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

January, 16, 2004

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. 05000325/2003006 AND 05000324/2003006

Dear Mr. Gannon:

On December 20, 2003, 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 December 18, 2003, 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.

This report documents one finding concerning an inadequate design review of a Unit 2 reactor feed pump speed control system modification. This finding has potential safety significance greater than very low significance. This finding did present an immediate safety concern. However, compensatory measures are in place while long-term corrective measures are being implemented. In addition, the report documents one self-revealing finding of very low safety significance (Green). This finding was determined to involve a violation of NRC requirements. However, because of the very low safety significance and because it is entered into your corrective action program, the NRC is treating this finding as a non-cited violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy. Additionally, a licensee-identified violation which was determined to be of very low safety significance is listed in this report. 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 10CFR 2.790 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, 324/2003006
w/Attachment: Supplemental Information

cc w/encl: (See page 3)

cc w/encl:
W. C. Noll, Director
Site Operations
Brunswick Steam Electric Plant
Carolina Power & Light
Electronic Mail Distribution

David H. Hinds
Plant Manager
Brunswick Steam Electric Plant
Carolina Power & Light Company
Electronic Mail Distribution

James W. Holt, Manager
Performance Evaluation and
Regulatory Affairs PEB 7
Carolina Power & Light Company
Electronic Mail Distribution

Edward T. O'Neil, Manager
Support Services
Carolina Power & Light Company
Brunswick Steam Electric Plant
Electronic Mail Distribution

Lenny Beller, Supervisor
Licensing/Regulatory Programs
Carolina Power and Light Company
Electronic Mail Distribution

William D. Johnson
Vice President & Corporate Secretary
Carolina Power and Light Company
Electronic Mail Distribution

John H. O'Neill, Jr.
Shaw, Pittman, Potts & Trowbridge
2300 N. Street, NW
Washington, DC 20037-1128

Beverly Hall, Acting Director
Division of Radiation Protection
N. C. Department of Environment
and Natural Resources
Electronic Mail Distribution

Peggy Force
Assistant Attorney General
State of North Carolina
Electronic Mail Distribution

Chairman of the North Carolina
Utilities Commission
c/o Sam Watson, Staff Attorney
Electronic Mail Distribution

Robert P. Gruber
Executive Director
Public Staff NCUC
4326 Mail Service Center
Raleigh, NC 27699-4326

Public Service Commission
State of South Carolina
P. O. Box 11649
Columbia, SC 29211

Donald E. Warren
Brunswick County Board of
Commissioners
P. O. Box 249
Bolivia, NC 28422

Warren Lee
Emergency Management Director
New Hanover County Department of
Emergency Management
P. O. Box 1525
Wilmington, NC 28402-1525

Distribution w/encl: (See page 4)

Distribution w/encl:
B. Mozafari, NRR
L. Slack, RII EICS
RIDSRIDSNRRDIPMLIPB
PUBLIC

OFFICE	DRP/RII	DRP/RII	DRP/RII	DRS/RII	DRS/RII	DRS/RII	DRS/RII
SIGNATURE	GTM	EMD	JDA	MAS1	JLK1	KFO	EXL2
NAME	GMacdonald:as	EDiPaolo	JAustin	MScott	JKreh	KO'Donohue	ELea
DATE	01/15/04	01/16/04	01/16/04	01/15/04	01/15/04	01/15/04	01/15/04
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO
PUBLIC DOCUMENT	YES NO						

OFFICIAL RECORD COPY DOCUMENT NAME: C:\ORPCheckout\FileNET\ML040160461.wpd

U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos:	50-325, 50-324
License Nos:	DPR-71, DPR-62
Report Nos:	05000325/2003006 and 05000324/2003006
Licensee:	Carolina Power and Light (CP&L)
Facility:	Brunswick Steam Electric Plant, Units 1 & 2
Location:	8470 River Road SE Southport, NC 28461
Dates:	September 21, 2003 - December 20, 2003
Inspectors:	E. DiPaolo, Senior Resident Inspector J. Austin, Resident Inspector G. MacDonald, Senior Project Engineer (Section 1R06) M. Scott, Senior Reactor Inspector (Section 1R12) J. Kreh, Emergency Preparedness Inspector (Section 1EP 2-5) K. O'Donohue, Senior Reactor Inspector (Section 1R11) E. Lea, Senior Operations Engineer (Section 1R11)
Approved by:	Paul Fredrickson, Chief, Reactor Projects Branch 4 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000325/2003-006, 05000324/2003-006; 09/21/2003-12/20/2003; Brunswick Steam Electric Plant, Units 1 and 2; Maintenance Effectiveness, Permanent Plant Modifications.

The report covered a three-month period of inspection by resident inspectors, a senior project engineer, a senior operations engineer, senior reactor inspectors, and a regional emergency preparedness inspector. One Green non-cited violation (NCV) was identified. The significance of most findings is indicated by its 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 non-cited violation was identified for the licensee's failure to position the Unit 2 high pressure coolant injection (HPCI) system turbine exhaust stop check valve in the open position following system maintenance, in accordance with plant procedures. This resulted in failure of the exhaust line rupture discs during testing, a primary containment isolation of the system, and activation of the HPCI room fire protection system.

  This finding is greater than minor because it is associated with system configuration control and affected the mitigating availability of the HPCI system. This finding was determined to be of very low safety significance (Green) because the HPCI system was returned to an operable status within the Technical Specification allowed outage time. The finding was related to the cross-cutting aspect of Human Performance because the cause was determined to be due to plant operators using improper techniques in verifying the valve's position. Other contributing causes including operator knowledge deficiencies of valve operation, failure to perform an independent check of valve position, and the pre-job brief's limited scope were also related to Human Performance. (Section 1R12)

Cornerstone: Initiating Events and Mitigating Systems

- To Be Determined (TBD). A self-revealing finding was identified for an inadequate design review of a Unit 2 reactor feed pump speed control modification. The modification replaced the existing mechanical-hydraulic speed control system with a digital speed control system. The system is powered by internal power supplies that would fault, and thus cease to supply output power, with one cycle of sensed abnormal supply voltage. As a result, the reactor feed pumps would trip following a unit trip due to the supply voltage transient caused by the swap of in-house loads from the unit auxiliary transformer to the startup auxiliary transformer.

