOP 13, Operation During Hurricane, Flood Conditions, Tornado, or Earthquake, due to turbine building flooding caused by storm drain system backup during heavy rains on June 29, 2005

b. Findings

No findings of significance were identified.

.2 Licensee Event Report (LER) 05000324/2005001: Compliance with Single Control Rod Withdrawal-Cold Shutdown Technical Specification.

The inspectors reviewed the LER which described personnel performance issues as a causal factor. On March 31, 2005, during equipment walkdown activities, the licensee identified that the amphenols for the directional control valves on the hydraulic control unit associated with control rod 38-47 were not removed as required by the equipment control process. The amphenols were required to be removed in order to satisfy TS 3.10.4, Single Control Rod Withdrawal-Cold Shutdown, while performing maintenance on another control rod.

The licensee entered the issue into the CAP as AR 155262. Corrective actions included revising the equipment control process to require independent verification of manipulations implemented which ensures compliance with TS or other regulatory requirements, coaching of the individuals involved in removing the amphenols on the directional control valves, and sharing of lessons learned with appropriate operations shift personnel. This finding is greater than minor because it is associated with the equipment configuration control attribute of the Initiating Events Cornerstone and affects the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions while shutdown. The finding was considered to have very low safety significance (Green) because, using Appendix G of the Significance Determination Process (SDP), it did not constitute a finding that required quantitative assessment. The enforcement aspects of the finding is discussed in 4OA7.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the operability evaluations associated with the following six issues, 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 TS limiting conditions for operations 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.

- Calculation 2B11-0033, Reactor Feed Pump 2B Lost Parts Analysis
- AR 158668, Loss of emergency bus E-1
- AR 159108, Failure of the normal fire pump feeder breaker
- AR 155746, Safety/relief valve “C” high tailpipe temperature
- AR 160368, Unit 2 torus to drywell vacuum breaker stroke time
- AR 155393, E-7 main breaker “B” phase sensor current transformer installed backwards

To assess the licensee’s ability to identify and correct adverse conditions, the inspectors reviewed the licensee’s actions in response to AR 151378 which documented an EDG load calculation discrepancy.

b. Findings

No findings of significance were identified.

1R16 Operator Work-Arounds (OWAs)

a. Inspection Scope

Selected OWAs

The inspectors reviewed the status of OWAs for Units 1 and 2 to verify that the functional capability of the system or operator reliability in responding to an initiating event was not affected. The inspectors reviewed OWA 1152 which documented the issue of the failure of the core flow computer point “WTCF” during recirculation pump transients which hindered operator ability to monitor operations near the scram avoidance region. The review was to evaluate the effect of the OWA on the operator’s ability to implement abnormal or emergency operating procedures during a transient or event conditions. The inspectors compared licensee actions to the requirements of Procedure 0OI-01.08, Control of Equipment and System Status and held discussions with operations personnel related to the OWA’s reviewed.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed a permanent plant modification documented in the three below listed ECs. The inspectors reviewed the design adequacy of the modification for material compatibility which included functional properties, environmental qualification, and seismic evaluation. The review verified that the modification was consistent with the plant’s design bases and the design assumptions. Where applicable, the review verified that modification preparation, staging, and implementation did not impair emergency/abnormal operating procedure actions and key safety functions. Post-modification testing was reviewed to confirm that operability would be established, unintended system interactions would not occur, and the testing demonstrated that modification acceptance criteria were met. Documents reviewed are listed in the Attachment.

- EC 50091, R2, Unit 2 Condensate Pump and Motor Replacement for EPUR
- EC 60789, Digital Feedwater Control System Noise Filtering to Minimize Reactor Feedpump Turbine Control Valve Oscillations (Units 1 and 2)
- Unit 1 and 2 digital feedwater control system filter time constant introduction implemented in accordance with Special Procedures 1SP-EC50095-01, Unit 1 Extended Power Uprate Digital Feedwater Control System Testing, Revision 1, and 2SP-02-201, Unit 2 Extended Power Uprate Digital Feedwater Control System Testing, Revision 2

b. Findings

Introduction.

A Green self-revealing non-cited violation (NCV) of 10CFR50, Appendix B, Criterion III, Design Control, was identified for failure to assure that TS requirements for the feedwater and main turbine high water trip function remained operable with the introduction of a filtered time constant for reactor vessel level.

Description.

On April 9, 2005, Unit 2 experienced an automatic reactor scram due to low reactor vessel water level caused by of loss of feedwater. Following the loss of feedwater, operators had planned to initiate a manual scram if reactor water level lowered to a predetermined water level,171 inches, which was above the automatic scram setpoint. At the time of the automatic scram, water level was noted to be in the range of 172 to 173 inches as indicated on control room level indicators (2-C32-LI-R606A, B, and C). The licensee initiated an investigation into the apparent discrepancy between the level indicators and the automatic scram setpoint (166 inches).

