UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
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April 19, 2004

EA 04-076

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 NOS. 05000325/2004002 AND 05000324/2004002; PRELIMINARY WHITE FINDING

Dear Mr. Gannon:

On March 20, 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 March 25, April 2, and April 19, 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.

As described in Section 4OA2 of the enclosed report, a finding was identified concerning the failure to take adequate corrective action for conditions adverse to quality associated with the No. 3 emergency diesel generator (EDG 3) jacket water cooling (JWC) system. This corrective action failure resulted in EDG 3 being inoperable for a period greater than the Technical Specifications (TS) allowed outage time. Specifically, excessive leakage occurred from the JWC system during EDG 3 operation from December 8, 2003, until it was corrected on January 7, 2004.

This finding was assessed based on the best available information, including influential assumptions, using the Reactor Safety - Significance Determination Process (SDP) and was preliminarily determined to be a White finding for Unit 2 (i.e., a finding of low-to-moderate safety significance, which may require additional NRC inspection) and a Green finding for Unit 1 (very low safety significance). The difference in risk significance between the units is due to differences in electric bus loads. EDG 3 supplies No. 3 emergency bus with electrical power, which in turn powers a substantial amount of the Unit 2 safety-related loads. However, a lesser amount of Unit 1 loads are powered from emergency bus 3. This finding does not present a current safety concern because several missing jacket water cooling system structural supports were reinstalled, the system leak was appropriately repaired, and EDG 3 was restored to an operable status on January 7, 2004.

The finding was also determined to involve three violations of NRC requirements. Two apparent violations were identified for Unit 2, involving the failure to take adequate corrective actions in accordance with 10 CFR 50, Appendix B, Criterion XVI, that resulted in the failure to comply with TS 3.8.1. These two apparent violations are being considered for escalated enforcement action in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600. One non-cited violation was identified for Unit 1. Because the finding was determined to be of very low safety significance for Unit 1, and because it is entered into your corrective action program, the NRC is treating this finding as an NCV consistent with Section VI.A of the NRC Enforcement Policy. If you contest the NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Brunswick Steam Electric Plant. The current Enforcement Policy is included on the NRC's web site at http://www.nrc.gov/what-we-do/regulatory/enforcement/enforce-pol.html.

Before we make a final decision on this matter, we are providing you an opportunity to (1) present to the NRC your perspectives on the facts and assumptions used by the NRC to arrive at the finding and its significance, and whether you agree with the apparent violations at a Regulatory Conference or (2) submit your position on the finding to the NRC in writing. If you request a Regulatory Conference, it should be held within 30 days of the receipt of this letter, and we encourage you to submit supporting documentation at least one week prior to the conference in an effort to make the conference more efficient and effective. If a Regulatory Conference is held, it will be open for public observation. The NRC will also issue a press release to announce the conference. If you decide to submit only a written response, such submittal should be sent to the NRC within 30 days of the receipt of this letter.

Please contact Paul Fredrickson at (404) 562-4530 within 10 business days of the date of your receipt of this letter to notify the NRC of your intentions. If we have not heard from you within 10 days, we will continue with our significance determination decision and you will be advised by separate correspondence of the results of our deliberations on this matter.

Since the NRC has not made a final determination in this matter, no Notice of Violation is being issued for this inspection finding at this time. In addition, please be advised that the number and characterization of apparent violations described in the enclosed inspection report may change as a result of further NRC review.

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

Sincerely,

/RA by L. Wert for/

Victor M. McCree, Director
Division of Reactor Projects

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

Enclosure: Inspection Report 05000325, 324/2004002 w/Attachments: 1. Supplemental Information, and 2. Significance Determination Process Phase III Summary

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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:	05000325/2004002 and 05000324/2004002
Licensee:	Carolina Power and Light (CP&L)
Facility:	Brunswick Steam Electric Plant, Units 1 & 2
Location:	8470 River Road SE Southport, NC 28461
Dates:	December 21, 2004 - March 20, 2004
Inspectors:	E. DiPaolo, Senior Resident Inspector J. Austin, Resident Inspector W. Loo, Senior Health Physicist (Sections 2PS, 4OA1 and 4OA5) R. Hamilton, Health Physicist (Sections 2PS, 4OA1 and 4OA5)) H. Gepford, Health Physicist (Sections 2PS, 4OA1 and 4OA5) J. Lenahan, Senior Reactor Inspector (Section 1R08) K. Van Doorn, Senior Reactor Inspector (Section 1R08)
Approved by:	Victor M. McCree, Director Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000325/2004002, 05000324/2004002; 12/21/2003 - 03/20/2004; Brunswick Steam Electric Plant, Units 1 and 2; Problem Identification and Resolution.

The report covered a three-month period of inspection by resident inspectors, regional reactor inspectors, and regional health physics inspectors. One preliminary White finding involved two apparent violations for Unit 2 and one Green non-cited violation (NCV) for Unit 1 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

TBD (Unit 2), Green (Unit 1). An inspector-identified finding was identified for the failure to take adequate corrective actions for conditions adverse to quality associated with the No. 3 emergency diesel generator (EDG 3) jacket water cooling (JWC) system. This condition resulted in EDG 3 being inoperable from December 8, 2003, until January 7, 2004, which was contrary to the requirements of Technical Specification (TS) Limiting Condition for Operation (LCO) 3.8.1, AC Sources-Operating. Two apparent violations were identified for Unit 2: 10CFR50, Appendix B, Criterion XVI, Corrective Actions; and TS LCO 3.8.1. One non-cited violation was identified for Unit 1: 10CFR50, Appendix B, Criterion XVI.

This finding is greater than minor because it is associated with the availability and reliability of EDG 3 to mitigate events such as a loss of offsite power. The finding was preliminarily determined to have low to moderate safety significance (White) for Unit 2 and determined to have very low safety significance (Green) for Unit 1 because the ability of EDG 3 to mitigate a loss of offsite power event was effected. The difference in risk significance between the units is due to differences in electric bus loads. EDG 3 supplies No. 3 emergency bus with electrical power, which in turn provides a substantial amount of the Unit 2 safety-related loads. However, a lesser amount of load is provided to Unit 1 from emergency bus 3. (Section 4OA2)

B. Licensee Identified Violations

None.