The finding is unresolved pending completion of a significance determination. This issue is greater than minor because, if left uncorrected, it would increase the likelihood of initiating events caused by a loss of reactor feed pumps following transients and affect the reliability and functional capability of the reactor feed pumps to mitigate events (unit trips). The finding was determined to have potential safety significance greater than very low because of the increased likelihood of initiating events, resultant reduced functional capability of the reactor feed pumps to mitigate events as a result, and the length of time the condition existed. (Section 1R17)

B. Licensee Identified Violations

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

REPORT DETAILS

Summary of Plant Status

Unit 1 began the report period operating at full power. On September 27, 2003, power was reduced to approximately 50 percent to perform planned maintenance on the reactor feed pumps and testing on the control rods and main steam isolation valves. The unit returned to maximum power on September 29. On November 14, power was reduced to approximately 50 percent for planned maintenance, control rod scram solenoid pilot valve maintenance, and surveillance testing. The unit returned to maximum power on November 16. Power was reduced to approximately 50 percent on November 21 to troubleshoot speed control problems on the A reactor feed pump. Full power was achieved on November 23 where it remained for the duration of the inspection period.

Unit 2 began the report period operating at full power. On September 23, 2003, power was reduced to approximately 50 percent to facilitate repairs to a main condenser tube leak. The unit returned to full power on September 25. On November 4, Unit 2 tripped due to a loss of main generator field. The loss of field was caused by the failure of the main generator alternator brush/collector ring (see Section 4OA3 for additional details). Following repairs to the main generator exciter, a unit startup was commenced on November 7, and maximum power was achieved on November 9. On December 5, the unit reduced power to approximately 50 percent to facilitate a modification to the reactor feed pumps to supply the governors with an uninterruptible power source (see Section 1R17 for details). The unit returned to maximum power on December 8 where it remained for the duration of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors assessed the effectiveness of the licensee’s cold weather protection program as it related to ensuring that the facility’s diesel-driven fire pump, emergency diesel generators, and condensate storage tank low level switches would remain functional and available in cold weather conditions. In addition to reviewing the licensee’s program-related documents and procedures, walkdowns were conducted of the freeze protection equipment (e.g., heat tracing, area space heaters, etc.) associated with the above systems/components. Licensee problem identification and resolution was also assessed. This included review of Action Request (AR) 110949 which documented that freeze protection preventive maintenance was not completed as scheduled for Unit 2. Documents reviewed during the course of this inspection 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 reviewed the resolution of licensee identified equipment alignment problems that could cause initiating events or impact mitigating system availability. The inspectors reviewed available structures, systems or components (SSCs) to verify that they met the requirements of the licensee's configuration control program. The inspectors reviewed documents listed in the Attachment.

- Unit 1 conventional and B train nuclear service water pumps when A train nuclear service water pump was OOS for planned maintenance on October 1, 2003.
- Unit 2 HPCI system when reactor core isolation cooling (RCIC) system was OOS for planned maintenance on October 23, 2003.
- Unit 2 RCIC system when HPCI system was inoperable due to ruptured exhaust diaphragms on November 13-14, 2003.

Complete System Walkdown

The inspectors conducted a detailed review of the alignment and condition of the emergency diesel generators (EDGs). The inspector reviewed the Updated Final Safety Analysis Report, associated attachments of Diesel Generator Operating Procedure 0OP-39, and the system flow diagram (drawing numbers D-02265 through D-02274). The inspectors reviewed pending design and equipment issues to verify that the identified deficiencies did not significantly impact the system's functions. Items included in this review were: 1) the operator workaround list; 2) the temporary modification list; 3) outstanding maintenance work requests/work orders (WOs); and 4) operator turnover sheets. The following related ARs were reviewed to assure that the licensee had properly characterized and prioritized equipment problems in the corrective action program:

- AR 49367 EDG 2 Inoperable due to high cylinder exhaust temperature
- AR 55517 EDG 4 light socket short
- AR 86529 Unexpected diesel generator start during SCRAM on January 1, 2003
- AR 102323 480V maintenance rule functional failure - feeder breaker to motor control center DGC (2-E7-AY8-52) failed to trip

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 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 areas important to reactor safety and reviewed documents listed in the Attachment to verify that the requirements for fire protection design features, fire area boundaries, and combustible loading were met:

- Units 1 and 2 north and south emergency core cooling pipe tunnels (4 areas)
- EDG fuel cells, -1 foot 6 inch elevation (1 area)
- EDG basement, 2 foot elevation (1 area)

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

Internal Flooding

The inspectors reviewed the licensee's internal flooding analysis as described in Updated Final Safety Analysis Report (UFSAR) Section 3.4.2, Protection From Internal Flooding. Due to the risk significance of equipment in the diesel generator building and the reactor buildings, the inspectors reviewed UFSAR Section 3.4.2 analysis of the effects of postulated piping failures for these two areas to determine if the analysis assumptions and conclusions were based on the current plant configuration. The internal flooding design features and equipment for coping with internal flooding was inspected. The walkdown included sources of flooding and drainage, sump pumps, level switches, watertight doors, curbs , pedestals and equipment mounting. The inspectors reviewed the testing of the level alarms and reviewed the procedures for coping with internal flooding. Documents reviewed are listed in the Attachment.

External Flooding

The inspectors reviewed the licensee's external flooding analysis as described in UFSAR Section 3.4.1, Protection from External Flooding, to determine the external flood control design features. Walkdowns were conducted to inspect the external flood protection barriers including watertight doors, curbs, sealing of external building penetrations below floodline, and the sump pumps and level alarm circuits. Procedures for coping with external flooding were reviewed and the inspectors walked down the portable flood protection equipment listed in Procedure 0AI-68, Brunswick Nuclear Plant Response to Severe Weather Warnings. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

a. Inspection Scope

The inspectors reviewed activities associated with the cleaning of the EDG 4 turbocharger intercooler heat exchange per WO 453062. The inspectors reviewed the results of the EDG 4 intercooler inspection conducted in accordance with preventive maintenance procedures. The inspection results were analyzed to determine if inspection frequencies were adequate to detect degradation prior to loss of heat removal capability below design-basis values. The inspectors reviewed the documents listed in the Attachment.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

Quarterly Review

On November 4, 2003, the inspectors observed licensed operator performance and reviewed the associated training documents during two simulator examinations. The simulator observations and reviews included evaluations of emergency operating procedure and abnormal operating procedure utilization. The inspectors reviewed LORX-001 and LORX-035 which documented the associated simulator examination scenarios. The simulator examination evaluated operator response to plant transients initiated by plant equipment problems, reactivity manipulations, a small break loss of coolant accident with failures of emergency core cooling systems, and an unisolable steam leak outside containment. The inspectors reviewed operator activities to verify consistent clarity and formality of communication, conservative decision-making by the crew, appropriate use of procedures, and proper alarm response. Group dynamics and supervisory oversight, including the ability to properly identify and implement appropriate 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. The inspectors reviewed documents listed in the Attachment.

