UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
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July 19, 2004

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

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT - NRC INTEGRATED INSPECTION REPORT 05000325/2004003 AND 05000324/2004003

Dear Mr. Gannon:

On June 19, 2004, the 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 June 21, 2004, with you and other members of your staff.

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

The report documents two NRC-identified findings and one self-revealing finding, all of very low safety significance (Green). These findings were determined to involve violations of NRC requirements. 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) consistent with Section VI.A of the NRC Enforcement Policy. If you contest any non-cited violation 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, 324/2004003
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:	05000325/2004003 and 05000324/2004003
Licensee:	Carolina Power and Light (CP&L)
Facility:	Brunswick Steam Electric Plant, Units 1 & 2
Location:	8470 River Road SE Southport, NC 28461
Dates:	March 21, 2004 - June 19, 2004
Inspectors:	E. DiPaolo, Senior Resident Inspector J. Austin, Resident Inspector J. Moorman, Senior Reactor Inspector (Section 4OA5.2) M. Scott, Senior Reactor Inspector (Sections 1R02 and 1R17) P. VanDoorn, Senior Reactor Inspector (Sections 1R02 and 1R17) R. Winter, Senior Reactor Inspector (Sections 1R02 and 1R17) J. Fuller, Reactor Inspector (Sections 1R02 and 1R17)
Approved by:	Paul Fredrickson, Chief Reactor Projects Branch 4 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000325/2004003, 05000324/2004003; 03/21/2004 - 06/19/2004; Brunswick Steam Electric Plant, Units 1 and 2; Maintenance Effectiveness, Permanent Plant Modifications, and Other Activities.

The report covered a three-month period of inspection by resident inspectors, several region-based reactor inspectors, and a follow-up inspection performed by a regional senior reactor inspector. Three Green non-cited violations (NCVs) 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: Mitigating Systems

- Green. A self-revealing Green non-cited violation of Technical Specifications (TS) 5.4.1 was identified for failure to implement a maintenance procedure. Maintenance personnel failed to follow the emergency diesel generator (EDG) barring procedure (predictive maintenance which slowly cranks the engine) by not closing the right bank engine cylinder petcocks while performing the evolution on EDG 1 on June 6, 2004. This resulted in the EDG being inoperable until the condition was discovered when the EDG was started later that day.

  This finding is greater than minor because it affected the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to an event. The finding is of very low safety significance because the EDG was restored to an operable status within the TS limiting condition for operation allowed outage time. The finding was related to the cross-cutting area of human performance because the cause was due to maintenance workers failing to properly follow procedural requirements (Section 1R12).

- Green. The inspectors identified a non-cited violation of 10CFR50, Appendix B, Criterion V, for failure to install dielectric insulators on a service water isolation valve, required by a modification package. This resulted in a galvanic coupling between the carbon steel piping and the stainless steel valve, which could result in corrosion of the pipe flange at the bolt holes, accelerating corrosion of the interior of the pipe in areas where the cement lining had failed.

  This finding is greater than minor because it affected the equipment performance attribute of the mitigating systems cornerstone objective to ensure reliability of systems required to respond to initiating events. The finding is of very low safety significance because there was no actual loss of function, and a redundant valve was available for the isolation function (Section 1R17).

- Green. The inspectors identified a non-cited violation of 10 CFR 50, Appendix B, Criterion III, Design Control, for failure of design calculations to adequately address the potential for air entrainment in the high pressure coolant injection (HPCI) process flow due to vortexing. The Technical Specifications allowable value for the condensate storage tank (CST) level - low function, for automatic HPCI pump suction transfer to the suppression pool, was not adequately supported by these design calculations.

  The finding is greater than minor because it affects the design control attribute of the mitigating systems cornerstone objective. It is of very low safety significance because the finding is a design deficiency that would not result in loss of the HPCI function, and because the likelihood of having a low level in the CST that would challenge the CST level - low automatic HPCI suction transfer function is very low. In addition, alternate core cooling methods would normally be available, including reactor core isolation cooling as well as automatic depressurization system and low pressure coolant injection (Section 4OA5).

B. Licensee Identified Violations

None

REPORT DETAILS

Summary of Plant Status

Unit 1 began the report period in Mode 4 (Cold Shutdown) following the completion of refueling activities for the scheduled refueling outage B115R1. Mode 2 (Startup) and criticality were achieved on March 31. Unit 1 entered Mode 1 (Power Operation) on April 2 and commenced a power uprate (120 percent of original design) test program. Following the completion of the test program, Unit 1 achieved 100 percent power on April 18. On May 28, the unit reduced power to approximately 68 percent, at the load dispatcher's request, in response to an electrical grid condition. Power was returned to 100 percent later that day. On June 11, Unit 1 performed a planned downpower to approximately 60 percent to facilitate valve testing and secondary plant maintenance. The unit returned to approximately 100 percent power on June 13, where it remained for the duration of the inspection period.