REPORT DETAILS

Summary of Plant Status

Unit 1 began the report period operating at full power. On January 29, 2004, the unit performed an unplanned downpower to approximately 50 percent to facilitate repairs to an electro-hydraulic control fluid leak on the No. 3 main turbine control valve. Also on that date, the unit implemented final feedwater temperature reduction and commenced final coastdown. Maximum power was restored later that day. On February 29, 2004, Unit 1 performed a plant shutdown to commence refueling outage (RFO B115R1). At the end of the inspection period, refueling operations were complete and the unit was in Mode 4 (cold shutdown).

Unit 2 began the report period operating at full power. On December 31, 2003, an unplanned downpower to approximately 70 percent was performed to facilitate repairs to an electro-hydraulic control fluid leak on No. 4 main turbine control valve. The unit returned to full power on January 1, 2004. On January 17, 2004, the unit performed a planned downpower, to approximately 50 percent, to perform secondary plant maintenance, control rod testing and valve testing. During the subsequent power ascension on January 19, 2004, the unit reduced power to approximately 50 percent in response to abnormal indications (power supply faults) on the A reactor feed pump speed control system. On March 12, 2004, reactor recirculation pump 2A tripped on low lubricating oil pressure, due to a failed open pressure regulating valve, which resulted in an unplanned downpower to approximately 50 percent. Following repairs later that day, the unit returned to maximum power 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 demineralized water system and condensate storage tank low level switches would remain functional and available in cold weather conditions (1 sample of 2 systems). 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 heater, etc.) associated with the above systems/components. Licensee problem identification and resolution were also addressed. This included a review of Action Request (AR) 116118, which documented that the Unit 1 condensate storage tank (CST) level instrumentation had been agitated during cold weather monitoring. This caused the reactor core isolation cooling (RCIC) system suction to automatically transfer from the CST to the suppression pool. Documents reviewed are listed in Attachment 1.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

The inspectors performed three partial walkdowns of the below listed systems to verify that the systems were correctly aligned while the redundant train or system was inoperable or out-of-service (OOS), or following system restoration after maintenance. The inspectors assessed conditions such as equipment alignment (i.e., valve positions, damper positions, and breaker alignment) and system operational readiness (i.e., control power and permissive status) that could affect operability. The inspectors verified that the licensee identified and resolved equipment alignment problems that could cause initiating events or impact mitigating system availability. Administrative Procedure ADM-NGGC-0106, Configuration Management Program Implementation, was reviewed by the inspectors to verify that available structures, systems or components (SSCs) met the requirements of the licensee's configuration control program. Documents reviewed are listed in Attachment 1.

- Unit 2 B loop of residual heat removal when A loop was OOS January 28-29, 2004
- Unit 1 residual heat removal while on shutdown cooling following system restoration on March 12, 2004
- Unit 1 supplemental spent fuel pool cooling system when residual heat removal-shutdown cooling system was OOS on March 13, 2004

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors reviewed current ARs and work orders (WOs) associated with the fire suppression system to confirm that their disposition was in accordance with 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 below listed areas important to reactor safety and reviewed the associated Prefire Plans to verify that the requirements for fire protection design features, fire area boundaries, and combustible loading were met. The inspectors reviewed Prefire Plan 0PFP-DG, Diesel Generator Building Prefire Plans (Rev. 8) to determine correct configuration:

- Switchgear rooms E1, E2, E3 and E4 in Diesel Generator Building, 50' elevation (4 areas)

- Switchgear rooms E5, E6, E7 and E8 in Diesel Generator Building, 23' elevation (4 areas)

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities

a. Inspection Scope

Inservice Inspection (ISI)

The inspectors reviewed ISI procedures, observed in-process ISI work activities, and reviewed selected ISI records. The observations and records were compared to the Technical Specifications (TS) and the applicable Code (ASME Boiler and Pressure Vessel Code, Section XI, 1989 Edition, with no Addenda) to verify compliance.

The inspectors observed portions of ultrasonic (UT) examinations performed on four welds to verify the exams were being performed in accordance with the ASME Boiler and Pressure Vessel Code, Section XI, Appendix VIII, 1995 Edition including 1996 Addenda as modified by the Performance Demonstration Initiative (PDI) Program. The exams included an austenitic weld, a ferritic weld, a reactor pressure vessel weld, and a nozzle inner corner radius area. The inspectors also observed the UT exam of a dissimilar metal weld, which was conducted based on a recent industry leak discovered on a nozzle weld with a similar configuration and composition. These inspections included welds 1B32RECIRC-4-A-2, 1E213-1-3-SWA, 1B11-RPV-N9, and 1B11N9-RPV-FW1CRD274 and nozzle 1B11-RPV-N4B-IRS.

The inspectors reviewed non-destructive examination (NDE) reports for visual (VT-3) inspection of 60 hydraulic snubbers and functional testing of five snubbers performed during the current outage. Qualification and certification records for examiners, and equipment for selected examination activities were reviewed. In addition, the inspectors examined snubbers on pipe supports during a walkdown of the Unit 1 drywell. Examination of the snubbers included attachment to supporting structures and piping, fluid levels in reservoirs, absence of fluid leakage from the snubbers, and overall condition of the snubbers. A sample of ISI issues in the licensee's corrective action program were reviewed for adequacy. The inspectors also reviewed the results of a self-assessment of the snubber inspection program completed in November, 2002 and the associated corrective actions to address three assessment findings. Documents reviewed are listed in Attachment 1.