Periodic Evaluation (Biennial)

The inspectors reviewed documentation, interviewed licensee personnel, and observed the administration of simulator operating tests associated with the licensee's operator requalification program. Job performance measures (JPMs) associated with the licensee's operator requalification program, which the licensee administered at the beginning of the year, were reviewed by the inspectors. Each of the activities performed by the inspectors was done to assess the effectiveness of the licensee in implementing requalification requirements identified in 10 CFR 55, "Operators' Licenses." Evaluations were also performed to determine if the licensee effectively implemented operator requalification guidelines established in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors." The inspectors also reviewed and evaluated the adequacy of the licensee's simulation facility for use in operator licensing examinations. The inspectors observed three crews during the performance of the operating tests. Documentation reviewed included written examinations, JPMs, simulator scenarios, licensee procedures, on-shift records, licensed operator qualification records, watchstanding and medical records, simulator modification request records and performance test records, the feedback process, and remediation plans. Documents reviewed during the inspection are listed in the Attachment.

Following the completion of the annual operating examination testing cycle which ended on December 9, 2003, the inspectors reviewed the overall pass/fail results of the individual JPM operating tests, and the simulator operating tests administered by the licensee during the operator licensing requalification cycle. These results were compared to the thresholds established in NRC Inspection Manual Chapter 0609 Appendix I, Operator Requalification Human Performance Significance Determination Process.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

Periodic Evaluation (Biennial)

The inspectors reviewed the licensee's Maintenance Rule periodic assessment, "BNP Maintenance Rule Program Periodic Self-Assessment Plan," for June 1, 2001 to May 31, 2003, dates of assessment July 21-24, 2003, while on-site the week of September 14, 2003. The report was issued to satisfy paragraph (a)(3) of 10 CFR 50.65, and covered the period as indicated for two units. The inspection was to determine the effectiveness of the assessment and that it was issued in accordance with the time requirement of the Maintenance Rule (MR) and included evaluation of: balancing reliability and unavailability, (a)(1) activities, (a)(2) activities, and use of industry operating experience. To verify compliance with 10 CFR 50.65, the inspectors reviewed selected MR activities covered by the assessment period for the following MR systems: containment isolation valves, radiation monitors, main steam isolation valves, instrument air system, and isolated-phase bus duct. Specific procedures and documents reviewed are listed in the Attachment.

During the inspection, the inspectors reviewed selected plant WO data, the site guidance implementing procedure, discussed and reviewed relevant corrective action issues (ARs/CRs), reviewed generic operations event data, probabilistic risk data, and discussed issues with system engineers. Operational event information was evaluated by the inspectors in its use in MR functions. The inspectors selected WOs, and MR assessments, and other corrective action documents of systems recently removed from 10 CFR 50.65 a(1) status and those in a(2) status for some period to assess the justification for their status. The documents were compared to the site's MR program criteria, and the MR a(1) evaluations and rule related data bases.

Routine Maintenance

For the 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 evaluated licensee work controls or practices to assess whether these activities contributed to the degraded performance or condition. 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 110705 Actuation of the Unit 2 high pressure coolant injection system steam exhaust rupture discs during the performance of testing following a system maintenance outage
- AR 110948 Radioactive waste effluent radiation monitor alarmed and automatically secured detergent drain tank discharge

b. Findings

Introduction. A Green NCV was identified for failure to position a Unit 2 HPCI system valve as required by a clearance order following maintenance activities.

Description. During the performance of Unit 2 HPCI system testing following maintenance activities on November 12, 2003, the HPCI system automatically isolated on a primary containment Group 4 isolation signal as a result of high pressure as measured between the two (series mounted) turbine exhaust line rupture discs. High turbine exhaust pressure resulted in a HPCI turbine trip, exhaust rupture disc failures, and the actuation of the HPCI room carbon dioxide system due to the resultant room high temperatures. The licensee found that the high turbine exhaust pressure was caused by the turbine exhaust line to suppression pool stop check valve (2-E42-F021) improperly being left in the closed position following the completion of maintenance activities earlier that day. Following damage assessment and repairs caused by the event, the HPCI system was declared operable on November 16, 2003.

The licensee determined the root cause to be the failure of auxiliary operators performing the restoration lineup to properly check that valve 2-E41-F021 was in the open position in accordance with plant practices. They failed to position the valve in the open position in accordance with system restoration Clearance Order 60551. Administrative Procedure 0AO-013, Plant Equipment Control, Revision 9, directs operators to first stroke a valve in the close direction, and then return the valve to the fully open position, when checking valves in the open position. Valve 2-E41-F021 has an impacting type handwheel which allows the valve to be positioned with the aid of valve handwheel inertia. The auxiliary operators were unfamiliar with this valve operational feature and, as a result, were unsuccessful in moving the valve stem when they attempted to turn the handwheel without the aid of impact. The operators deduced that the valve was already in the open position based on the inability to turn the valve in the open direction and the appearance that the valve was open based on valve stem position.

In addition, the licensee's investigation revealed other human performance-based problems including: 1) one auxiliary operator performing the check did not attend the prejob brief; 2) the prejob brief was limited in scope leaving the attending auxiliary operator uncertain of valve locations, which contributed the position check being performed concurrently; and 3) the auxiliary operators did not consult supervision when they performed the check concurrently, versus independently, which was also not in accordance with licensee expectations. The licensee planned corrective actions to address the identified issues.

Analysis. The failure to position HPCI system valve 2-E41-F021 in accordance with Clearance Order 60551 following maintenance activities is greater than minor because it is associated with system configuration control and affected the mitigating availability of the HPCI system. This finding was determined to be of very low safety significance (Green) because the HPCI system was returned to an operable status within the TS allowed outage time. The finding was related to the cross-cutting aspect of Human Performance because the cause was determined to be due to plant operators using improper techniques in verifying the valve's position. Other contributing causes including operator knowledge deficiencies of valve operation, failure to perform an independent check of valve position, and the pre-job brief's limited scope were also related to Human Performance.

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 equipment control (e.g., locking and tagging). Equipment control Clearance Order 60551 required HPCI system valve 2-E41-F021 (turbine exhaust line to suppression pool stop check valve), to be in the open position following maintenance activities on the HPCI system on November 12, 2003. Contrary to Clearance Order 60551, valve 2-E41-F021 was left in the closed position following the completion of maintenance activities on November 12, 2003. Because this failure to follow Clearance Order 60551 is of very low safety significance and has been entered into the licensee's corrective action program (AR 110705), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000324/2003006-01, Failure to Position HPCI System Valve in Accordance with Clearance Order.

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. 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 maintenance WOs for the following OOS equipment or conditions, and the documents listed in the Attachment:

- WO 59720 Service water intake structure bay cleaning
- AR 108100 EDG 4 inoperable due to failed low lubricating oil pressure switch relay (2-DG4-LPSCR)
- WO 473659 Vital battery 1B-1 declared inoperable due to low voltage on cell #10 during Work Week 41
- AR 110705 Unit 2 HPCI system steam exhaust rupture disc failure and restoration to operable status

- AR 110399 Elevated integrated core damage probability on Unit 2 due to reactor feed pump speed control system power supply sensitivity to voltage fluctuations
- AR 105246 Unit 2 power reduction to repair main condenser tube leak
- WO 464194 Repair motor-driven fire pump breaker compartment

To assess the licensee's identification and resolution of problems, the inspectors reviewed AR 112681 associated with inconsistent risk evaluations for EDG wipedowns, and AR 112544 which documented issues associated with risk reduction compensatory actions during the implementation of modifications on the Unit 2 reactor feed pump speed control system.