Unit 2 began the report period operating at maximum power. The unit performed a planned downpower to approximately 50 percent on April 17 to facilitate valve testing and secondary plant maintenance. The unit returned to maximum power on April 19. On May 8, operators reduced power to approximately 50 percent in response to the inadvertent actuation of a reactor feed pump room deluge (fire protection) system. Maximum power was restored on May 9. On May 29, a planned shutdown to Mode 4 was performed to address elevated (1.1 gallon per minute), and upward trending, drywell unidentified leakage. Following temporary repairs to body-to-bonnet leakage on a residual heat removal/shutdown cooling injection supply check valve, the unit performed a startup and reached full power on June 3. A planned downpower to approximately 60 percent was initiated and completed on June 19 to facilitate a control rod sequence exchange and secondary plant maintenance. The unit remained at full power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R02 Evaluations of Changes, Tests or Experiments

a. Inspection Scope

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

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

The inspectors also reviewed a recent audit of the 10CFR50.59 process and selected several action requests (ARs) and work orders (WOs) to confirm that problems were identified at an appropriate threshold, were entered into the corrective action process, and appropriate corrective actions had been initiated. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

Partial System Walkdowns

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 had 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 licensee's configuration control program. Documents reviewed are listed in the Attachment.

- Emergency Diesel Generator (EDG) #2 while EDG #1 was OOS on May 11, 2004
- Unit 2 A loop of residual heat removal (shutdown cooling mode) while B loop was OOS on May 30, 2003
- Unit 1 A nuclear service water pump and A, B, and C residual heat removal pumps (redundant required features) while EDG #2 was OOS on June 14-18, 2004

Complete System Walkdown

The inspectors conducted a detailed review of the alignment and condition of the Units 1 and 2 control building ventilation (CREV) system (system number 8220). The inspectors reviewed the UFSAR, associated attachments of Operating Procedure 0PT 46.5, Control Room Air Conditioning Performance Test, and the system flow diagram (drawing numbers 4080 through 4080 SH1). In determining correct system lineup, the inspectors reviewed the documents listed in the Attachment.

To assess the licensee's identification and resolution of problems associated with the system, the inspectors reviewed the below listed ARs:

- AR 127550, Galvanic corrosion between CREV supply fan duct work and supports
- AR 128033, Validate proper operation of CREVs during a chlorine leak

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors reviewed current ARs and 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 for any existing conditions or deficiencies which would impair the operability of that equipment. Documents reviewed are listed in the Attachment. 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:

- Unit 1 North and South Core Spray Rooms, -17' elevation (2 areas)
- Unit 1 Drywell and Air Lock (2 areas)
- Service Water Building, 4' and 20' elevations (2 areas)
- Unit 1 Reactor Building - North and South, 20' elevation (2 areas)

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

On June 16, 2004, the inspectors observed licensed operator performance and reviewed the associated training documents during simulator training sessions for training cycle 2003-04. 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 training scenario tested the operators' ability to respond to a recirculation pump seal failure and utilize reactor pressure/level control emergency operating procedures.

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 Technical Specification (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 Maintenance Effectiveness

a. Inspection Scope

For the two equipment issues described in work documents 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 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. Documents reviewed are listed in the Attachment.

- AR 123488, Unit 1 main steam line pipe whip restraint condition monitoring
- Special Process Procedure 0SPP-ENG507, Diesel Generator Barring Procedure, Rev. 1, performed on EDG 1 on June 6, 2004

b. Findings

Introduction. A self-revealing Green non-cited violation (NCV) was identified for failure to close the EDG 1 engine right bank cylinder petcocks in accordance with procedural requirements, following predictive maintenance.

Description. On June 6, 2004, mechanical maintenance workers performed scheduled barring of EDG 1 (predictive maintenance which slowly cranks the engine). The evolution involves opening all 16 engine cylinder head petcocks (8 on each of the right and left bank of cylinders) and slowly cranking the engine in order to detect any fluids which may have leaked into the individual cylinders. During the evolution, the EDG is inoperable and unavailable. Following the barring evolution, Special Process Procedure 0SPP-ENG507, Diesel Generator Barring Procedure, required the engine cylinder head petcocks to be closed prior to declaring the EDG operable. The procedure required that two maintenance workers concurrently (simultaneously) verify and close the petcocks.

Following the engine barring evolution on June 6, 2004, the two maintenance workers each independently attempted to close the right and left bank cylinder petcocks. Subsequently, the workers independently verified the petcocks closed and operators declared the EDG operable. On startup of EDG 1 later that day, operations personnel discovered that the eight right bank cylinder petcocks were open. The cylinder petcocks were immediately closed. No damage to the EDG resulted from the condition. The inspectors questioned the affect that the open cylinder petcocks would have on engine performance. The licensee concluded that the open cylinder petcocks would result in engine cylinder loading imbalances which would have a detrimental affect on engine performance, reliability and availability.

The licensee determined that the cause of the failure to close the right bank cylinder petcocks following the barring evolution was due to human performance errors. Human performance barrier tools for self checking, peer checking, questioning attitude, and effective communication were identified as failed defenses. The licensee reviewed expectations in this area with mechanical maintenance workers. The licensee planned to review this event, as well as other events, to determine additional corrective actions for an adverse trend in the area of maintenance worker human performance.