The inspectors reviewed records for the following Code repairs:

- WO 132417, Build Up Weld on RHR Heat Exchanger 1B Bypass Valve
- WO 195958, MSIV Disc Piston Refurbishment
- WO 525425, Repair of Through Wall Leak on 1-SW-103-24-157
- WO 031176, Replacement of SLC Flexhoses

IWE Containment Vessel Inspection

The inspectors reviewed the licensee's ISI procedures for the containment (drywell and torus) inspection to determine if the procedures complied with the TS, ASME Boiler and Pressure Vessel Code, Article IWE of Section XI, 1992 Edition and 1992 Addenda, and 10 CFR 50.55a. The inspectors also reviewed records documenting visual inspections performed on the containment during the current outage and during the March, 2002 refueling outage to determine if the licensee program for inspection of the containment was being performed in accordance with the requirements specified in Article IWE of Section XI, 1992 Edition and 1992 Addenda, and 10 CFR 50.55a. The inspectors examined the interior surfaces of the Unit 1 containment liner and the moisture barrier at the intersection of the liner and interior concrete floor area. Documents reviewed are listed in Attachment 1.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

Quarterly Review

The inspectors observed licensed operator performance and reviewed the associated training documents during simulator training sessions for Cycle 2004-01. The simulator observation and review included an evaluation of alarm response 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 simulator training observed tested the operators' ability to perform a reactor startup. The inspectors reviewed the operators activities to verify consistent clarity and formality of communication, conservative decision-making by the crew, and appropriate use of procedures. Group dynamics and supervisory oversight, including the ability to properly identify and implement appropriate TS actions, regulatory reports, reactivity management, and the use of peer checking, were observed. The inspectors assessed whether appropriate feedback was planned to be provided to the licensed operators. The inspectors reviewed documents listed in Attachment 1.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

For the equipment issues described in ARs listed below, the inspectors reviewed the licensee's implementation of the Maintenance Rule (10 CFR 50.65) with respect to the characterization of failures, the appropriateness of the associated Maintenance Rule a(1) or a(2) classification, and the appropriateness of the associated a(1) goals and corrective actions. The inspectors 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 105773, Unit 2 B core spray pump inoperability due to room fan breaker failure
- AR 114576, EDG #3 turbocharger/intercooler jacket water leak
- AR 114950, emergency core cooling system cabinet 2-XU-63 inadvertently de-energized during maintenance

To assess the licensee's identification and resolution of problems, the inspectors reviewed AR 116354 which documented that the time to repair a Unit 2 core spray room fan breaker was not counted toward core spray system unavailabilty.

b. Findings

No findings of significance were identified.

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 following OOS equipment or conditions, and the documents listed in Attachment 1:

- AR 116796, Unit 1 electro-hydraulic leak on main turbine control valve #3 and subsequent downpower to facilitate repairs (emergent)
- AR 116670, vital battery chargers susceptible to tripping during motor starts with the batteries disconnected from the DC switchboards (emergent)
- AR 114456, 2A-2 vital battery charger trip (emergent)

- AR 116109, EDG #1 inoperable due to cracked casing on auxiliary lubricating oil pump (emergent)
- WO 224629, Unit 2 online risk assessment due to replacement of 125VDC vital battery 1A-2 (planned)
- AR 121226, Unit 2 A recirculation pump motor-generator trip on low lubricating oil pressure resulting in single loop operation (emergent)

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 transients and abnormal conditions to verify the response was in accordance with procedures and training. Operator logs, plant computer data, and associated operator actions were reviewed. Documents reviewed are listed in Attachment 1.

- Unit 2 downpower to repair electro-hydraulic control leak on main turbine control valve on December 31, 2003
- Unit 2 entry into Abnormal Operating Procedure 0AOP29, Malfunction of Annunciators (Rev. 7), due to loss of control room annunciator Panel P603 on February 12, 2004
- Unit 2 downpower due to A recirculation pump trip on March 12, 2004

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the operability evaluations associated with the following five 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) the adequacy of any compensatory measures in place, including their intended use and control; and 4) where continued operability was considered unjustified, the impact on TS 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.

- WO 503125, EDG 3 jacket water system leak of approximately 1.5 GPM
- AR 114576, EDG 3 past operability determination
- AR 114454, through-wall leak discovered on the conventional service water discharge header in the Unit 2 reactor building

- AR 115446, RCIC oil filter high differential pressure alarm
- AR 120719, Vital Battery 1-1B terminal connections not torqued to vendor-recommended value due to out-of-tolerance maintenance and test equipment

b. Findings

No findings of significance were identified.

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 Attachment 1.

- WO 495917, calibration check of EDG 3 jacket cooling water expansion tank level switch
- WO 431339, Unit 2 high pressure coolant injection system electric governor capacitor removal
- WO 526183, repair Unit 1 A core spray pump breaker relay logic following delayed trip during 0MST-DG11R
- WO 224628, retorque Vital Battery 1B-1 terminal connections following discovery of out-of-tolerance torque wrench
- WO 289505, disassemble, inspect, and repair Unit 1 reactor-building-to-suppression-chamber vacuum breaker (1-CAC-X20A/B)

To assess the licensee's identification and resolution of problems, the inspectors reviewed AR 121303. The AR addressed an inspector-identified issue regarding additional post-maintenance testing requirements added after initial planning of work associated with torquing of vital battery terminal connections.

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities

a. Inspection Scope

The inspectors evaluated Unit 1 RFO B115R1 activities which commenced on February 28, 2004. At the completion of the inspection, fuel movement was complete and the unit was in Mode 4 (cold shutdown) and preparing for startup activities. Documents reviewed are listed in Attachment 1. The following specific areas were reviewed:

Outage Plan. The inspectors reviewed Brunswick Nuclear Plant Unit 1 Safe Shutdown Risk Assessment, for RFO B115R1. The inspectors verified 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. The inspectors' review of this report was compared to the requirements in Procedure 0AP-022, BNP Outage Risk Management. The review was also to verify that, for identified high risk significant conditions, contingency measures were identified and incorporated into the risk plan, and that defense-in-depth was maximized to the extent possible. The inspectors frequently monitored the risk condition during the outage.

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

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 and that reactor vessel, reactor cavity, and spent fuel pool level instruments were configured to provide accurate indication. 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 conducted a review of the higher outage risk conditions including the periods of EDG load testing and natural circulation flow to both loops of shutdown cooling being secured for maintenance. 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 during shutdown, outage, and refueling activities 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. Licensee performance was compared to Procedure 0AP-038, Reactivity Management Program Manual.