The licensee determined that the apparent difference between control room indicators and the automatic scram setpoint was due to a change to a digital feedwater control system (DFWCS) software parameter. The change, implemented following the Spring 2003 refueling outage by Special Procedure 2SP-02-201, Unit 2 Extended Power Uprate Digital Feedwater Control System Testing, Revision 2, introduced a time constant filter (3 seconds) into the DFWCS output. Following startup from that outage, the licensee observed unacceptable reactor feed pump turbine control valve oscillations which were equipped with new speed governors. The time constant filter was introduced in an effort to reduce the oscillations to within acceptable manufacturer specifications. The use of the filter was, at that time, considered to be an adjustment of a pre-existing tuning parameter rather than a modification of the system design. The DFWCS time constant filter resulted in indicated level taking 5 time constants to equal actual reactor water level. Therefore, control room indicated level, which receives input from the DFWCS, lagged actual water level. The effect of the time constant filter would be greater for faster level transients than slower changes in level. Similarly, the DFWCS time constant filter was introduced on Unit 1 following the Spring 2004 refueling outage by Special Procedure 1SP-EC50095-01, Unit 1 Extended Power Uprate Digital Feedwater Control System Testing, Revision 1.

Further review determined that the level lag would also affect other DFWCS outputs. This included TS 3.3.2.2, Feedwater and Main Turbine High Level Trip Instrumentation, which is designed to protect against exceeding core thermal limits during the abnormal operating transient of a failed feed water level controller. Upon discovery on April 10, 2005, the instrumentation was declared inoperable. The licensee promptly removed the time constant filter later that day to restore operability of the instrumentation. The licensee reported the issue in LER 05000325,324/2005-001, Operation Prohibited by

Technical Specification-Inoperable Feedwater and Main Turbine High Water Level Trip, dated June 8, 2005 (see Section 4OA3.5).

Analysis.

The failure to assure that the TS function for the feedwater and main turbine high water level trip instrumentation remained operable with the introduction of a time constant filter for reactor vessel level, is greater than minor because it is associated with the design control attribute of the Barrier Integrity Cornerstone and affects the cornerstone objective of providing reasonable assurance that physical design barriers (i.e., fuel cladding) protect the public from radionuclide releases caused by events. The finding was considered to have very low safety significance (Green) because although it could affect the integrity of the fuel cladding, it could not affect the integrity of the reactor cooling system (RCS). Additionally, during the time period in question, the feedwater controller failure was a non-limiting event due to plant specific configuration (e.g., main turbine bypass system was operable and sufficient margin between off-rated thermal limits and the lowest observed minimum operating minimum critical power ratio existed). The inspectors also determined that the cause of this finding is a performance aspect of the human performance cross-cutting area, in that the cause was attributed to a lack of sufficient questioning attitude from engineering personnel, related to the impact of a parameter change on all system output responses combined with a lack of detailed engineering knowledge and documentation associated with the DFWCS software.

Enforcement.

10CFR50 Appendix B, Criterion III, Design Control, requires, in part, that measures be established to assure that applicable regulatory requirements and the design basis are correctly translated into procedures. Contrary to the above, Special Procedures 1SP-EC50095-01, Unit 1 Extended Power Uprate Digital Feedwater Control System Testing, Revision 1, and 2SP-02-201, Unit 2 Extended Power Uprate Digital Feedwater Control System Testing, Revision 2, improperly changed the Units 1 and 2 DFWCS output by inserting time constant filters. This change did not assure that TS requirements for the feedwater and main turbine high water level trip functions remained operable. As a result, the feedwater and main turbine high water level trip instrumentation (included in TS 3.3.2.2), was inoperable from April 30, 2004 for Unit 1 and April 30, 2003 for Unit 2, until the time constant filters were removed on April 10, 2005, from both units' instrumentation. Because this issue is of very low safety significance and has been entered into the CAP as AR 156037, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000325,324/2005003-01, Inadequate Design Control for Digital Feedwater Control System Modification.

1R19 Post Maintenance Testing

a. Inspection Scope

For the five post maintenance tests and maintenance activities listed below, the inspectors reviewed the 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 and 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 709489, Megger 1-E1 4KV Emergency Switchgear
- WO 709449, Calibration Check of E1 Undervoltage Relay
- WO 709451, Calibration Check of E1 Undervoltage Relay
- WO 709476, Control Building HVAC Air Compressor Failed to Start
- WO 694780, SRV "C" Tailpipe temperature elevated, potential leak-by

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities

a. Inspection Scope

.1 Unit 1 Maintenance/Refueling Outage

The inspectors evaluated Unit 1 maintenance/refueling outage B115M1 activities which commenced on April 15, 2005. The planned outage was performed in order to address detected leaking fuel assemblies and elevated drywell unidentified leakage (approximately 1.8 gpm). Unit 1 completed the outage and entered Mode 1 (Power Operation) on April 23. The following specific areas were reviewed:

Outage Plan. The inspectors reviewed Brunswick Nuclear Plant Unit 1 Outage Risk Assessment for Maintenance Outage B115M1. The inspectors reviewed the outage plan to verify that the licensee had considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth.