Analysis. The failure to close EDG 1's right bank cylinder petcocks, while barring the EDG is greater than minor because it is associated with system configuration control and effected the mitigation availability of EDG 1. This finding was determined to be of very low safety significance (Green) because the EDG was returned to an operable status within the TS allowed outage time. The finding was related to the cross-cutting area of human performance because the cause was due to maintenance workers failing to properly follow procedural requirements.

Enforcement. Technical Specification 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 procedures for the performance of maintenance. Special Process Procedure 0SPP-ENG507, Diesel Generator Barring Procedure, Revision 1, step 7.1.11, requires that EDG 1 engine cylinder head petcocks be closed following engine maintenance barring. Contrary to Procedure 0SPP-ENG507, the right bank cylinder petcocks were left in the open position following the completion of the engine maintenance barring evolution on June 6, 2004. Because this issue is of very low safety significance and has been entered into the licensee's corrective action program (ARs 128848 and 129173), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000324,325/2004003-01, Failure to Follow EDG Barring Procedure.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

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 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 maintenance activities on the following five OOS equipment and conditions:

- WO 428918, Unit 2 D service air compressor OOS during planned generator exciter collection ring stoning (planned)
- AR 123551, Unit 1 HPCI system declared inoperable due to failure to meet ASME test requirements (emergent)
- AR 128178, Unit 1B control rod drive pump OOS on May 20, 2004, due to failed cooling water supply valve (emergent)
- AR 123877, power reduction due to high reactor feed pump casing level (emergent)
- WO 501906, Unit 1B nuclear service water pump and EDG 2 OOS concurrently resulting in yellow risk profile (planned)

The inspectors reviewed the following ARs to assess the licensee's identification and resolution of emergent problems:

- AR 129965, Corroded bolts on EDG#2 jacket water cooler outlet flange studs
- AR 128230, Unit 1 power reduction on May 28, 2004, due to loss of Weatherspoon power line (dispatcher directed downpower)
- AR 121910, Unit 2 unidentified leakage increase

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed or observed the following two evolutions to assess operator performance during non-routine evolutions and events. Operator logs, plant computer data, and associated operator actions were reviewed as well as the procedures listed in the Attachment.

- Unit 1 heatup following refueling outage on March 31, 2004
- Unit 2 unplanned downpower to approximately 50 percent power on May 8, 2004, due to inadvertent actuation of the A reactor feed pump room fire protection deluge system (AR 126451)

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the operability evaluations associated with the five following 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 (LCOs) 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 123551, Unit 1 HPCI pump failed during testing
- AR 123564, cracking observed adjacent to terminals on Unit 2 B-2 battery
- AR 123132, metal detached from EDG muffler
- WO 527097, Unit 1 drywell high range radiation monitor erratic indication
- WO 536513, unable to adjust EDG 2 frequency locally

To assess the licensee’s ability to identify and correct adverse conditions, the inspectors reviewed the licensee’s actions in response to the following ARs:

- AR 128208, Unit 2 turbine building main steam line tunnel elevated temperature
- AR 129803, Foreign material found in EDG 2 turbocharger intercooler
- AR 126630, Incorrect relay installed in main steam line isolation circuitry

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, in detail, an OWA associated with EDG 3 jacket water head tank drain valve leakage documented in AR 127060. The review evaluated the effect of the OWA on the operator's ability to implement abnormal or emergency operating procedures during 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.

Cumulative Effects Review

The inspectors reviewed the cumulative effects of all identified Units 1 and 2 OWAs to verify that they did not adversely impact the following: 1) the reliability, availability, and potential for misoperation of the effected systems; 2) the potential for increasing an initiating event frequency; and 3) impact on the ability of operators to respond in a correct and timely manner to a plant transient and accident. Aggregate impacts of the identified work-arounds on each individual operator watch station were also reviewed.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors evaluated engineering design change (EC) packages for eight modifications, in the barrier integrity, initiating events, and mitigating systems cornerstone areas, to evaluate the modifications for adverse affects on system availability, reliability, and functional capability. The modifications and the associated attributes reviewed are as follows:

EC 47524, Main Steam Isolation Valve Solenoid Replacement (Mitigating Systems, Barrier Integrity)

- Design basis equivalency
- Seismic & environmental qualification
- Installation records
- Functional testing adequacy and results
- Plant document updating

EC 50180, Freeze Seal Application on Lines 1-SW-490-6-046 and 2-SW-490-6-046 (Mitigating Systems)

- Evaluation of potential effects
- Pre and post nondestructive evaluations
- Failure mode analysis and associated compensatory actions
- Installation and restoration records

EC 50052, Iso-Phase Bus Duct Replacement (Mitigating Systems)

- Functional requirements in accordance with design bases
- Heat removal requirements met
- Documents updated
- Replacement components compatible with physical interface

EC 46730, Replace U2 Power Range Neutron Monitoring System (Initiating Events, Mitigating Systems)

- Conformance to design basis
- Post-Modification testing for operability and design basis
- Accident considerations

EC 46823, Replace Reactor Feedwater Pump Turbine Control Console with Digital Governor (Initiating Events, Mitigating Systems)

- Conformance to design basis
- Accident considerations
- Temporary modification for reliability (dryer and ventilation)