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

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

  - Controls over electrical power systems and components to ensure emergency power was available as specified in the outage risk report
  - Controls and monitoring of electrical power systems and components and work activities in the power transmission yard
  - Operator monitoring of electrical power systems and outages to ensure that TS requirements were met

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

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 121086, Drywell personnel airlock penetration sleeve liner below minimum wall thickness
- AR 121592, Drywell liner indications missed during initial inspection of H Downcomer
- AR 121137, Incorrect rotation of supplemental spent fuel pool cooling system cooling tower fans
- AR 119714, Potential overthrust/overtorque of residual heat removal system Suppression Pool Suction Valves (1-E11-F004B/D)
- AR 120388, Valve actuator motor removed from Valve 1-B32-F031A vice 1-B32-F031B (recirculation pump discharge valves)
- AR 120524, Foreign material found in drywell-to-torus vent lines
- AR 119921, Loss of control room reactor vessel level indication while in Mode 5 (refueling)
- AR 121061, Higher than expected heatup of spent fuel pool/reactor cavity while removing decay heat using supplemental spent fuel pool cooling system

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, Updated Final Safety Analysis Report 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 following documents as well as documents listed in Attachment 1:

- Periodic Test 0PT-12.2C, No. 3 Diesel Generator Monthly Load Test (Rev. 77)
- Maintenance Surveillance Test 0MST-DG-11R, DG-1 Loading Test (Rev. 1)
- Operating Instruction 0OI-03.6, Radioactive Waste Operator Daily Surveillance Report (Rev. 19)
- Containment isolation valve local leak rate testing, Periodic Test 0PT-20.3a.1, B21-F022A and B21-F028A (A main steam isolation valves) Leak Test (Rev. 10)

Inservice Surveillance Testing (IST)

The inspectors reviewed the performance of Maintenance Surveillance Test 0MST-CAC501R, CAC Reactor Building to Suppression Chamber Vacuum Breaker Channel Calibration (Rev. 2) 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.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control To Radiologically Significant Areas

a. Inspection Scope

Access Controls

The inspection reviewed the licensee's program activities for monitoring workers and controlling access to radiologically significant areas and tasks. The inspectors evaluated procedural guidance, directly observed implementation of administrative and established physical controls, assessed worker exposures to radiation and radioactive material, and appraised radiation worker and technician knowledge of, and proficiency in, implementing Radiation Protection (RP) program activities.

During the onsite inspection, radiological controls for maintenance activities were observed and discussed with cognizant licensee representatives. The inspectors observed work associated with new fuel receipt, shielding of a residual heat removal (RHR) heat exchanger, and movement of an empty spent fuel shipping cask into the reactor building. In addition, the inspectors independently measured radiation dose rates and evaluated the established postings. Radiological postings, labeling and access controls were directly observed by the inspectors during tours of the Unit 1 and Unit 2 reactor buildings, lower elevations of the turbine building and radioactive waste (radwaste) processing areas. Control of locked high radiation area (LHRA) keys and the physical status of LHRA doors were also independently evaluated by the inspectors.

Occupational workers' adherence to selected radiation work permits (RWPs) and health physics technician (HPT) proficiency in providing job coverage were evaluated by the inspectors through direct observations, review of selected exposure records and investigations, and interviews with cognizant licensee representatives. Occupational exposure data associated with direct radiation, potential radioactive material intakes, and discrete radioactive particle or dispersed skin contamination events were reviewed and assessed independently by the inspectors.

RP program activities were evaluated against 10 CFR Part 19.12; 10 CFR Part 20, Subparts B, C, F, G, H, and J; UFSAR details in Section No. 11, Radioactive Waste Management, and Section No. 12, Radiation Protection; TS Sections 5.4.1, Procedures, and 5.7, High Radiation Area (HRA); and approved licensee procedures. Licensee guidance documents, records, and data reviewed within this inspection area are listed in Attachment 1.

Problem Identification and Resolution

Corrective Action Program (CAP) documents associated with radiological controls, personnel monitoring, and exposure assessments were reviewed and discussed with cognizant licensee representatives. The inspectors assessed the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with Nuclear Generation Group Common Procedure CAP-NGGC-0200, Corrective Action Program, Revision (Rev.) 9. Specific condition report (CR) documents that were reviewed and evaluated in detail for these program areas are identified in Attachment 1.

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation and Protective Equipment

a. Inspection Scope

Radiation Monitors and Protective Equipment

The inspectors reviewed the operability and maintenance of selected radiation detection and respiratory protective equipment. The inspection consisted of document review, discussions with plant personnel, and observation of routine testing for the following items: area radiation monitors (ARMs), continuous air monitors (CAMs), personnel monitors, portable detection instruments, and self-contained breathing apparatus (SCBA) units.

The inspectors reviewed calibration records for selected ARMs and CAMs and HPTs regarding the results. The placement and use of CAMs inside containment during the previous Unit 1 refueling outage was evaluated and discussed with cognizant licensee representatives.

Whole body counter (WBC) calibration records and daily source check trends were reviewed by the inspectors and discussed with dosimetry personnel. Intakes of radioactive material by individuals for 2003 were also reviewed by the inspectors and discussed with cognizant dosimetry personnel.

Procedural guidance for the use and calibration of portable survey instruments was evaluated by the inspectors. The inspectors observed the daily source check of an RO-20 and a Teletector survey meter and compared the results to specified tolerances. The inspectors interviewed cognizant licensee representatives regarding the licensee's program for the use of electronic dosimeters (including use in high noise areas) and observed the functional test and calibration of an alarming dosimeter. In addition, calibration records were reviewed for a Teletector, RO-2, RO-20, and RM-14 survey instruments and a SAM No. 9 article monitor in use at the time of the inspection.

The licensee's respiratory protection program guidance and its implementation for SCBA use were evaluated by the inspectors and discussed with cognizant licensee representatives. The number of available SCBA units and their general material and operating condition were observed during tours of the control room and reactor auxiliary building. Current records associated with supplied air quality for staged SCBA equipment were evaluated by the inspectors. In addition, the inspectors interviewed control room operators to determine their level of knowledge of available SCBA equipment storage locations and availability of prescription lens inserts, if required. Procedures and training for performing a SCBA bottle change - out were also reviewed by the inspectors.