Shutdown and Cooldown. The inspectors observed portions of the Unit 2 shutdown to enter the outage to verify that activities were in accordance with General Procedure 0GP-5.0, Unit Shutdown. The inspectors verified that the licensee monitored cooldown restrictions by performing Periodic Test (PT) Procedure 1PT-01.7, Heatup/cooldown Monitoring, to assure that TS cooldown restrictions were satisfied.

Licensee Control of Outage Activities. The inspectors observed and reviewed activities and plant conditions to verify that the licensee maintained defense-in-depth commensurate with the outage risk control plan. The inspectors reviewed the following specific items:

- Decay Heat Removal. The inspectors reviewed decay heat removal procedures and observed decay heat removal systems' parameters to verify proper removal of decay heat. The inspectors conducted main control room panel walkdowns and walked down portions of the systems in the plant to verify system availability.

- Reactivity Control. The inspectors observed licensee performance during the outage to verify that reactivity control was conducted in accordance with procedures and TS requirements.

- Inventory Control. The inspectors observed operator monitoring and control of reactor coolant temperature and level and monitored outage work and configuration control for activities that had the potential to drain the reactor vessel. This was performed to verify that they were performed in accordance with the outage risk plan.

- Electrical Power. The inspectors reviewed the following activities related to electrical power during the refueling outage to verify that they were in accordance with the outage risk plan:

    - Controls over electrical power systems and components to ensure emergency power was available as specified in the outage risk report

    - Controls and monitoring of electrical power systems and components and work activities in the power transmission yard

Refueling Activities. The inspectors reviewed refueling activities to verify fuel handling operations were performed in accordance with TS and fuel handling procedures and that controls were in place to track fuel movement. The inspectors reviewed refueling floor and plant controls to verify that the foreign material exclusion controls were established.

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

Identification and Resolution of Problems. The inspectors reviewed ARs to verify that problems related to outage activities were being identified at an appropriate threshold and entered into the CAP. The inspectors reviewed the following issues identified during the outage to verify that the appropriate corrective actions were implemented:

- AR 156640, Fuel bundle misorientation
- AR 156834, Fuel bundle scheduled for reinsert identified as a leaker

.2 Unit 2 Refueling Outage

The inspectors evaluated Unit 2 RFO B217R1 activities which commenced on March 4, 2005. At the start of the inspection period, fuel movement was complete and the unit was in Mode 4 (Cold Shutdown) and preparing for startup activities. Unit 2 entered Mode 1 (Power Operation) on April 4 to complete the outage. 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 to verify that unexpected conditions were controlled in accordance with the outage risk control plan. The inspectors reviewed the following specific items:

- 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 also walked down portions of plant systems to verify system availability and to confirm that no work was ongoing that might cause a loss decay heat removal capability.

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

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

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 CAP. The inspectors reviewed the following issues:

- AR 155746, Safety/relief valve 2-B21-F013C high tailpipe temperature following startup
- AR 156017, Loss of supplemental spent fuel pool cooling on April 8, 2005

b. Findings

Introduction.

A Green self-revealing NCV of TS 3.0.5., which allows some inoperable equipment, declared as such through a TS Action, to be returned to service, solely for the purpose of demonstrating operability, was identified for failure to properly utilize this TS when returning a control rod to service following maintenance with Unit 1 in Mode 5 (Refueling).

Description:

During Unit 1 maintenance/refueling outage B115M1 which commenced on April 15, 2005, maintenance was performed on the position indicating probe (PIP) for control rod 46-15. The maintenance rendered the PIP inoperable and was being performed in Mode 5 (Refueling). In order to allow fuel replacement activities to continue, a PIP position simulator was installed, which provided a full-in indication for control rod 46-15 to the one-rod-out interlock and control room indicators. These conditions (i.e., PIP and full-in indication to the one-rod-out interlock) required control rod 46-15 be fully inserted and disabled from control rod movement (i.e., disarmed) in order to satisfy the TS Action requirements of TS 3.9.2, Refueling Position One-Rod-Out Interlock, and TS 3.9.4, Control Rod Position Indication.

Following the completion of maintenance activities on control rod 46-15 on April 20, 2005, operators planned to perform control rod testing in accordance with PT Procedure 0PT-90.2, Friction Testing of Control Rods. A review by Operations of the work order for control rod 46-15 showed that the work was still open (i.e, testing to demonstrate operability had not yet been performed). However, subsequent discussions with Maintenance incorrectly indicated that replacement of the PIP had been completed and that the PIP position simulator had been removed. No field verification was performed to confirm the status of the work order.