EC 46810, Unit 2 Standby Liquid Control (SLC) Concentration Change for Extended Power Uprate (EPU) (Mitigating Systems)

- Design basis and licensing basis equivalency
- Seismic & environmental qualification
- Installation records
- Functional testing adequacy and results
- Plant document updating
- Nondestructive examination requirements

EC 51048, Evaluate the Use of Two New Safety Relief Valve (SRV) Main Body Assemblies From Fermi Unit 2 For BNP (Mitigating Systems, Barrier Integrity)

- Design basis and licensing basis equivalency
- Seismic & environmental qualification
- Installation records
- Repair and Replacement records
- Welding requirements

- Material compatibility and design evaluation
- Functional testing adequacy and results
- Plant document updating
- Nondestructive examination requirements

EC 49001, Replacement For 2-SW-V3 (Mitigating Systems)

- Design basis equivalency
- Material compatibility and design evaluation
- Materials / Replacement Components: Material compatibility, Code Requirements, and Seismic Requirements
- Installation records
- Functional testing adequacy and results
- Plant document updating

For selected modification packages, the inspectors observed the as-built configuration. Documents reviewed included procedures, engineering calculations, modification design and implementation packages, WOs, site drawings, corrective action documents, applicable sections of the UFSAR, supporting analyses, the TS, and design basis information.

The inspectors also reviewed selected self-assessments and corrective action documents associated with modifications to confirm that problems were identified at an appropriate threshold, were entered into the corrective action process, and appropriate corrective actions had been initiated. Documents reviewed are listed in the Attachment.

b. Findings

Introduction

An inspector-identified Green NCV was identified for failure to correctly install service water (SW) outboard isolation valve 2-SW-V3 in accordance with the associated EC package.

Description

In April 2004, the inspectors identified that dielectric insulators were not installed on valve 2-SW-V3. This valve is the Unit 2 SW outboard isolation valve on the turbine building service water closed cooling water (TBCCW) supply header. This valve permits the conventional service water (CSW) header to be used to supply safety-related SW loads and provides isolation of TBCCW from SW, when required. This valve also prevents the CSW pumps from operating in a runout condition.

EC 49001 required a dielectric kit be used to electrically isolate the carbon steel piping and bolts from the stainless steel valve. Instructions for completing this modification were either inappropriate for the circumstances or valve installation was not accomplished in accordance with the instructions, in that, dielectric insulators were not installed on valve 2-SW-V3. This valve was installed in March 2003.

Failure to install the dielectric kit created a galvanic coupling between the carbon steel piping and the stainless steel valve. Because of the electrical potential difference between the two metals, the carbon steel piping and bolts acted as the anode and the SA351-CN3MN stainless steel valve body and disc acted as the cathode. This galvanic coupling, if left uninsulated, would lead to corrosion of the pipe flange at the bolt holes and could potentially accelerate corrosion of the interior of the pipe in areas where the cement lining had failed. This valve and associated piping is covered in insulation, which would prohibit identification of a corrosion degradation problem during a routine system walkdown.

The licensee issued AR 123991 for the deficiency identified by the inspectors. Although the valve was improperly installed, no visible corrosion was currently present at the flange bolts, and there was no immediate operability concern.

Analysis

This finding is greater than minor because it affected the equipment performance attribute of the mitigating systems cornerstone objective to ensure reliability of systems required to respond to initiating events. Because there was no actual loss of function, and a redundant valve was available for the isolation function, this finding is of very low safety significance (Green).

Enforcement

10CFR50, Appendix B, Criterion V, requires, in part, that activities affecting quality shall be prescribed by documented instructions and shall be accomplished in accordance with these instructions. Contrary to the above, instructions for completing modification EC 49001 were either inappropriate for the circumstances or valve installation was not accomplished in accordance with these instructions, in that, required dielectric insulators were not installed on valve 2-SW-V3, which resulted in an accelerated corrosive environment for the valve bolting and associated piping, beginning when the valve was installed in March 2003. Because the failure to install dielectric insulators on valve 2-SW-V3 is of very low risk significance and has been entered into the licensee's corrective action program (AR 00123991), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000324/2004003-02, Failure to Install Dielectric Insulating Kit Between Service Water Valve and Pipe Flange.

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 effected 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. Documents reviewed are listed in the Attachment.

- WO 170083, perform preventive maintenance on Unit 2 D residual heat removal system pump torus suction valve (2-E11-F004D) motor operator
- WO 536513, Unable to locally adjust EDG #2 frequency
- AR 123488, Unit 1 main steam line pipe whip restraint bolt tightening
- AR 121137, Unit 1 refuel outage supplemental spent fuel pool cooling tower fan
- AR 125558, Unit 2 HPCI system oscillations of speed, flow, and pressure

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities

a. Inspection Scope

.1 Unit 1 Refueling Outage

The inspectors evaluated Unit 1 RFO B115R1 activities which commenced on February 28, 2004. At the start of the inspection, fuel movement was complete and the unit was in Mode 4 (Cold Shutdown) and preparing for startup activities. Unit 1 entered Mode 1 (power operation) on April 2, to complete the outage. Documents reviewed are listed in the Attachment. The following specific areas were reviewed during the inspection period:

Licensee Control of Outage Activities. The inspectors observed and reviewed several specific activities, evolutions, and plant conditions to verify that the licensee maintained defense-in-depth commensurate with the outage risk control plan. 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. The inspectors reviewed operational logs to verify that procedure and TS requirements to monitor and record reactor coolant temperature were met.