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 operating crews' performance during the following unplanned transient/abnormal conditions to verify the response to the event 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.

- Operating crew performance and reactivity management during portions of the Unit 2 down power and power escalation to repair a main condenser tube leak occurring September 23-25, 2004.

- Unit 2 reactor scram due to failure of the main generator exciter occurring on November 4, 2003. Plant response to the failure resulted in the load shedding of various plant loads, the tripping of the reactor feed pumps, and the loss of the normal decay heat removal heat sink (main condenser) due to the receipt of a primary containment isolation of Group 1 (i.e., main steam line isolation).

- Operating crew performance and reactivity management during portions of the Unit 2 control rod pull to criticality following repairs to the main generator exciter on November 7, 2003.

- Unit 2 HPCI system primary containment isolation and fire protection system activation occurring on November 12, 2003. The transient resulted in challenges to operators including unexpected isolation indications. Additionally, adverse atmospheric conditions in the reactor building required deferral of fire protection compensatory measures.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the operability evaluations associated with the following six issues, 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. The inspectors reviewed the documents listed in the Attachment.

- AR 70787 Unit 2 residual heat removal system loop B pressurization due to leakage past inboard low pressure coolant injection isolation valve (2-E11-F015B)
- AR 109435 Vital battery 1A-2, cracked positive plate discovered cell #50
- AR 110037 Unit 2 standby gas treatment train A failure to start as required following the Unit 2 reactor scram occurring on November 4, 2003
- AR 105510 Foreign material found in EDG 4 inter-cooler end cap
- AR 111812 Unit 2 residual heat removal system pipe support (2-E11-34FH88) broken
- AR 110948 Radioactive waste radiation monitor Hi-Hi alarm received while discharging the detergent drain tank

To assess the licensee’s identification and resolution of problems, the inspectors reviewed AR 109291, associated with the loss of standby liquid control loop B squib valve continuity on Unit 2, while working in the A loop circuit.

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 review was to evaluate 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's reviewed.

The OWAs reviewed were:

- 1093 Interlock between reactor building doors 402 and 403 is broken
- 1028 Low pressure core injection line is pressurizing due to reactor coolant inleakage

Cumulative Effects Review

The inspectors reviewed the cumulative effects of all identified operator work-arounds and their: 1) impact on the reliability, availability, and potential for misoperation of the effected systems; 2) 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.

The inspectors held discussions with the OWA coordinator and reviewed the OWA database to determine their cumulative effects. The effect of the work-arounds on reliability, availability, and potential misoperations of the systems involved were reviewed. The inspectors reviewed the OWAs on Unit 1 and Unit 2 to verify that no increase in initiating event frequency occurred and that the OWA could not affect multiple mitigating systems. The cumulative effects of OWAs on operators' correct and timely response to plant transients and accidents were also reviewed by the inspectors.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed a permanent plant modification documented in Engineering Change (EC) 46822 that modified the Unit 2 reactor feed pump speed control system with a digital speed control system. One purpose of the review was to verify that the modification met the design bases and the design assumptions. Another purpose was to verify that modification implementation did not impair emergency/abnormal operating procedure actions and key safety functions. The inspectors also reviewed the modification to verify that unintended system interactions would not occur, and that no additional failure modes were introduced.

b. Findings

Introduction. An unresolved item (URI) was identified for an inadequate design review of a modification implemented on the Unit 2 reactor feed pump speed control system. This is a URI pending completion of the SDP.

Description. During the Spring 2003 Unit 2 outage, the licensee implemented a modification to the reactor feed pump speed control system. This modification, EC 46822, replaced the existing mechanical-hydraulic speed control system with a digital speed control system (Woodward TMR 5009). During investigation as to the cause of the reactor feed pumps tripping during the November 4, 2003 reactor trip, (See Section 40A3.2) the licensee determined that a trip of both reactor feed pumps would occur following Unit 2 turbine trips. The licensee found that the digital speed control system power supplies (two auctioneered for each reactor feed pump) were designed to sense a fault condition within one cycle of abnormal supply voltages. The power supplies would fault, and thus cease to supply output power, if incoming voltage was sensed greater than 132 volts (AC) or less than 88 volts (AC). Simultaneous faults in the power supplies would result in the reactor feed pump tripping.

The speed control system power supplies ultimately receive power from the 2C and 2D Buses. These buses are provided with an automatically initiated, automatically executed, quick, dead bus transfer. The scheme is capable of quickly transferring each bus section and its loads from the normal source (Unit Auxiliary Transformer) to the preferred source (Startup Auxiliary Transformer) in the event of a loss of the normal power source or unit/turbine trip. This transfer results in the buses being disconnected from both voltage sources for a period of between one and five cycles, per the UFSAR. The licensee concluded that there was a high probability that reactor feed pump speed control power supplies would fault during the period that the 2C and 2D are disconnected from both voltage sources due to the sensitivity of the power supplies to detect abnormally low voltage (i.e., less than 88 volts for 1 cycle). As a result, the reactor feed pumps would trip following unit/turbine trips and during certain voltage transients on the 2C and 2D Buses. The licensee's evaluation of the modification failed to recognize this vulnerability.

The licensee implemented compensatory work risk measures because of the resultant elevated integrated core damage probability introduced by the reactor feed pump speed control system vulnerabilities. The licensee promptly initiated corrective actions (AR 110399) and developed a modification to supply the reactor feed pump control system with an uninterruptible power source.

On December 7, 2003, the licensee completed the modifications to the reactor feed pump speed control system, which eliminated the vulnerabilities introduced by EC 46822. The licensee also plans to include uninterruptible power sources to the reactor feed pump governors planned to be modified on Unit 1 in the Spring 2004 refueling outage.

Analysis. The inadequate design review of the Unit 2 reactor feed pump speed control system modification (EC 46822) affects the Initiating Events and Mitigating System cornerstones. This issue is greater than minor because if left uncorrected, it would increase the likelihood of initiating events caused by a loss of reactor feed pumps following transients and affect the reliability and functional capability of the reactor feed pumps to mitigate events. The condition existed since Unit 2 startup on April 6, 2003 until completion of a modification to install an uninterruptible power source to the system on December 7, 2003. The dominant core damage sequence of the Significance Determination Process Phase 2 analysis was Transients without Power Conversion System. Because the finding increased the likelihood of transients with loss of reactor feed pumps and because the reactor feed pumps would not be available to mitigate events, the Phase 2 analysis determined that this finding has potential safety significance greater than very low significance.