Based on this information, Operations prepared to perform PT Procedure 0PT-90.2 on control rod 46-15 to demonstrate operability. However, a prerequisite of Procedure 0PT-90.2 required that a reactor subcriticality test be performed. This test involves withdrawing the highest worth control rod to verify the core remains subcritical. Reactor engineers recommended that two control rods be individually withdrawn to satisfy the requirements of Procedure 0PT-90.2.

The senior reactor operator assigned to assist in the performance of control rod testing determined that TS 3.0.5 would allow control rod 46-15 to be re-armed for the purpose of performing the prerequisite reactor subcriticality testing to establish operability of control rod 46-15. An independent check by another SRO also concluded that TS 3.0.5 allowed this. Based in this interpretation of TS 3.0.5, control rod 46-15 was re-armed, and the subcriticality test was then performed using the two control rods selected. Following that test, operators attempted to move control rod 46-15 to demonstrate it's operability. However, after several attempts, no movement was indicated. Operators concluded that the control rod was difficult to move which is not uncommon following a refueling outage. To verify the control rod drive system was functioning properly, operators stopped testing the control rod and performed a test of another control rod which was successful. After additional testing attempts of control rod 46-15 were unsuccessful, operators concluded an undetermined problem existed with the rod and terminated testing. Investigation revealed the position simulator was still installed for control rod 46-15 which rendered the associated PIP inoperable.

A review of this event by the inspectors revealed several problems. First, the interpretation of TS 3.0.5 used to re-arm control rod 46-15 for the purpose of performing the reactor subcriticality test was incorrect. TS 3.0.5 allows equipment removed from service or declared inoperable to comply with Actions, to be returned to service under administrative control <u>solely</u> to perform testing required to demonstrate its operability. At the completion of maintenance activities on control rod 46-15, it was still inoperable, with rod fully inserted and disarmed as required by TS 3.9.2, and TS 3.9.4. As such there was only one TS authorized basis for re-arming inoperable control rod 46-15 and thus administratively exiting the "disabled from control rod movement (i.e., disarmed" TS 3.9.2, and TS 3.9.4 Action statements. That basis was "<u>solely</u> to perform testing required to demonstrate its operability, as stated in TS 3.0.5; it was not to perform a reactor subcriticality test. Second, the decision to continue with manipulating other control rods after control rod 46-15 was also incorrect. Because control rod 46-15 had been re-armed, when it failed to demonstrate operability, it should have been fully inserted and disarmed as required by TS 3.9.2, and TS 3.9.4., prior to any further other control rod movement. Third, the inspectors noted that the reactor subcriticality test could have been performed using the two control rods selected, without having to re-arm control rod 46-15.

The licensee reported the issue in LER 05000325/2005-003, Inappropriate Use of Technical Specification 3.0.5 During Control Rod Manipulations, dated June 16, 2005 (see Section 4OA3.4). The licensee determined the cause of the event to be due to the misapplication of TS 3.0.5 during return-to-service testing of control rod 46-15. Specifically, TS 3.0.5 should not have been used as part of subcriticality testing, nor should have subsequent friction testing continued, without first inserting and disarming control rod 46-15.

<u>Analysis</u>:

The failure to properly utilize TS 3.0.5 when returning control rod 46-15 to service is greater than minor because it is associated with the equipment configuration control attribute of the Initiating Events Cornerstone and affects the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions while shutdown. The finding was considered to have very low safety significance (Green) using Appendix G of the SDP because it does not constitute a finding that requires a quantitative assessment. Additionally, although manipulations of control rods with the PIP simulator installed potentially allowed withdrawal of two control rods (46-15 with an inoperable PIP, and another rod) while in Mode 5, an analysis performed by the licensee showed that the reactor would have remained subcritical with sufficient shutdown margin even if control rod 46-15 and the face-adjacent control rod (worst case) were simultaneously withdrawn. The inspectors also determined that the cause of this finding is a performance aspect of the human performance cross-cutting area, in that the cause was due to operator knowledge of the requirements of TS 3.0.5 and communication errors between Maintenance and Operations as to the status of the PIP position simulator used during the maintenance activities, prior to returning control rod 46-15 to service.

Enforcement:

TS 3.0.5 requires that equipment removed from service or declared inoperable to comply with Actions may be returned to service under administrative control solely to perform testing required to demonstrate its operability. Contrary to TS 3.0.5, on April 20, 2005 with Unit 1 in Mode 5 (Refueling), a demonstration of operability testing was not performed on control rod 46-15 after it was returned to service (re-armed) following maintenance. Instead, a non-testing-related activity of three other control rods were conducted after control rod 46-15 was re-armed. This resulted in the failure to meet the required actions of TS 3.9.2 and TS 3.9.4 (control rod inserted and disarmed), with the unit in Mode 5. Because this issue is of very low safety significance and has been entered into the CAP as AR 156838, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000325,/2005003-02, Inappropriate Use of Technical Specification 3.0.5 in Mode 5 Operations.