- 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. The inspectors reviewed reactor physics testing results to verify that core operating limit parameters were consistent with the design.

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 reviewed the following issues identified during the outage to verify that the appropriate corrective actions were implemented:

- AR 123106, Unit 1 main steam line A whip restraint found with loose nuts (inspector-identified)
- AR 121925, Outage human performance error trend
- AR 121464, Refuel bridge drive shaft failure
- AR 122176, Reactor mode switch reactor protection system trip

.2 Unit 2 Maintenance Outage

The inspectors evaluated Unit 2 maintenance outage B216M1 activities which commenced on May 29, 2004. The planned outage was performed in order to address elevated drywell unidentified leakage (approximately 1.1 gpm). Documents reviewed are listed in the Attachment. The following specific areas were reviewed:

Outage Plan. The inspectors reviewed Brunswick Nuclear Plant Unit 2 Outage Risk Assessment for Maintenance Outage B216M1. 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.

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, as specified:

- 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.

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.

Identification and Resolution of Problems. The inspectors reviewed ARs to verify that the licensee was identifying problems related to outage activities at an appropriate threshold and entering them in the corrective action program. The inspectors reviewed the following issues identified during the outage to verify that the appropriate corrective actions were implemented:

- AR 128360, Nondestructive testing and tensile strength requirements not met for caps nuts installed for repair of 2-E11-F050B
- AR 128338, Operators attempted to start residual heat removal in shutdown cooling with suction valves closed (2-E11-F006A/6C)

b. Findings

No findings of significance were identified.

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, 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. The inspectors reviewed the documents listed in the Attachment.

- Periodic Test 0PT-12.2A, No. 1 Diesel Generator Monthly Load Test
- Periodic Test 0PT-12.2B, No. 2 Diesel Generator Monthly Load Test
- Periodic Test 0PT-46.5, Control Room Air Conditioning Performance Test

- Maintenance Surveillance Test 1MST-RPS34R, RPS Main Steam Line Isolation Valve Closure Circuit Response Time

To assess the licensee's identification and resolution of problems in this area, the inspectors reviewed the following ARs:

- AR 124575, HPCI system low level 2 initiation instrument failure
- AR 129450, Unit 1 main steam isolation valve dual indication

Inservice Surveillance Testing

The inspectors reviewed the performance of Periodic Test 0PT 9.2, HPCI System Operability Test, performed on Unit 1. 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 the two temporary modifications ECs identified below. The associated 10 CFR 50.59 screenings were reviewed against the system design bases documentation, including the UFSAR and TS, to verify that the modifications did not affect system operability. The inspectors reviewed modification documents to verify that configuration control was adequate. Post-installation testing requirements and test results (if applicable) were reviewed to confirm that the modification did not adversely impact interfacing systems. Licensee planned corrective actions necessary to remove the temporary modifications were also reviewed. Documents reviewed are listed in the Attachment.

- EC 58056, Temporary Body-to-Bonnet Leak Repair for 2-E11-F050B (residual heat removal B-loop injection check valve)
- EC 56053, Connect Ultrasonic Feedwater Flow Detectors for Trending/Comparison with Installed Instrumentation

b. Findings

No findings of significance were identified.

4OA1 Performance Indicator Verification

a. Inspection Scope

The inspectors sampled licensee submittals for the Units 1 and 2 performance indicators (PIs) listed below for the period April 2003 through March 2004. To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in Nuclear Energy Institute (NEI) 99-02, “Regulatory Assessment Performance Indicator Guideline”, Revision 2, were used to confirm the reporting basis for each data element.

Reactor Safety Cornerstone

- Safety System Unavailability, HPCI system
- Unplanned Power Changes per 7000 Critical Hours

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

a. Inspection Scope

Daily Reviews

To aid in the identification of repetitive equipment failures or specific human performance issues for followup, the inspectors performed frequent screenings of items entered into the licensee’s corrective action program (CAP). The review was accomplished by reviewing daily AR reports.

Annual Sample Review

The inspectors performed an in-depth annual sample review of selected ARs to determine whether 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 following issues and associated corrective actions were reviewed:

- AR 121086, Unit 1 drywell personnel lock penetration sleeve below minimum wall thickness
- AR 119700, During EDG #1 load testing, the Unit 1A core spray pump breaker failed to load shed resulting in the inability to tie EDG #1 to the emergency bus

Semi-Annual Trend Review

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 (discussed above), licensee trending efforts, and licensee human performance results. The review nominally considered the period of January through June, 2004, 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 specific items reviewed are listed in the Attachment. The inspectors compared and contrasted their results with the results contained in the licensee's latest quarterly trend reports. 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. Additional documents reviewed are listed in the Attachment.

b. Findings and Observations

No findings of significance were identified. The inspectors evaluated the licensee's trending methodology and observed that the licensee had performed a detailed review for trends. The licensee routinely reviewed cause codes, involved organizations, key words, and equipment reliability data to identify potential trends in its CAP data. The inspectors compared the licensee's process results with the results of the inspectors' CAP item screening and did not identify any discrepancies or potential trends that the licensee had failed to identify.