Enforcement. No violation of regulatory requirements occurred because the reactor feed pumps are not classified as safety-related and the UFSAR does not credit the reactor feed pumps for abnormal operating occurrence or accident mitigation. This issue is unresolved pending determination of the safety significance and is identified as URI 05000324/2003006-02, Unit 2 Reactor Feed Pump Speed Control System Modification.

1R19 Post-Maintenance Testing

a. Inspection Scope

For the 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. Documents reviewed are listed in the Attachment.

- WO 474971 Repair EDG 4, lubricating oil low pressure switch relay (2-DG4-LPSCR)
- WO 245950 Unit 1A nuclear service water pump discharge check valve refurbishment
- WO 464194 Repair motor-driven fire pump breaker/compartment
- WO 469646 Repair EDG 2 jet assist solenoid valve (2-DG2-6552-2)

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 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. The inspectors reviewed the documents listed in the Attachment.

- Maintenance Surveillance Test 2MST-DG22R, EDG 4 Trip Bypass Logic Test
- Periodic Test 2PT-01.7, Heatup/Cooldown Monitoring, following Unit 2 reactor trip on November 4, 2003
- Periodic test 0PT-20.3, Local Leak Rate Testing, performed on Unit 2 low pressure coolant inboard injection valve (2-E11-FO15B)
- Periodic Test 0PT-12.3.2.B, Number 2 Diesel Generator Starting Air Valve Operability Test

Inservice Surveillance Testing

The inspectors reviewed the performance of Periodic Test 0PT-09.2, High Pressure Coolant Injection System 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 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 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. Documents reviewed are listed in the Attachment.

- ECs 5597 & 45694 - Review temporary shielding installation on the residual heat removal (RHR) and primary containment systems

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP2 Alert and Notification System Testing

a. Inspection Scope

The inspectors ascertained the licensee’s commitments with respect to the testing and maintenance of the alert and notification system (ANS), which comprised 36 sirens in the ten-mile-radius emergency planning zone (31 in Brunswick County, 5 in New Hanover County). The inspectors evaluated the design of the ANS, the licensee’s methodology for testing the system, and the adequacy of the testing program design. Assessment of the program as actually implemented included review of siren test records (with an emphasis on identification of any repetitive individual siren failures), system changes during the past two years, procedures for periodic preventative maintenance (including post-maintenance testing), and a sample of corrective actions and their effectiveness for siren failures and issues. The review of this program area encompassed the period January 2002 through November 2003. Licensee procedures, records, and other documents reviewed within this inspection area are listed in the Attachment.

b. Findings

No findings of significance were identified.

1EP3 Emergency Response Organization (ERO) Augmentation

a. Inspection Scope

The inspectors identified the licensee's commitments with respect to timeliness and numbers of personnel for staffing emergency response facilities (ERFs) in the event of an emergency declaration at Alert or higher. The licensee's automated paging system and manual backup system for call-out of ERO personnel were reviewed to determine whether they would support staff augmentation in accordance with the criteria for ERF activation timeliness. Methodologies for testing the primary and backup systems for augmenting the ERO were reviewed and discussed with cognizant licensee personnel. The inspectors also reviewed and discussed the changes to the augmentation system and process during the past two years. The inspectors reviewed records of the last off-hour ERO augmentation drill which involved actual travel to the plant and activation of ERFs (conducted on April 23, 2003). Records of ERO pager tests (the backup system for ERO notification) were reviewed. Follow-up activities for a sample of problems identified through augmentation testing were evaluated to determine whether appropriate corrective actions were implemented. Licensee procedures, records, and other documents reviewed within this inspection area are listed in the Attachment.

b. Findings

No findings of significance were identified.

1EP4 Emergency Action Level (EAL) and Emergency Plan Changes

a. Inspection Scope

The inspectors reviewed a selected sample of changes made to the Emergency Response Plan (ERP) since the last inspection in this program area (conducted in November 2002) against the requirements of 10 CFR 50.54(q) to determine whether any of the changes decreased ERP effectiveness. The subject changes, which were incorporated in ERP Revisions 60, 61, and 62, did not include modifications to the emergency action levels (EALs). The inspectors reviewed documentation of the licensee's 10 CFR 50.54(q) screening evaluations for Revisions 60 and 62. Licensee procedures, records, and other documents reviewed within this inspection area are listed in the Attachment.

b. Findings

No findings of significance were identified.

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies

a. Inspection Scope

The inspectors evaluated the efficacy of licensee programs that addressed weaknesses and deficiencies in emergency preparedness. The procedure governing the plant corrective action program was reviewed for applicability to the Emergency Preparedness Program. Since the last inspection of this program area (conducted in November 2001), one emergency declaration (a Notification of Unusual Event [NOUE]) was made by the licensee, as a result of the projected threat from Hurricane Isabel on September 16, 2003. The inspectors reviewed event documentation to assess the adequacy of implementation of ERP requirements, as well as the licensee's self-assessment of ERO performance during the event. The inspectors evaluated selected drill scenarios and associated critiques to determine whether the licensee had properly identified failures to implement regulatory requirements and planning standards. A sample of weaknesses and deficiencies identified by means of these licensee processes was evaluated to determine whether corrective actions were effective and timely. Licensee procedures, records, and other documents reviewed within this inspection area are listed in the Attachment.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

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 October 2002 through September 2003. 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

- Reactor Coolant System Specific Activity
- Reactor Coolant System Leak Rate

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.

Emergency Preparedness Cornerstone

- Emergency Response Organization (ERO) Drill/Exercise Performance
- ERO Drill Participation
- Alert and Notification System Reliability

For the specified review period, the inspectors examined data reported to the NRC, procedural guidance for reporting PI information, and records used by the licensee to identify potential PI occurrences. The inspectors verified the accuracy of the PI for ERO drill and exercise performance through review of a sample of drill and event records. The inspectors reviewed selected training records to verify the accuracy of the PI for ERO drill participation for personnel assigned to key positions in the ERO. The inspectors verified the accuracy of the PI for alert and notification system reliability through review of a sample of the licensee's records of periodic system tests. The inspectors also interviewed the licensee personnel who were responsible for collecting and evaluating the PI data. Licensee procedures, records, and other documents reviewed within this inspection area are listed in the Attachment.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution

a. Inspection Scope

The inspectors performed an in-depth annual sample review of a selected AR 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 issue and associated corrective actions were reviewed:

- AR 110705 Actuation of the Unit 2 high pressure coolant injection system steam exhaust rupture discs during the performance of testing following a system maintenance outage

b. Findings and Observations

No findings of significance were identified.