1R22 Surveillance Testing

a. Inspection Scope

Routine Surveillance Testing

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), 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- 40.2.8, Main Steam Isolation Valve Closure Test performed on Unit 2

- WO 473430, Perform Manhole Inspection in accordance with EGR-NGGC-0351, Condition Monitoring of Structure, Rev. 12, performed on man hole 2Y
- Periodic Test 0PT-11.1.2 Automatic Depressurization System and Safety Relief Valve Operability Test for SRV “C”
- Periodic Test PT-2.3.1 Suppression Pool to Drywell Vacuum Breaker Position Check for Unit 2

Inservice Surveillance Testing

The inspectors reviewed the performance of Periodic Test OPT-09.7, High Pressure Coolant Injection (HPCI) System Valve Operability Test performed on Unit 2. 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 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.

To assess the licensee’s ability to identify and correct adverse conditions, the inspectors reviewed the licensee’s actions in response to AR 157145 which documented the Unit 2 drywell floor drain flow monitoring failing to meet surveillance requirement acceptance criteria.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed Plant Operating Manual 0PLP-22, Temporary Changes, to assess implementation of the below listed temporary modifications. The inspectors reviewed these temporary modifications to verify that the modifications were properly installed and whether they had any effect on system operability. The inspectors also assessed drawings and procedures for appropriate updating and post-modification testing.

- EC 61111, Bypass Motor-Driven Fire Pump Transfer Switch
- EC 60740, Reroute Condenser Vacuum Instrumentation Piping

b. Findings

No findings of significance were identified.

Enclosure

1EP6 Drill Evaluation

a. Inspection Scope

The inspectors observed one site emergency preparedness training drill/evolution conducted on June 2, 2005. The inspectors reviewed the drill scenario narrative to identify the timing and location of classification, notification, and protective action recommendation (PAR) 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 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 associated Emergency Response Organization Team Training Drill Critique Report.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution

a. Inspection Scope

.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 CAP. The review was accomplished by reviewing daily AR reports.

.2 Annual Sample Review

a. Inspection Scope

The inspectors performed an in-depth annual sample review of selected ARs 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 associated corrective actions for AR 156020, Unit 2 Reactor Scram, that occurred on April 9, 2005.

b. Findings

Introduction:

A Green self-revealing NCV of TS 5.4.1.a. Procedures, was identified for failure to provide adequate condensate system procedural guidance to preclude the reactor feed pumps (RFPs) from tripping on low suction pressure during plant operations.

Description:

On April 9, 2005 (following RFO B217R1), with Unit 2 operating at 65 percent power, the 2A RFP was idled to support power ascension testing. Immediately after the 2A RFP flow was terminated, an unanticipated low pressure condition occurred in the condensate system which tripped the condensate booster pumps (CBP) and the 2B RFP. After the 2B RFP tripped, operators attempted to return the 2A RFP to service, but the feed pump suction pressure had not completely recovered when reactor vessel water level lowered to the reactor protection system setpoint, resulting in an automatic reactor trip.

The cause of the loss of condensate booster pump and reactor feed pump suction pressures was attributed to the failure to establish condensate system flow procedural limitations. The condensate and feedwater systems had been modified (during the RFO) to support extended power uprate. These modifications resulted in changes to the flow and pressure operating characteristics. At the time of the reactor scram, the condensate system was being controlled within a pressure band of 150-190 psig per Operating Procedure 2OP-32 (Rev.130), Condensate and Feedwater Operating Procedure. This pressure band was desired in order to reduce adverse system interactions (e.g., CBP minimum flow line vibrations) based operating experience from Unit 1 post extended power uprate testing. The pressure was being controlled by diverting condensate flow to the condenser via operation of condensate return valve CO-FV-V49. During the idling of the 2A RFP, the pump's minimum flow valve opened which diverted additional flow to the condenser. The combined flow demand on the condensate system (i.e., flow through valve CO-FV-V49, 2A RFP minimum flow, and flow to 2B RFP) caused the operating condensate pumps to cavitate. As a result, condensate system pressure was reduced which initiated the transient and caused the reactor scram. The licensee reported the issue in LER 05000324/2005002, dated June 8, 2005 (see Section 4OA3.6). Further review determined that Unit 1 Procedures 1OP-32, Condensate and Feedwater System Operating Procedure (Rev 98) also did not contain appropriate condensate system flow procedural limitations, based on the modifications of the condensate and feedwater systems.

Enclosure

Analysis:

The failure to establish adequate procedure controls to assure sufficient condensate system pressure was available to preclude a loss of feedwater is greater than minor because the finding is associated with the procedure quality attribute of the Initiating Events Cornerstone and affects the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during power operations. This finding is of very low safety significance because, although it contributes to the likelihood of a reactor trip, it does not contribute to the likelihood that mitigation equipment or functions would be unavailable.