The inspectors noted that during the first six months of calendar year 2004, the licensee identified human performance trends in maintenance, outage work and general site performance. On June 6, 2004, the licensee discovered that EDG 1 was in an inoperable condition due to the engine right bank cylinder petcocks being left open following an engine barring evolution (see Section 1R13 for additional details). The licensee determined that the condition was caused by human performance errors performed by maintenance workers. Based on this event and the site's history of eight other human performance events in the previous nine months, the licensee concluded that an adverse human performance trend existed in the maintenance area. Several of the ARs associated with the previous events were categorized as Priority 1 which required a formal root cause and corrective actions to prevent recurrence. The licensee initiated AR 129173 to evaluate the previous events, determine any common cause elements between the events, and to determine if previous actions were adequately comprehensive.

During the Spring 2004 Unit 1 refueling outage, the inspector noted, and discussed with licensee management, several issues/events entered into the CAP which may have indicated a broader adverse trend in human performance. Subsequently, the licensee performed a collective review of observations and ARs generated during the outage. The licensee concluded that a significant adverse trend in outage human performance errors existed. The licensee initiated AR 121925 and performed an evaluation of the individual errors (seven) to determine if any common cause issues needed to be acted upon during the outage. As a result of the licensee's investigation, actions were taken to heighten site awareness to human performance issues during the outage and additional future enhancements were planned. Additionally, the licensee's first quarter 2004 CAP data evaluation report identified a potential adverse trend in general site human performance. The licensee initiated AR 127455 to address this issue.

4OA3 Event Followup

(Closed) Licensee Event Report (LER) 05000325,324/2003-004-00: Loss of Generator Excitation Results in Reactor Protection System and Other Specified System Actuations.

This LER documented a Unit 2 reactor trip on November 4, 2003, and other engineered-safety feature (ESF) actuations as a result of a loss of generator excitation. Prior to the scram, the licensee noted a degraded condition of the generator exciter collector ring brushes. An action plan was developed and was being implemented to address this condition when the failure occurred. The generator exciter inboard collector ring and brush holders failed, resulting in a loss of generator excitation. The resultant voltage decrease on plant buses caused several ESF actuations. All actuations operated successfully with the exception of the Unit 2 A Train of standby gas treatment failure to start. The voltage transient adversely affected fire protection relays, which required operators to reset the start circuit. The licensee implemented modifications to eliminate the vulnerability. The voltage transient also affected other equipment on both units. The reactor feed pump turbines unexpectedly received trips following the scram. This issue was later determined to be due to a design oversight of a modification to the reactor feed pump turbine speed control system. NRC follow-up of this issue resulted in a Green finding (see NRC Inspection Report 05000324/2004007, dated April 2, 2004). The response of the other equipment was reviewed and the licensee determined that the effects were to be expected based on the nature of voltage transient and load stripping of buses and equipment. The licensee has implemented modifications to the Units 1 and 2 exciters and made program changes to improve system performance. This LER is closed.

4OA4 Cross Cutting Aspects of Findings

Section 1R12 documents a self-revealing Green finding. The cause of EDG #1 right bank cylinder petcocks being inadvertently left open was due to maintenance workers failing to properly follow procedural requirements. This cause relates to the cross cutting aspect of human performance.

4OA5 Other Activities

.1 (Open) NRC Temporary Instruction 2515/156, Offsite Power System Operational Readiness

a. Inspection Scope

The inspectors collected data from licensee maintenance records, event reports, corrective action documents and procedures and through interviews of station engineering, maintenance, and operations staff, as required by Temporary Instruction (TI) 2515/156. The data was gathered to assess the operational readiness of the offsite power systems in accordance with NRC requirements such as Appendix A to 10 CFR Part 50, General Design Criterion (GDC) 17; Criterion XVI of Appendix B to 10 CFR Part 50, plant TS for offsite power systems; 10 CFR 50.63; 10 CFR 50.65 (a)(4), and licensee procedures. The specific documents reviewed by the inspectors are listed in the Attachment.

b. Findings

No findings of significance were identified. Based on the inspection, no immediate operability issues were identified. In accordance with TI 2515/156 reporting requirements, the inspectors provided the required data to the headquarters staff for further analysis. This TI will remain open pending completion of that analysis.

.2 (Closed) Unresolved Item (URI) 05000325,324/2003008-01: Failure to Adequately Consider Vortexing in the Calculation for CST Level for Automatic Transfer of the HPCI Pump Suction.