Program guidance, performance activities, and equipment material condition were reviewed against details documented in 10 CFR Parts 20 and 50; UFSAR Section 12.1.4, Area Monitoring; applicable sections of NUREG-0737, Clarification of Three Mile Island (TMI) Action Plan Requirements, November 1980; Regulatory Guide (RG) 1.97, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident, Rev. 3; RG 8.15, Acceptable Programs for Respiratory Protection, Rev. 1; and applicable licensee procedures. All documents reviewed are listed in Attachment 1.

Problem Identification and Resolution

Selected CRs associated with area radiation monitoring equipment, portable radiation detection instrumentation, and respiratory protective program activities were reviewed and assessed. The inspectors also assessed the licensee's ability to characterize, prioritize, and resolve the identified issues in accordance with Procedure CAP-NGGC-0200, Corrective Action Program, Rev. 9.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety

2PS1 Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems

a. Inspection Scope

Effluent Processing Equipment

The inspectors reviewed and evaluated the operability, availability and reliability of selected radioactive effluent process sampling and detection equipment used for routine and accident monitoring activities. Inspection activities consisted of direct observation of installed equipment configuration and operation and review of calibration and performance data for the liquid and gaseous effluent process systems.

The inspectors directly observed Unit 1 and Unit 2 equipment material condition and assessed selected gaseous and liquid effluent processing and monitoring components against design configuration and operating specifications. During walk-downs, accessible sections of the liquid waste system, including waste monitor tanks, system piping, and radioactive waste liquid effluent monitor equipment were assessed for material condition and conformance with current system design diagrams. Inspected components of the main gaseous effluent process and release system included the parts of the plant vent radiation monitoring system along with associated sample lines. The inspectors interviewed cognizant chemistry supervision and system engineering personnel regarding liquid and gaseous radwaste system configurations, system reliability, system modifications, and effluent monitor operation.

The inspectors reviewed applicable sections of effluent monitor calibration procedures and evaluated results of calibration and/or functional tests for the radwaste liquid effluent monitor and its associated flow monitor, the main stack radiation monitor, the reactor building roof vent samplers, the turbine building vent monitor, the process gas (noble gas) effluent monitor and high efficiency particulate (HEPA) filter systems. Reviewed data included isotopic calibration records, source check results, flowmeter calibration records, and HEPA surveillance records. The inspectors also reviewed OOS data and contingency sampling records for selected effluent monitors for the period of November 2002 - October 2003.

Installed configuration, material condition, operability, and reliability for selected effluent sampling and monitoring equipment were reviewed by the inspectors against details documented in the following: 10 CFR Part 20; RG 1.33, Quality Assurance Program Requirements (Operation), February 1978; RG 1.21, Measuring, Evaluating and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water Cooled Nuclear Power Plant, June 1974; ANSI-N13.1-1969, Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities; ANSI-N13.10-1974, ANS Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents; TS Section 5.6.3; the Offsite Dose Calculation Manual (ODCM), Rev. 26; and UFSAR Chapters 9 and 11. Procedures and records reviewed during the inspection are listed in Attachment 1.

Effluent Release Processing and Quality Control Activities

The inspectors evaluated the licensee's performance in conducting effluent release processing and quality control (QC) activities including implementation of program guidance and chemistry staff proficiency. The inspection consisted of interviews of cognizant chemistry staff and supervision and observation of a chemistry staff member demonstrating the normal processing of routine release logging/permitting. The review included release documentation and applicable licensee procedures.

The effluent release program was evaluated against the following guidance: 10 CFR Part 20 and Appendix I to 10 CFR Part 50; ODCM; RG 1.21, RG 4.15, Quality Assurance for Radiological Monitoring Programs (Normal Operation) - Effluent Streams and the Environment, December 1977; and RG 1.109, Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50 Appendix I, October 1977. Procedures and records reviewed during the inspection are listed in Attachment 1.

Problem Identification and Resolution

Two licensee CRs and one audit associated with effluent release activities were reviewed and assessed. The inspectors evaluated the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with Nuclear Generation Group

Common Corrective Action Procedure CAP-NGGC-0200, Corrective Action Program, Rev. 9. Documents reviewed are listed in Attachment 1.

b. Findings

No findings of significance were identified.

2PS3 Radiological Environmental Monitoring Program (REMP) and Radioactive Material Control Program

a. Inspection Scope

REMP Implementation

The inspectors reviewed and discussed with cognizant licensee representatives the results published in the Brunswick Annual Radiological Environmental Operating report for CY 2002. The inspectors observed the collection and preparation of weekly particulate and radioiodine samples by licensee personnel and assessed material condition of three air sampling stations (Station Nos. 201, 202, 203), one river water sampling station (Station No. 400), one vegetation sampling station (Station No. 800), and six thermoluminescent dosimeters (TLDs) (Station Nos. 5, 8, 14, 24, 25, 81) to evaluate procedural compliance. The inspectors assessed the calibration status of each air sampling pump. The inspectors also evaluated the placement of collection station locations against the sectors specified in the ODCM using the NRC global positioning system. The inspectors observed and discussed with cognizant licensee representatives the procedures, methods, and equipment used to perform vegetation, sediment, and fish/invertebrate sampling. The inspectors reviewed and discussed with cognizant licensee representatives the procedures used to calibrate and determine the LLD for environmental sample gamma spectroscopy analysis.

REMP guidance, implementation, and results were reviewed against ODCM guidance and applicable procedures listed in Attachment 1.

Meteorological Monitoring Program

The inspectors reviewed the operability of the meteorological monitoring equipment and operator access to meteorological data. Current meteorological monitoring equipment performance and calibration were reviewed with the system engineer. Licensee technicians primarily responsible for equipment maintenance and surveillance were interviewed by the inspectors concerning equipment performance, reliability, and routine inspections. Inspectors compared the meteorological data available in the control room against the meteorological data recorder at the tower location.

Meteorological instrument operation, calibration, and maintenance were reviewed against UFSAR, Chapter 2; NRC Safety Guide 23, Onsite Meteorological Programs-1972; and applicable licensee procedures. Documents reviewed are listed in Attachment 1.

Unrestricted Release of Materials from the Radiologically Controlled Area (RCA)

The inspectors reviewed calibration records for two personnel contamination monitors and one material release monitor. The inspectors also observed source checking of three material survey monitors. Types of sources used for checks and minimum detectable activities were discussed with an instrument technician.