4OA3 Event Follow-up

.1 All Oscillation Power Range Monitors (OPRMs) Declared Inoperable

a. Inspection Scope

On October 5, 2003, the licensee received a 10 CFR 21 notification from General Electric that the Units 1 and 2 OPRMs have the potential for numerous, unexpected confirmation count resets in the event of a reactor power instability condition, and were therefore inoperable. The inspectors reviewed the licensee actions to verify proper response in accordance with plant TS. The inspectors also reviewed the initial 10 CFR 50.72 notification to assess for appropriate reporting with established criteria.

b. Findings

No findings of significance were identified.

.2 Unit 2 Reactor Scram

a. Inspection Scope

The inspectors reviewed the licensee’s action in response to a Unit 2 main turbine trip and reactor scram due to main generator loss of field that occurred on November 4, Unit 2. The loss of field was caused by the failure of the main generator alternator brush/collector ring. During the scram all safety systems operated properly with the exception of the 2A standby gas treatment train failing to start, the isolation of containment isolation group 1 (main steam isolation), and the trip of the reactor feed pumps. The licensee determined momentary degraded voltage on the AC emergency buses (about 40% of rated) caused the relays associated with the group 1 isolation (resulting in a loss of normal decay heat removal) and standby gas treatment fire protection features to drop out. For further discussion of the trip of the reactor feed pumps, see Section 1R17. In addition, the inspectors reviewed Operating Instruction (OI) 0OI-01.06, Post Trip Review to verify the initial data gathering, equipment response and post-trip review were conducted in accordance with the procedure requirements. The inspectors also reviewed the initial 10 CFR 50.72 notification to verify proper reporting with established criteria. The licensee entered this event into their corrective action program as AR 109923. Licensee personnel performance is discussed in Section 1R14.

b. Findings

No findings of significance were identified.

.3 HPCI Exhaust Rupture Disc Failures

a. Inspection Scope

The inspectors reviewed the licensee's response to a Unit 2 HPCI system isolation and room fire protection actuation occurring on November 12, 2003. The actuations occurred in response to the HPCI turbine exhaust line rupture discs actuating during system testing. See Section 1R12.1 for further discussion of the rupture discs actuating. The inspectors reviewed the initial 10 CFR 50.72 notification to verify proper reporting with established criteria. The licensee entered this event into their corrective action program as AR 110705.

b. Findings

No findings of significance were identified.

.4 New Hanover County Sirens

a. Inspection Scope

The inspectors reviewed the licensee's actions in response to the New Hanover County emergency sirens not responding from the county emergency operations center on November 12, 2003. The sirens were subsequently tested from the Emergency Offsite Facility. The cause was determined to be due to radio frequency interference. Testing to demonstrate the sirens could be successfully initiated from the county facility was completed. The inspectors reviewed the licensee's 10 CFR 50.72 notification against established reporting criteria. The licensee entered the event into the corrective action program as AR 110623.

b. Findings

No findings of significance were identified.

4OA4 Cross Cutting Aspects of Findings

Section 1R12 describes a finding for the failure to position a HPCI system valve in accordance with a clearance order following maintenance activities. The finding is related to the cross-cutting aspect of Human Performance because the cause was determined to be due to plant operators using improper techniques in verifying the valve's position. Other contributing causes including operator knowledge deficiencies of valve operation, failing to perform an independent check of valve position, and the pre-job brief's limited scope were also related to Human Performance.

4OA6 Meetings, Including Exit

On December 18, 2003, 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.

4OA7 Licensee-Identified Violations

The following violation of very low safety significance (Green) was identified by the licensee and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV.

10 CFR 50.74 requires in part that each licensee shall notify the commission in accordance with section 50.4 within 30 days of the following in regard to a licensed operator or senior operator ... (c) permanent disability or illness as described in 10CFR 55.25 of this chapter. Contrary to this, in June of 2001 a licensed operator had a change in medical condition as described in ANSI/ANSA 3.4-1983, that was not reported to the commission within 30 days. This finding was identified by the licensee during an audit of medical records in April 2003. The NRC was notified of this finding in a letter dated April 21, 2003. The licensed individual was administratively restricted to “no solo” operation on April 4, 2003. The finding is of very low safety significance because records indicate that the individual did not stand “solo” watch while performing licensing duties after the change in medical condition occurred. This issue is documented in licensee correction action request number 8992.

ATTACHMENT : SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

E. Atkinson, Supervisor - Emergency Preparedness
A. Brittain, Manager - Security
E. Conway, Senior Nuclear Security Specialist
W. Dorman, Manager - Nuclear Assessment
C. Elberfeld, Lead Engineer, Technical Support
C. Gannon, Site Vice President (former Director - Site Operations)
J. Gawron, Training Manager
D. Hinds, Plant General Manager (former Engineering Manager)
J. Keenan, Past Site Vice President
D. Makosky, Lead Nuclear Security Specialist
W. Noll, Director - Site Operations (former Plant General Manager)
E. O'Neil, Manager - Site Support Services
E. Quidley, Manager - Outage and Scheduling
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

05000324/2003006-02	URI	Unit 2 Reactor Feed Pump Speed Control System Modification (Section 1R17)

Opened and Closed

05000324/2003006-01	NCV	Failure to Position HPCI System Valve in Accordance with Clearance Order (Section 1R12)

Closed

NONE

Discussed

NONE

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Plant Operating Manual (POM), Volume VII, Operating Instruction 0OI-01.03, Non-Routine Activities
POM, Volume XII, Preventive Maintenance, 0PM-HT001, Preventive Maintenance on Plant Freeze Protection and Heat Tracing System
System Description SD-53, Freeze Protection and Heat Tracing System

Section 1R04: Equipment Alignment

Procedures

Administrative Procedure ADM-NGGC-0106, Configuration Management Program Implementation
POM, Volume III, 2OP-10, High Pressure Coolant Injection System Operating Procedure
POM, Volume III, 1OP-43, Service Water System Operating Procedure
POM, Volume III, 2OP-16, Reactor Core Isolation Cooling System Operating Procedure

Section 1R05: Fire Protection

Procedures

POM, Volume XIX, Prefire Plan, 1PFP-RB and 2PFP-RB, Reactor Building Prefire Plans
POM, Volume XIX, Prefire Plan, 0PFP-DG, Diesel Generator Building Prefire Plans

Reports

Analysis No. BNP-E-9.004, Safe Shutdown Analysis Report

Section 1R06: Flood Protection Measures

Miscellaneous Documents

UFSAR section 3.4.1, Protection From External Flooding
UFSAR section 3.4.2, Protection From Internal Flooding
Design Basis Document (DBD)-106, Hazards Analysis
DBD-105, Postulated Pipe Failure
BNP Maintenance Rule a(1) System Action Plan For Site Cables Within Manholes (system 5259)
Maintenance Rule Expert Panel Meeting Minutes System 5259 (11/19/02 - 11/19/03)