During the Safety System Design and Performance Capability (SSDPC) Inspection completed on August 29, 2003, the inspection team identified a violation of 10 CFR 50, Appendix B, Criterion III, Design Control requirements. The TS allowable value for the condensate storage tank (CST) level - low function, for automatic HPCI pump suction transfer to the suppression pool, was not adequately supported by design calculations. The calculations did not adequately address the potential for air entrainment in the HPCI process flow due to vortexing. The SSDPC Inspection had identified that the finding was greater than minor because it affected the design control attribute of the mitigating systems cornerstone objective. It was also of very low safety significance (Green) because the finding was a design deficiency that would not result in loss of the HPCI function, and because the likelihood of having a low level in the CST that would challenge the CST level - low automatic HPCI suction transfer function was very low. In addition, alternate core cooling methods would normally be available, including reactor core isolation cooling as well as automatic depressurization system and low pressure coolant injection

However, the regulatory disposition of this item as a cited or non-cited violation was unresolved pending further NRC review of the requirements for the CST level - low function and of the licensee's corrective actions related to restoration of compliance with Criterion III of 10 CFR 50, Appendix B. An NRC review of the TS requirements for the CST level - low function was documented in Task Interface Agreement (TIA 2003-05); "NRC Policy Questions on Technical Specification Adequacy And Related Technical Specification Operability at Brunswick Nuclear Plant." (ADAMS Accession No. ML040160210). This TIA confirmed that the licensee was required to have in place adequate design basis documentation, supporting calculations, and proceduralized operator actions in order for the CST level - low function to be considered operable. The licensee documented this issue and the corrective actions in AR 102456. The inspectors reviewed the licensee's proposed corrective actions with respect to the information provided in the TIA and determined that they were adequate to restore compliance with Criterion III of 10 CFR 50, Appendix B. As such, this finding meets the criteria for dispositioning as an NCV. Therefore, because the failure to address vortexing in the CST design calculations is of very low safety significance and has been entered into the licensee's corrective action program (AR 102456), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000325,324/2004003-03, Failure to Adequately Consider Vortexing in the Calculation for CST Level for Automatic Transfer of the HPCI Pump Suction.

4OA6 Meetings, Including Exit

On June 21, 2004, the resident inspectors presented the inspection results to Mr. C. J. Gannon and other members of his staff. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

G. Atkinson, Supervisor - Emergency Preparedness
L. Beller, Supervisor - Licensing/Regulatory Programs
A. Brittain, Manager - Security
E. Conway, Senior Nuclear Security Specialist
D. DiCello, Manager - Nuclear Assessment
C. Elberfeld, Lead Engineer - Technical Support
C. Gannon, Site Vice President
J. Gawron, Training Manager
D. Hinds, Plant General Manager
R. Kitchen, Lead Nuclear Security Specialist
W. Noll, Director - Site Operations
E. O'Neil, Manager - Site Support Services
E. Quidley, Manager - Outage and Scheduling
S. Tabor, Lead Engineer - Technical Support
H. Wall, Manager - Maintenance
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

NONE

Opened and Closed

05000325,324/2004003-01	NCV	Failure to Follow EDG Barring Procedure (1R12)
05000324/2004003-02	NCV	Failure to Install Dielectric Insulating Kit Between Service Water Valve and Pipe Flange (1R17)
05000325,324/2004003-03	NCV	Failure to Adequately Consider Vortexing in the Calculation for CST Level for Automatic Transfer of the HPCI Pump Suction (Section 4OA5.2)

Closed

05000325,324/2003008-01	URI	Failure to Adequately Consider Vortexing in the Calculation for CST Level for Automatic Transfer of the HPCI Pump Suction (Section 4OA5.2)
05000325,324/2003-004-00	LER	Loss of Generator Excitation Results in Reactor Protection System and Other Specified System Actuations (Section 4OA3)

Discussed

2515/156	TI	Offsite Power System Operational Readiness (Section 4OA5.1)

LIST OF DOCUMENTS REVIEWED

Section 1R02 and 1R17: Evaluations of Changes, Tests or Experiments and Permanent Plant Modifications

Evaluations

02-0371, EC 46861, Unit 1 Power Uprate
00-1820, ESR 00-00260, GE14 Fuel Evaluation
99-1472, EC 47524, Main Steam Isolation Valve Solenoid Valve Replacement
01-1477, ESR 00-00452, Replacement of Unit 1 Power Range Neutron Monitoring recorders
02-0693, TCF-02-013, Troubleshooting Reactor Building pressure control problem
02-1791, 0GP-03, Procedure changes for Turbine pressure regulator inoperable without backup
02-1146, EC 47898, Generator out of step Protection
02-1058, EC 46730, Replace U2 Power Range Neutron Monitoring
03-1206, POM 0OI-01.03 Attachment 12, Procedure Revision for Entry into MSIV Pit to Facilitate Inspection for Leaks