Enforcement:

TS 5.4.1.a requires written procedures be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Appendix A, November 1972. Regulatory Guide 1.33 requires written procedures for condensate and feedwater system operation. Contrary to TS 5.4.1.a, Procedures 1OP-32 and 2OP-32, were inadequate in that the procedures failed to establish guidance and controls to ensure adequate pressure to the suction of the RFPs. The procedural inadequacy of Procedure 1OP-32 contributed to a Unit 2 automatic reactor scram on April 9, 2005, due to low reactor vessel level. Because the issue is of very low safety significance and has been entered into the CAP as AR 156020, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000325,324/2005003-03, Inadequate Condensate System Operating Procedure.

.3 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a review of the 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 (discussed above), licensee trending efforts, and licensee human performance results. The review considered the period of January through June 2005, although some examples expanded beyond these dates as warranted by the scope of the trend. The review further included issues documented outside the normal CAP in major equipment lists, repetitive and/or rework maintenance lists, equipment and "hit" lists, quality assurance audit/surveillance reports, key performance indicators, self-assessment reports, and Maintenance Rule assessments. The inspectors compared and contrasted their results with the results contained in the latest quarterly trend reports. Corrective actions associated with a sample of the issues identified in the trend reports were reviewed for adequacy. The inspectors also evaluated the reports against the requirements of the CAP as specified in Nuclear Generation Group Standard Procedure CAP-NGGC-0200, Corrective Action Program, and 10 CFR 50, Appendix B. The inspectors performed a review of licensee actions to address site human performance trends which were previously discussed in NRC Inspection Report 05000325, 324/2004005, dated January 28, 2005.

b. Findings and Observations

No findings were identified. During the semi-annual trend review period, the previously identified adverse trend in site human performance was observed to generally improve. However, during the inspection period, three issues (two self-revealing and one licensee-identified) were identified. Two of the issues documented in this inspection report, as well as other lower level issues, were attributed to the operations area. As a result, the Operations Department held a stand-down which aimed at focusing attention on human performance during the inspection period. Following this stand-down, the inspectors noted a significant reduction in site human performance issues. .

4OA3 Event Follow-up

.1 Plant Events

a. Inspection Scope

The inspectors observed and/or reviewed plant parameters, equipment performance, and operator actions as a result of the below listed plant events to assure proper equipment operation and appropriate operator response. The inspectors verified that the licensee made timely notifications as required by 10CFR50.72. Documents reviewed are listed in the Attachment.

- AR 156020, Unit 2 reactor scram due to low reactor vessel water level on April 9, 2005
- AR 158668, Loss of emergency bus E-1 on May 12, 2005

b. Findings

No findings of significance were identified.

.2 Notice of Enforcement Discretion (NOED) 05-2-001: On May 12, 2005, the NRC granted a Unit 1 NOED in accordance with IMC 9900, Technical Guidance, Operations-Notices of Enforcement Discretion, related to enforcing compliance with the requirements of Technical Specification (TS) 3.4.5, RCS Leakage Detection Instrumentation. The details of the failure and the request is documented in a letter dated May 13, 2005, from the licensee to the NRC. Without power available to bus E-1, the drywell floor drain sump flow monitoring system, and the primary containment atmosphere and primary containment atmospheric gaseous RCS leakage monitoring systems were rendered inoperable. This was primarily the result of closure of the containment isolation valves associated with the systems and the inability to reopen the valves due to the inoperable bus E-1. The inspectors reviewed the applicable TS requirements, assessed the impact of the inoperable instrumentation, and monitored for compliance with the compensatory measures established as conditions for granting of the NOED. IMC 9900 requires that an unresolved item (URI) be opened to review the causes that may have led to the need for the NOED to determine whether any Enforcement actions are warranted.

Pending review of the causes that may have led to the need for the NOED, this item is identified as URI 0050325/2005003-01: NOED for Reactor Coolant System Leakage Detection Instrumentation.

.3 (Closed) LER 05000324/2005001: Compliance with Single Control Rod Withdrawal-Cold Shutdown Technical Specification. This event is discussed in Section 1R14.2 because operator human performance was a causal factor. The issue resulted in a finding of very low safety significance and an NCV. This LER is closed.

.4 (Closed) LER 05000325/2005003: Inappropriate Use of Technical Specification 3.0.5 During Control Rod Manipulations. This event is discussed in Section 1R20 because the issue was inspected during the Unit 1 maintenance/refueling outage B115M1. The issue resulted in a finding of very low safety significance (Green) and an NCV. This LER is closed.

.5 (Closed) LER 05000325,324/2005001: Operation Prohibited by Technical Specification-Inoperable Feedwater and Main Turbine High Water Level Trip. This event is discussed in Section 1R17 because the issue was inspected while reviewing permanent plant modifications. The issue resulted in a finding of very low safety significance (Green) and an NCV. This LER is closed.