Work Orders

47043, Functional test of flood status level switches for DG 1 and 2 fuel oil tank rooms
47102, Functional test of flood status level switches for DG 4 fuel oil tank room
45871, Functional test of DG 1 pipe trench level switches
45867, Functional test of DG 2 pipe trench level switches
172678,Functional test of DG 4 pipe trench water level switches
46974, Functional test of service water intake structure level switch
46091, Functional test of flood status level switches (Unit 1 core spray (CS) rooms, residual heat removal (RHR) rooms, and high pressure coolant injection (HPCI) room)
180508, Functional test of flood status level switches (Unit 2 CS rooms, RHR rooms, and HPCI room)

Corrective Action Documents

54376, DG1 jacket water cooler service water supply valves leak by
66048, Functional test of diesel generator building cell trench flood level switches
88221, Storm drain basin overboard valves open due to weather
94558, Pump unplugged allowing floor area to flood
98218, Oil and water leak on floor of heater drain pump room Unit 1

Procedures

0AI-68, Brunswick Nuclear Plant Response to Severe Weather Warnings
0AOP-13.0, Operation During Hurricanes
0MST-Flood11Q, Flood Protection Intake Canal Level Channel Functional
0OP-47, Floor and Equipment Drain System Operating Procedure

Alarm Response Procedures

1(2)APP-UA-01 3-8, SW INTAKE STRC SUMP LEVEL HI-HI
1(2)APP-UA-28 3-5, SW INTAKE STRUCTURE FLOOD LVL HI
1(2)APP-UA-24 6-8, INTAKE CANAL FLOOD LEVEL HI
1(2)APP-UA-06 3-3, DRAINAGE BASIN TROUBLE
1(2)APP-UA-12 2-1, NORTH CS RM FLOOD LEVEL HI
1(2)APP-UA-12 1-1, NORTH CS RM FLOOD LEVEL HI-HI
1(2)APP-UA-12 2-3, SOUTH CS RM FLOOD LEVEL HI
1(2)APP-UA-12 1-3, SOUTH CS RM FLOOD LEVEL HI-HI
1(2)APP-UA-12 2-2, NORTH RHR RM FLOOD LEVEL HI
1(2)APP-UA-12 1-2, NORTH RHR RM FLOOD LEVEL HI-HI
1(2)APP-UA-12 2-4, SOUTH RHR RM FLOOD LEVEL HI
1(2)APP-UA-12 1-4, SOUTH RHR RM FLOOD LEVEL HI-HI
1(2)APP-UA-12 2-5, HPCI ROOM FLOOD LEVEL HI
1(2)APP-UA-12 1-5, HPCI ROOM FLOOD LEVEL HI-HI
1(2)APP-UA-28 3-6, DG NO 1 FUEL TNK RM FLOOD LVL HI
1(2)APP-UA-28 3-7, DG NO 2 FUEL TNK RM FLOOD LVL HI
1(2)APP-UA-28 3-8, DG BLDG BASEMENT FLOOD LVL HI
1(2)APP-UA-28 4-5, DG BLDG VALVE PIT FLOOD LVL HI
1(2)APP-UA-28 4-6, DG NO 3 FUEL TNK RM FLOOD LVL HI

1(2)APP-UA-28 4-7, DG NO 4 FUEL TNK RM FLOOD LVL HI
1(2)APP-UA-28 5-6, TURB BLDG PIPE TUNNEL FLOOD LVL HI
1(2)APP-UA-28 5-7, TURB BLDG PIPE TUNNEL FLOOD HI-HI
1(2)APP-UA-28 5-8, DG NO 1 PIPE TRENCH FLOOD LEVEL HI
1(2)APP-UA-28 5-9, DG NO 2 PIPE TRENCH FLOOD LEVEL HI
1(2)APP-UA-28 6-8, DG NO 3 PIPE TRENCH FLOOD LVL HI
1(2)APP-UA-28 6-9, DG NO 4 PIPE TRENCH FLOOD LVL HI

Section 1R07: Heat Sink Performance

Procedures

POM, Volume X, OMST-DG501R3, Emergency Diesel Generators 72 Month Inspection

Work Order

WO 453062-01, 2-DG4-ENG Turbocharger Intercooler Cover

Section 1R11: Licensed Operator Requalification

Procedures and Logs

POM, Volume I, Book 3, Training Program Procedure 0TPP-200, Licensed Operator Continuing Training Program
Conduct of Examinations, TAP - 403, Rev. 6
Initial Licensing and Continued Training - Annual/Biannual Examination Development, TAP-411
License Activation and Maintenance; 0OI-01.05, Rev. 10
Licensed Operator Continued Training Program, TAP - 200, Rev. 2
Logs Recent Changes to BNP Simulator Logs
Medical Records
Operation Logs
Reactivation Records
Reactivity Anomaly Check; OPT-14.5.2, Rev. 34
Simulator Change Reports
Simulator Operation and Maintenance, TAP-412, Rev. 0
Simulator Service Request (SSR)
Simulator Test Procedure; STP-SS-002; 50% Power Steady State Comparison, Rev. 9
Simulator Program, OTPP-206, Rev. 1
Training Administration Procedure (TAP) 409, Conduct of Simulator Training and Evaluation
U1 Core Model Upgrade Report (B1C14)
U2 Core Model Upgrade Report (B2C16)

Section 1R12: Maintenance Effectiveness

Corrective Action (CR/AR) Reports

5051, instrument air repetitive MPFFs
5120, MSIV 1A PMG MR A(1)
22366, missed A(1) Assessment for system 5045
46853, isolated-phase bus system is not monitored in accordance with plan
46855, missed unavailability events
46857, three functional failures not logged in the MR data base
46861, AR investigation of improperly answered MR functional failure question
47884, coupling failed on Unit 1 service air dryer
48428, system 3080 has a condition monitoring criteria that was not being monitored
48432, scoping for Nuclear Boiler Instrumentation not adequate for assigning functional failures
48443, a missed functional failure in system 1080 was identified
54595, binding of transmitter 2-B21-LT-N024B-2
58722, 1-B21-F022B would not pressurize
71058, 1-iso-ph-cool-fan-2 tripped
86919, 2-B21-F028A would not pressurize

Administrative Procedures

NGGC-ADM-0101, Maintenance Rule Program, Rev. 16

Miscellaneous

BNP-PSA-056, PSA Evaluation of Maintenance Rule Performance Criteria, Unit 0, Rev. 0
Self Assessment Report 27639, dates July 23-26, 2001
Self Assessment Report 27639, dates July 22-25, 2002

1R13: Maintenance Risk Assessments and Emergent Work Evaluation

Procedures

POM, Volume 0AI-81, Water Chemistry Guidelines
Procedure OAP-025, BNP Integrated Scheduling
Technical Requirements Manual (TRM) 5.5.13, Configuration Risk Management Program