Screened Out Items

EC 46361, Temporary Modification - Installation, Operation, and Removal of Temporary Vibration Monitoring Equipment on the Main Steam and Feedwater Piping and Supports
EC 49058, Replacement of Pneumatic Controller in 4kV Emergency Switchgear Room
EC 50051, Bypass Switches for Turbine First Stage Pressure Permissive
EC 51180, Steam Dryer Repair and Modification
ONEP-54, Building Ventilation Pressure Control, Rev. 20
EC 50052, Iso-Phase Bus Duct Replacement
EC 45936, U1 AC System Service Load Assessment Testing Requirements
EC 50054, U1 Main Power Transformer Replacement
EC 46798, Replacements for Obsolete GE Emergency 4kV Switchgear Relays
EC 46810, Unit 2 SLC Concentration Change for EPU
EC 53116, Replacement Diesel Starting Air Pressure Reducing Valves
EC 47829, RHR Room Coolers Setpoints Changed
EC 51048, Evaluate the Use of Two New (SRV) Main Body Assemblies From Fermi U2 For BNP
EC 49001, Replacement for 2-SW-V3
EC 46823, Replace Reactor Feedwater Pump Turbine Control Console with Digital Governor

Self-Assessment Related Documents

Self Assessment Report AR 78513, Fixed Crane Evaluation
Self Assessment Report AR 78464, Technical Assessment of Calculations
Self Assessment Report AR 50953, Design Quality
Brunswick Engineering Assessment, B-ES-02-01 (BNAS 02-067)
OE 17801 - Water Entered HPCI Steam Line Following Scram - NA
Action Request (AR) 120550, Incorrectly listed elevation for new RPV level transmitter
AR 110798, Valve wiring discrepancies
AR 106672, EC Package No. 1-EC-03-124, degraded greater than 30 days

AR 00123991, Dielectric Insulating Washers Not Installed on 2-SW-V3
AR 00123972, Incomplete Documentation For A/R Corrective Action
AR 00058079, Temporary power without 10 CFR 50.59 evaluation
AR 00063726, Temporary power without 10 CFR 50.59 evaluation
AR 00060680, Inappropriate 10 CFR 50.59 exemption
AR 00092000, 10 CFR 50.59 screen versus evaluation
AR 00092093, Quality Control Non Destructive Examination Reports

Section 1R04: Equipment Alignment

POM, Vol. III, 0OP-39, Diesel Generator Operating Procedure, Rev. 101
POM, Vol. III, 2OP-17, Residual Heat Removal System Operating Procedure, Rev. 133
POM, Vol. III, 1OP-17, Residual Heat Removal System Operating Procedure, Rev. 81
POM, Vol. III, 1OP-43, Service Water System Operating Procedure, Rev. 74
POM, Vol. III, 0OP-37, Control Building Ventilation System Operating Procedure, Rev. 44
Units 1 and 2 Technical Specifications
Technical Requirements Manual

Section 1R05: Fire Protection

POM, Vol. XIX, Prefire Plan, 1PFP-RB, Reactor Building Prefire Plans, Rev. 5
POM, Vol. XIX, Prefire Plan, 0PFP-PBAA, Power Block Auxiliary Areas Prefire Plans, Rev. 8

Section 1R12: Maintenance Effectiveness

Nuclear Generation Group Standard Procedure, EGR-NGGC-0351, Condition Monitoring of Structures, Rev. 12
Nuclear Generation Group Standard Procedure OPS-NGGC-1303, Independent Verification, Revision 3

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

POM, Vol. XXI, Abnormal Operation Procedure 0AOP-23.0, Condensate/Feedwater System Failure, Rev.21
POM, Vol. XXI, Abnormal Operating Procedure 0AOP-34.0, Chlorine Emergencies, Rev. 23

Section 1R19: Post Maintenance Testing

POM, Vol. XII, Preventive Maintenance, 0PM-MO504, Mechanical Inspection and Lubrication of Limitorque Operators, Rev. 22

Section 1R20: Refueling and Other Outage Activities

POM, Vol. I, Administrative Procedure 0AP-022, BNP Outage Risk Management, Rev. 16
Procedure 0AP-038, Reactivity Management Program Manual
Procedure 0AP-022, BNP Outage Risk Management

Section 1R22: Surveillance Testing

POM, Vol. X, Periodic Test 0PT-12.2A, No. 1 Diesel Generator Monthly Load Test, Rev. 78

Section 1R23: Temporary Plant Modifications

AR 128356, 2-E11-F0508 body-to-bonnet flange leakage
Nuclear Generation Group Standard Procedure EGR-NGGC-0005, Engineering Change, Vol. 99, Book/Part 99, Rev. 21
Nuclear Generation Group Standard Procedure CHE-NGGC-0045, NGG Chemical Control Program, Vol. 99, Book/Part 99, Rev. 9
NRC Generic Letter 90-05, Guidance for Performing Temporary Non-Code Repair of ASME Class 1, 2, and 3 Piping
POM, Vol. XXII, Plant Programs, 0PLP-22, Temporary Changes, Rev. 17

Section 4OA1: Performance Indicator Verification

System engineer records for system unavailability
NRC website performance indicator information
Operator Logs (April 2003-March 2004)
Monthly Operating Reports (April 2003-March 2004)

Section 4OA2: Identification and Resolution of Problems

Nuclear Generation Group Standard Procedure, Volume 99, Book/Part 99, CAP-NGGC-0200, Corrective Action Program, Rev. 11
AR 1278455, Potential adverse trend in sitewide human performance
AR 129173, Maintenance adverse trend in human performance
AR 121925, Outage human performance error trend
Brunswick Plant CAP Data Evaluation Report (2004 First Quarter)