The inspectors verified that radiation detection sensitivities were consistent with NRC guidance in IE Circular 81-07 and IE Information Notice 85-92. Documents reviewed are listed in Attachment 1.

Problem Identification and Resolution

Licensee CAP issues associated with environmental monitoring, meteorological monitoring, and release of materials were reviewed and discussed with cognizant licensee representatives. The inspectors assessed the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with Procedure CAP-NGGC-0200, Corrective Action Program, Revision 9. Specific documents that were reviewed and evaluated in detail for these program areas are identified in Attachment 1.

b. Findings

No findings of significance were identified.

4OA1 Performance Indicator Verification

a. Inspection Scope

The inspectors sampled licensee submittals for the Unit 1 and 2 performance indicators (PIs) listed below. 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

- Unplanned Scrams per 7000 Critical Hours
- Scrams with Loss of Normal Heat Removal
- Emergency AC Power-Safety System Unavailability

A sample of plant records and data was reviewed for the period April 2003 through December 2003, and compared to the reported data to verify the accuracy of the PIs. This included operating logs and licensee event reports. The licensee's corrective action program records were also reviewed to determine if any problems with the collection of PI data had occurred.

Occupational Radiation Safety Cornerstone

- Occupational Exposure Control Effectiveness

The inspectors reviewed CR records for HRAs, very high radiation areas (VHRA), and unplanned exposure occurrences for the period of December 2002 through December 2003, to verify that TS and 10 CFR 20 non-conformances were properly classified as PIs. The inspectors also reviewed radiological controlled area exit transactions with exposures greater than 100 millirem (mrem) and investigated a sample of those transactions to determine whether they were within the respective RWP and to verify that those greater than 100 mrem unplanned exposure were entered in the corrective action program and listed as a PI. Reviewed documents are listed in Attachment 1.

Public Radiation Safety Cornerstone

- RETS/ODCM Radiological Effluent Occurrences

The inspectors reviewed radiological control effluent release occurrences for the from January 2002 through December 2003. For the period, the inspectors reviewed data reported to the NRC, procedural guidance for reporting PI information, and two CRs listed in Attachment 1. In addition, the inspectors reviewed monthly PI reports from January 2002 through December 2003.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution

a. Inspection Scope

While performing a plant status review on January 5, 2004, the inspectors noted that during the monthly performance of an EDG 3 surveillance, an operability determination was made relative to an identified jacket cooling water (JWC) system leak. Subsequent to the EDG run, the EDG was considered inoperable. The licensee determined that a leak had also existed during the same surveillance in December 2003, with corrective action repair being performed shortly after the leak was identified. However, after the January 2004 leak was noted, further review identified that the corrective action repair in December 2003 did not fix the leak, leading directly to the problem in January 2004. Because of the potential significance of the corrective action problem from the December 2003 repair, the inspectors selected AR 114576, DG3 Past Operability Determination, for detailed review. The review was performed to verify that conditions adverse to quality were addressed in a manner that was commensurate with the safety significance of the issue. Additional documents reviewed are listed in Attachment 1.

b. Findings

Introduction. A preliminary White finding with two apparent violations was identified for Unit 2, and one Green NCV was identified for Unit 1, associated with the failure to take adequate corrective actions on the EDG 3 JWC system leak.

Description. During the monthly (surveillance) testing of EDG 3 on December 7, 2003, a 0.5 gallons per minute (gpm) leak on the JWC system was noted on engine start-up. The leak later decreased to approximately 40 drops per minute following system heat-up to normal operating temperature. At that time, Operations, with Engineering's recommendation, determined the EDG was operable based on the low operating leak rate and the fact that makeup was available using water supplied (hard piped) from the demineralized water system. On December 8, 2003, a maintenance mechanic tightened the pipe coupling to the vendor recommended torque value using the minor maintenance work process. However, a post-maintenance functional test to verify that the leak had stopped, was not performed.

On January 4, 2004, during monthly testing of EDG 3, a 1 gpm leak was observed on the JWC system from the same coupling. In consultation with the system engineer, Operations initially determined EDG 3 was operable for the same reasons used in the December 7, 2003 leakage event. That is, the decision was based on the ability to make up to the JWC system expansion tank from the site's demineralized water system. After a management review with respect to the impact of the leak size on the operability decision, Operations reevaluated the ability (and timeliness) to make up the water to the EDG. Subsequently, Operations determined the leak size was too large for timely makeup and declared the EDG inoperable.

Repairs were initiated and, following the repair activities, EDG 3 was restored to an operable status on January 7, 2004. The inspectors challenged the appropriateness of considering the EDG operable on December 7, 2003, based on both the leak size and the availability of the demineralized water system, since the system would not be available during an event for which the EDG would be called upon to function. The demineralized water system is not safety-related (i.e., not seismically qualified) or backed up by emergency power. Proper operation of this system relies on pumps that lose power upon a loss of off-site power. The unavailability of the demineralized water system was not considered during either operability evaluation. Additionally, the inspector noted that the JWC expansion tank low level alarm was out-of-service during the time period in question which could have complicated operator diagnosis/response to the leak.

The licensee entered the issue into the corrective action program as ARs 114576 and 114573. The cause of the condition was attributed to missing pipe supports which resulted in an inadequate pipe coupling alignment. The missing supports factor was considered an historical condition and, as such, the licensee did not pursue the cause of that condition, or the period of time the supports were missing. The licensee determined that an adequate functional test following the maintenance performed on December 8, 2003, would have been a leak check of the system at normal operating pressure. However, this test was not performed. The licensee concluded that the maintenance performed on December 8, 2003, had most likely aggravated the coupling leak. Accordingly, the licensee determined that the EDG had been inoperable since December 8, 2003, a significantly longer time than the seven days of allowed outage time in TS 3.8.1.

Timely and appropriate corrective action, commensurate with the potential safety significance, was not taken for leakage identified from EDG 3 jacket water cooling system on December 7, 2003. Missing pipe supports on the jacket water cooling system resulted in misalignment of a system pipe coupling which caused system leakage. Maintenance practices and controls for repairs performed on December 8, 2003 to correct the deficiency caused the leakage to increase which was not detected due to the failure to perform appropriate post-maintenance testing. Operability assessments of system leakage, on both December 7, 2003, and on January 4, 2004, did not consider the potential impact of a loss of off-site power on the ability of the demineralized water system to make up to the EDG to compensate for the degraded condition (inspector identified). This performance deficiency resulted in Units 1 and 2 not meeting TS Limiting Condition of Operation 3.8.1, AC Sources-Operating.