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

Procedures

Emergency Operating Procedure, 2EOP-01-RSP, Reactor Scram Procedure
POM Volume IV, General Operating Procedure, 0GP-2, Approach to Criticality and
Pressurization of the Reactor

Section 1R15: Operability Evaluations

Procedures

POM, Volume X, 0MST-BATT11Q, Batteries, 125 VDC, Quarterly Operability Test
POM, Volume XII, 0MMM-055, Cleanliness and Flushing Requirement

Section 1R19: Post Maintenance Testing

Procedures

POM, Volume X, OPT-34.1.1.0, Fire Pump Test (Motor-Driven and Engine-Driven)
POM, Volume X, 1PT-24.1-1, Service Water Pump and Discharge Valve Operability Test
POM, Volume X, 2MST-DG22R, DG-4 Trip Bypass Logic Test
POM, Volume XII, 0MMM-055, Cleanliness and Flushing Requirements
POM, Volume XII, 0MMM-015, Operation and Inspection of Cranes and Material Handling Equipment
POM, Volume XII, 0CM-VCK506, Technocheck Check Valves

Section 1R22: Surveillance Testing

Procedures

POM, Volume X, 0PT-20.7B, Pressure Isolation Valve Leak Rate Test in Conjunction with RPV Pressure Test

Section 1R23: Temporary Plant Modifications

Engineering Service Requests

ESR 00-00291, Rev. 0, Radioactive Floor Drains System
ESR 00-00291, Rev. 1, Fuel Pool Cooling & Supplemental Cooling

Sections 1EP2 - 1EP5: Reactor Safety—Emergency Preparedness

Plans and Procedures

BNP Siren Preventative Maintenance Checklist (no effective date or procedure number)
BNP Maintenance and Testing of Sirens - Customer/Supplier Agreement, 06/27/2003
CAP-NGGC-0200, Corrective Action Program, Rev. 9
Emergency Response Plan, Rev. 60 (effective 02/19/2003), Rev. 61 (effective 06/03/2003), and Rev. 62 (effective 07/31/2003)
0PEP-02.1.1, Emergency Control - Notification of Unusual Event, Alert, Site Area Emergency, and General Emergency, Rev. 9
0PEP-02.6.21, Emergency Communicator, Rev. 42
0PEP-02.6.26, Activation and Operation of the Technical Support Center, Rev. 13
0PEP-02.6.27, Activation and Operation of the Emergency Operations Facility, Rev. 15
0PEP-03.8.2, Personnel Accountability and Evacuation, Rev. 16

0PEP-04.2, Emergency Facilities and Equipment, Rev. 27
0PEP-04.3, Performance of Training, Exercises, and Drills, Rev.16
0PEP-04.7, Brunswick Emergency Notification (Automated Telephone) System, Rev. 4
REG-NGGC-0010, 10 CFR 50.59 and Selected Regulatory Reviews, Rev. 5
Siren Control System Manual (Motorola document, June 1999)

Records and Data

BNP Siren Preventative Maintenance Checklists dated 11/05/2002 (sirens 27, 28, 29) and 07/07/2003 (siren 10)
CP&L Brunswick Steam Electric Plant Siren System Test Reports for biweekly silent tests (01/10/2002 - 12/01/2003), quarterly growl tests (01/07/2002, 04/08/2002, 07/08/2002, 10/07/2002, 01/13/2003, 04/14/2003, 07/17/2003, 10/13/2003) and annual full-volume tests (11/06/2002, 11/12/2003)
Critique Report on 04/23/2003 Augmentation Drill, 04/25/2003
09/17/2003 NOUE Critique Report–Hurricane Isabel, 10/24/2003
Records of ERO pager tests, 01/2002 - 11/2003 (performed monthly through 06/2003, then quarterly)
REG-NGGC-0010, Attachment 2, 10 CFR 50.54(q) Emergency Preparedness Program Evaluation for ERP Rev. 60 (ID No. 03-0170) and Rev. 62 (ID No. 03-1007)
Unit 2 Operator Log for September 16-18, 2003

Audits and Self-Assessments

Nuclear Assessment Section (NAS) Assessment No. B-EP-02-01, Emergency Preparedness Assessment, 01/06/2003
Self-Assessment No. AR 79474, Emergency Preparedness Response to Security Events (conducted 06/23-26/2003)
Self-Assessment No. AR 79476, Evaluate Effectiveness of 2003 EP Team Training Drill Cycle (conducted 10/27-30/2003)

Action Requests (Corrective Action Documents)

Action Request (AR) 79138, NAS Assessment No. B-EP-02-01 identified examples where positions were staffed with non-qualified individuals, 12/10/2002
AR 00083236, CPLDOSE problems on Control Room computers, 01/30/2003
AR 00083532, ERO suspensions due to lapsed qualifications, 02/03/2003
AR 00085594, EP drill weakness: Alternate EOF competes for equipment use, 02/15/2003
AR 00091046, Brunswick Emergency Notification System unavailable, 04/21/2003
AR 00095497, EP drill 05/27/2003: Items for management consideration, 06/06/2003
AR 00100005, EP drill 07/15/2003: TSC weakness, 07/17/2003
AR 00110343. Hurricane Isabel: Items for management consideration, 11/04/2003
AR 00110623, Failure of New Hanover sirens during full volume testing, 11/12/2003
AR 00112517, Alert and Notification System process enhancement, 12/04/2003
AR 00112518, Alternate EOF augmentation goals, 12/04/2003
AR 00112529, Evaluate documenting offsite assembly area locations, 12/04/2003

Section 4OA1: Performance Indicator Verification

Procedures

POM, Vol. VII, 1OI-03.1, Control Operator Daily Surveillance Report
POM, Vol. VII, 2OI-03.1, Control Operator Daily Surveillance Report
Reactor coolant radiochemistry logs

Records and Data

CP&L Brunswick Steam Electric Plant Siren System Test Reports for biweekly silent tests (01/10/2002 - 12/01/2003), quarterly growl tests (01/07/2002, 04/08/2002, 07/08/2002, 10/07/2002, 01/13/2003, 04/14/2003, 07/17/2003, 10/13/2003) and annual full-volume tests (11/06/2002, 11/12/2003)
Documentation packages (scenario/time line/event notification forms/critique report) for ERO drills on 10/15/2002 and 05/27/2003
Documentation packages (event notification forms/evaluator critique) for Licensed Operator Continuing Training drills on 02/23/2003, 06/09/2003, 06/16/2003, 06/23/2003, and 06/30/2003
EPL-001, Emergency Phone List, 09/30/2003
ERO participation rosters (source record) for 05/27/2003 drill

Section 4OA3: Event Followup

Procedures

POM, Volume VII, Operating Instructions 0OI-01.07, Notifications, Rev. 11