.6 (Closed) LER 05000324/2005002: Automatic Shutdown Due to Condensate System Transient. The automatic shutdown portion of the LER is discussed in Section 4OA2.2 because the inspectors performed an in-depth review of the licensee's root cause analysis and corrective actions from this event as described in AR 156020. The issue resulted in a finding of very low safety significance (Green) and an NCV.

The LER also documented another reportable condition. On April 14, 2005, the licensee identified that the required evaluations within the Action time required by TS 3.4.9, Action a.2 had not been performed. Specifically, the licensee failed to perform the required evaluations within the action times required by the TS when the reactor cooling system heat up rate was exceeded on two occasions on April 9, 2005. The first occasion occurred while operations was warming up residual heat removal piping for shutdown cooling (SDC). The second occasion occurred while adjusting flow to establish SDC. The licensee entered the issue into the CAP as AR 157282 The failure to perform the evaluations within the Action time required by TS 3.4.9 is greater than minor because it is associated with equipment performance and affected the Initiating Events Cornerstone 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 finding was considered to have very low safety significance (Green) using Appendix G of the SDP because it did not constitute a finding that required quantitative assessment. The enforcement aspects of the finding are discussed in 4OA7. This LER is closed.

.7 (Closed) LER 05000325/2005002 : Reactor Core Isolation Cooling (RCIC) System Manual Actuation During Plant Shutdown. During the Unit 1 plant shutdown on April 15, 2005, the manual use of RCIC was required to control reactor vessel level. The required use of RCIC was do to unrelated issues associated with the loss of both reactor feed pumps. No findings of significance were identified. This LER is closed.

4OA5 Other Activities

Temporary Instruction (TI) 2515/163, Operational Readiness of Offsite Power

The inspectors collected data from licensee maintenance records, corrective action documents and procedures, and through interviews of station engineering, maintenance, and operations staff, as required by TI 2515/163. Appropriate documentation of the inspection results was provided to headquarters staff for further analysis, as required by the TI. This completes the Region II inspection requirements for this TI for the Brunswick site.

4OA6 Meetings, Including Exit

.1 Quarterly Integrated Inspection Report Exit

On July 8, 2005, the resident inspectors presented the inspection results to Mr. T. P. Cleary and other members of his staff, who acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

.2 Annual Assessment Meeting Summary

On April 12, 2005, the NRC's Chief of Reactor Projects Branch 4, and Resident staff assigned to the Brunswick Steam Electric Plant conducted a public meeting with Progress Energy - Carolina Power & Light (CP&L) to discuss the NRC's Reactor Oversight Process (ROP) and the Brunswick annual assessment of safety performance for the period of January 1, 2004 - December 31, 2004. Attendees included Brunswick management, site staff and local media.

This meeting was open to the public. The NRC's presentation material used during the meeting is available from the NRC's document system (ADAMS) as accession number ML051100457. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

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 meets the criteria of Section VI.A.1 of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV.

- TS 5.4.1.a requires that written procedures shall be implemented covering applicable procedures recommended in Regulatory Guide 1.33, Appendix A, November 1972. Regulatory Guide 1.33 requires written procedure for equipment control. Equipment Control Form 2-EC-05-115 requires the amphenols for the directional control valves on the hydraulic control unit associated with control rod 38-47 be removed. Contrary to 2-EC-05-115, on March 31, 2005, the control valve amphenols associated with control rod 38-47 were not removed. This resulted in the requirements of TS 3.10.4, Single Control Rod Withdrawal-Cold Shutdown, not being met while control rod 46-43 was withdrawn for maintenance. This was identified in the CAP as AR 155262. This finding is of very low safety significance because it did not constitute a finding that required quantitative assessment.

- TS 3.4.9 action a.2 requires an engineering evaluation be performed within 72 hours of exceeding a heatup rate to verify the reactor coolant system is acceptable for operation. Contrary to TS 3.4.9, two instances occurred on April 9, 2005 in which the heatup rate was exceeded and the engineering evaluation was not performed until April 14, 2005. The first instance occurred on April 9 at 0520 while operations was warming up the residual heat removal piping in preparation for placing shutdown cooling in service. The second instance occurred at 1020 while changing shutdown cooling system flow. This was identified in the CAP as AR 156020. This finding is of very low safety significance because it did not constitute a finding that required quantitative assessment.