Analysis. The finding affects the Mitigating System Cornerstone for Units 1 and 2. The finding is more than minor because it is associated with the availability and reliability of EDG 3 to mitigate events such as a loss of offsite power. Because this finding represented an actual loss of safety function of EDG 3 for greater than the TS LCO 3.8.1 allowed outage time for one EDG (i.e., seven days), an SDP Phase II analysis was performed. The dominant core damage sequence was Loss of Offsite Power (LOOP) and LOOP with Loss of One AC Division. The results of the Phase II analysis required a Phase III evaluation.

A phase 3 analysis was performed using the Brunswick SPAR model. Assumptions critical to this evaluation involved the ability of operators to maintain jacket water level by making up to the leaking system. This analysis assumed that for a period of 26 days, EDG 3 could have been recovered by an operator refilling the jacket water cooling system. For a 37.5 hour period when the demineralized water tank level was too low to support gravity feed the EDG was unavailable with no recovery. The EDG was also unavailable due to repair of the performance deficiency for a period of 28 hours. The probability of the operator failing to recover the EDG was estimated at 0.1 using SPAR human reliability analysis (HRA) methods. The SPAR analysis identified the loss of offsite power with failures to supply power to the emergency busses as dominant risk sequences. Because some of the sequences also involve failures to depressurize the reactor, the analysis of large early release frequency (LERF) was also an important consideration. However, based on the unique containment structure of Brunswick, LERF was determined to not be a significant factor for the loss of power sequences. External events were considered but also determined not to be significant contributors to this evaluation because their initiating event frequencies were small compared to the loss of offsite power frequency. The SPAR model result for the change in core damage frequency (CDF) was 1.1e-6.

The finding represented low to moderate safety significance and was determined to be preliminarily White on Unit 2 based on ΔCDF and ΔLERF. The finding was determined to be Green on Unit 1 based on ΔCDF and ΔLERF. The difference between the two units is primarily because EDG 3 is a primary EDG for Unit 2 and a backup EDG for Unit 1. See Attachment 2, Significance Determination Process Phase III Summary, for further details.

Enforcement. 10 CFR 50, Appendix B, Criteria XVI, requires in part that measures shall be established to assure that conditions adverse to quality, such as failures and malfunctions, are promptly identified and corrected.

Technical Specification (TS) Limiting Condition of Operation 3.8.1, AC Sources-Operating, requires four EDGs to be operable when in Mode 1 with operation with three EDGs allowed for a period of seven days.

Contrary to the above, two structural supports located on the jacket water turbo charger supply line for EDG 3 were removed prior to approximately January 1, 2001 and were not reinstalled until January 7, 2004. These supports are documented in Brunswick as-built Drawing No. FP-20323, Rev K, dated 6/16/03. This failure to reinstall the missing supports, a condition adverse to quality, resulted in an inadequate pipe coupling, which contributed to a jacket water system leak on December 7, 2003. A jacket water system leak on EDG 3, a condition adverse to quality, identified on December 7, 2003, was not promptly corrected. Maintenance performed on December 8, 2003, did not correct the deficiency, so that the leak was still present and larger on January 4, 2004. As a result, only three EDGs were operable from December 8, 2003 until January 7, 2004, while Units 1 and 2 were in Mode 1 and the licensee did not satisfy the requirements of TS LCO 3.8.1, AC Sources-Operating.

This finding does not present a current safety concern because the jacket water cooling system structural supports were reinstalled, the system leak was appropriately repaired, and EDG 3 was restored to an operable status on January 7, 2004. Planned corrective actions include actions to reinforce minor maintenance and functional verification requirements and communicating expectations to maintenance personnel. This issue has been entered into the corrective action program as ARs 114573 and 114576. The licensee documented the failure to meet TS LCO 3.8.1 in Licensee Event Report 05000325,324/2004-001-00, Emergency Diesel Generator No. 3 Condition Prohibited by the Technical Specifications, dated March 4, 2004. These NRC-identified apparent violations (AVs) of regulatory requirements are identified for Unit 2 as AV 05000324/2004002-01, Inadequate Corrective Actions for EDG Jacket Water Cooling Leak, and AV 05000324/2004002-02, Failure to Meet TS LCO 3.8.1.

Because the failure to promptly correct the jacket water cooling system leak is of very low safety significance for Unit 1 and has been entered into the corrective action program, this finding is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000325/2004002-03, Inadequate Corrective Actions for EDG Jacket Water Cooling Leak Results in Failure to Meet TS LCO 3.8.1.

4OA3 Event Follow-up

a. Inspection Scope

The inspectors reviewed the licensee's 10 CFR 50.73 telephone notification as a result of receiving an invalid primary containment isolation system Group 6 actuation on January 11, 2004. The isolation was caused by the reactor building exhaust radiation monitor (1-D12-RM-K609B) input signal spiking. All required actuations operated properly (i.e. secondary containment isolation, standby gas treatment system automatic start, etc.). The cause of the invalid actuation was determined to be a failed detector due to end-of-life. The inspectors reviewed the 10 CFR 50.73 notification to assess appropriate reporting within established criteria.

b. Findings

No findings of significance were identified.

4OA5 Other Activities

.1 Spent Fuel Material Control and Accounting

a. Inspection Scope

The inspectors completed Phase I and Phase II of Temporary Instruction 2515/154, "Spent Fuel Material Control and Accounting at Nuclear Power Plants".

b. Findings

No findings of significance were identified.

.2 Review of 2003 World Association of Nuclear Operators (WANO) Final Report

The inspectors reviewed the 2003 WANO Peer Review, Final Report, for the Brunswick Steam Electric Plant, dated March 15, 2004. The review determined that the results of the WANO report were generally consistent with the results of similar evaluations conducted by the NRC. The inspectors concluded that no additional Regional follow-up concerning this report was warranted.