- 10 CFR 50, Appendix B, Criteria XVI, requires in part that measures shall be established to assure that conditions adverse to quality, such as failures and malfunctions, are promptly identified and corrected. Information Notice 98-03 was issued January 1998 and described a 10 CFR 21 issue involving a facility that experienced a false breaker trip (approximately 50% loading) because a breaker current transformer was wired incorrectly. As a result of the information notice, the licensee revised OPM-BKR002A (Preventive Maintenance for K-Line Circuit Breakers) to incorporate current transformer polarity checks and performed the check on all safety related breakers by 1999. Contrary to Criteria XVI, on April 2, 2005, while again performing OPM-BKR002A on the E-7 main feeder breaker, it was noted that the current transformer was installed incorrectly and had not been previously corrected which would have caused the breaker to trip at a reduced load. This was identified in the CAP as AR 155387 and 155393. This finding is of very low safety significance because it did not constitute a finding that required quantitative assessment.

ATTACHMENT: SUPPLEMENTAL INFORMATION

Enclosure

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

G. Atkinson, Supervisor - Emergency Preparedness
L. Beller, Supervisor - Licensing/Regulatory Programs
A. Brittain, Manager - Security
T. Cleary, Director - Site Operations
D. DiCello, Manager - Nuclear Assessment
C. Elberfeld, Lead Engineer - Technical Support
C. Gannon, Site Vice President
J. Gawron, Training Manager
R. Kitchen, Manger - Engineering
D. Hinds, Plant General Manager
E. O'Neil, Manager - Site Support Services
A. Pope, Manager - Maintenance
E. Quidley, Manager - Outage and Scheduling
S. Tabor, Lead Engineer - Technical Support
M. Williams, Manager - Operations

NRC Personnel

P. Fredrickson, Chief, Reactor Projects Branch 4, Division of Reactor Projects Region II

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

050325/2005003-01	URI	NOED for Reactor Coolant System Leakage Detection Instrumentation (Section 4OA3.2)

Opened and Closed

05000325,324/2005003-01	NCV	Inadequate Design Control for Digital Feedwater Control System Modification (Section 1R17)
05000325/2005003-02	NCV	Inappropriate Use of Technical Specification 3.0.5 in Mode 5 Operations (Section 1R20)
05000325,324/2005003-03	NCV	Inadequate Condensate System Operating Procedure (Section 4OA2.2)

Closed

05000324/2005001	LER	Compliance with Single Control Rod Withdrawal-Cold Shutdown Technical Specification (Section 4OA3.3)
05000325/2005003	LER	Inappropriate Use of Technical Specification 3.0.5 During Control Rod Manipulations (Section 4OA3.4)
05000325,324/2005001	LER	Operation Prohibited by Technical Specification-Inoperable Feedwater and Main Turbine High Water Level Trip (Section 4OA3.5)
05000324/2005002	LER	Automatic Shutdown Due to Condensate System Transient (Section 4OA3.6)
05000325/2005002	LER	Reactor Core Isolation Cooling (RCIC) System Manual Actuation During Plant Shutdown (Section 4OA3.7)

Discussed

NONE

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Plant Operating Manual (POM) Volume I, Book 2, Administrative Instruction (AI) 0AI-68, Brunswick Nuclear Plant Response to Severe Weather Warnings, Rev. 24
POM Volume III, Plant Emergency Procedure 0PEP-02.6, Severe Weather, Rev. 9

Section 1R04: Equipment Alignment

POM Volume III, Operating Procedure 0OP-39, Diesel Generator Operating Procedure, Rev. 114
System Description SD-04, Primary Containment, Rev. 4
POM, Volume III, 1/2OP-43, Service Water System Operating Procedure, Rev. 76/115

Section 1R05: Fire Protection

POM, Volume XIX, 0PFP-PBAA, Power Block Auxiliary Areas Prefire Plans (SW, RW, AOG, TY, EY), Rev. 9
POM, Volume XIX, 0PFP-CB, Control Building Prefire Plans, Rev. 4

Section 1R11: Licensed Operator Requalification

Abnormal Operating Procedure 0AOP-16.0, RBCCW System Failure
Abnormal Operating Procedure 0AOP-5.0, Radioactive Spills, High Radiation, and Airborne Activity
Emergency Operating Procedure 0EOP-03, Secondary Containment Control Procedure
Emergency Operating Flow Charts, EOP-01-UG, User's Guide

Section 1R13: Maintenance Risk Assessments and Emergent Work Evaluation

AP-22, BNP Outage Risk Management
Abnormal Operating Procedure 0AOP37.1, Intake Structure Blockage, Rev. 7

Section 1R14: Operator Performance During Non-Routine Evolutions and Events

POM, Volume XXI, Abnormal Operating Procedure OAOP-23.0, Condensate/Feedwater System Failure, Rev. 22

System Description 19, High Pressure Coolant Injection System, Rev. 9

Section 1R17: Permanent Plant Modifications

EC 50091R1, Reactor Feed Pump Hydraulic Package Upgrade

Section 4OA3: Event Followup

POM, Volume VII, Operating Instruction OOI-50.1, 4160V Emergency Bus E-1 Electrical Load List, Rev. 35

POM, Volume XIII, Plant Emergency Procedure OPEP-02.1, Initial Emergency Actions, Rev. 48