.3 (Closed) Unresolved Item (URI) 05000324/2003006-02, Unit 2 Reactor Feed Pump Speed Control Modification

This item was opened for a finding involving an inadequate design modification of the Unit 2 reactor feed pump speed control system. The issue was unresolved pending significance determination. The NRC preliminarily determined the issue to be of low to moderate (White) risk significance as discussed in an NRC letter dated January 30, 2004. The licensee provided additional risk information in a Regulatory Conference on March 17, 2004. Following review and consideration of the information provided, the issue was determined to be of very low risk significance (Green). The final risk significance determination was documented in NRC Inspection Report 50-325/2004-007 dated April 2, 2004. This URI is closed.

.4 (Closed) URI 50000325, 324/2002003-02: Failure of SCBA Training Program to Include Demonstration of Proficiency in SCBA Cylinder Change-out

In September 2002, the inspectors determined that Lesson Plan GN6C10G for non-fire brigade workers did not require instruction for SCBA qualified personnel on how to replace air supply bottles nor require them to demonstrate their ability to do so.

The inspectors reviewed the licensee's followup to this URI by reviewing the actions taken by the licensee in response to AR 00067106, Training for SCBA Bottle Changeout, dated July 25, 2002. Based on a review of the actions taken by the licensee and discussions with cognizant licensee representatives, the inspectors determined that the licensee had revised applicable procedures to incorporate and formalize the process for health physics technician support for SCBA cylinder bottle change outs. In addition, the training was revised to include practicals for individuals to specifically demonstrate the proper bottle changeout procedure with the instructor present. The inspectors interviewed several individuals from the Maintenance and Radiation Protection Departments about SCBA bottle changeout and each individual was able to satisfactorily describe the process. In reviewing the historical significance of this issue, the inspectors noted that health physics staff had been trained on bottle changeout (as part of their fire brigade responsibility). They had been previously utilized as standby rescue personnel to aid individuals who would have required bottle changeout or who were experiencing trouble extricating themselves. However, this process had not been proceduralized. Also, an assigned loss prevention unit auxiliary operator for each shift had also been previous trained in the changeout process. As such, this finding was determined to be a violation of 10 CFR 20.1703(c), but of minor significance and, thus, is not subject to enforcement action in accordance with section IV of the NRC Enforcement policy. Reviewed documents are listed in Attachment 1. This URI is closed.

4OA6 Meetings, Including Exit

On March 25, 2004, the resident inspectors presented the inspection results to Mr. C. J. Gannon and other members of his staff. Additional discussions further clarifying the findings of this report were held on April 2 and 19, 2004. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

ATTACHMENTS
1. SUPPLEMENTAL INFORMATION
2. SIGNIFICANCE DETERMINATION PROCESS PHASE III SUMMARY

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

G. Atkinson, Supervisor - Emergency Preparedness
J. Bates, IWE Containment Engineer
L. Beller, Supervisor - Licensing/Regulatory Programs
E. Black, NDE Level III ISI Specialist
H. Bordeaux, QC Manager
A. Brittain, Manager - Security
E. Conway, Senior Nuclear Security Specialist
D. DiCello, Manager - Nuclear Assessment
C. Elberfeld, Lead Engineer - Technical Support
J. Frisco, Superintendent-Mechanical Maintenance
C. Gannon, Site Vice President
J. Gawron, Training Manager
M. Grantham, Superintendent-Design Engineering
S. Hamilton, Envirnonmental and Radiation Control Manager
D. Hinds, Plant General Manager
R. Kitchen, Engineering Manager
D. Makosky, Lead Nuclear Security Specialist
J. McIntire, Equipment Performance Superintendent
W. Noll, Director - Director of Site Operations
E. O'Neil, Manager - Site Support Services
A. Pope, Superintendent-Systems Engineering
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
R. Bernhard, Senior Reactor Analyst, Division of Reactor Safety, Region II

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000324/2004002-01	AV	Inadequate Corrective Actions for EDG Jacket Water Cooling Leak (Section 4OA2)
05000324/2004002-02	AV	Failure to Meet TS LCO 3.8.1 (Section 4OA2)

Opened and Closed

05000325/2004002-03	NCV	Inadequate Corrective Actions for EDG Jacket Water Cooling Leak Results in Failure to Meet TS LCO 3.8.1 (Section 4OA2)

Closed

05000324/2003006-02	URI	Unit 2 Reactor Feed Pump Speed Control Modification (Section 4OA5.3)
50000325, 324/2002003-02	URI	Failure of SCBA Training Program to Include Demonstration of Proficiency in SCBA Cylinder Change-out (Section 4OA5.4)

Discussed

NONE

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Plant Operating Manual (POM), Volume VII, Revision 18, Operating Instruction 0OI-01.03, Non-Routine Activities

Section 1R04: Equipment Alignment

POM, Vol. III, Rev. 80, 1OP-17, Residual Heat Removal System Operating Procedure
POM, Vol. III, Rev. 132, 2OP-17, Residual Heat Removal System Operating Procedure

Section 1R08: Inservice Inspection Activities

Procedures:

Engineering Procedure, 0ENP-16.15, Administrative Procedure for Component Support and Snubber Program, Rev. 13
Periodic Test Procedure, 0PT-20.5.1, Primary Containment Inspection, Rev. 13
Operation Periodic Test, 0PT-19.6.1, Snubber Functional Testing, Rev. 33
NDEP-0613, VT-3 Visual Examination of Nuclear Power Plant Components, Rev. 18
NDEP-0425, Ultrasonic Examination of Austenitic Pipe Welds (PDI), Rev. 5
NDEP-0437, Manual Ultrasonic Examination Procedure for Ferritic Pipe Welds (PDI), Rev. 0
NDEP-0452/PDI-UT-6, Manual Ultrasonic Examination Procedure for Reactor Pressure Vessel Welds (PDI), Rev. 0
NDEP-0456, Manual Ultrasonic Examination of Nozzle Inner Corner Radius Area Per ASME XI (Appendix VIII), Rev. 0
NDEP-0457, Ultrasonic Examination of Dissimilar Metal Welds (PDI), Rev. 0

Other Documents:

OBNP-TR-002, Containment Inspection Program, Rev. 7
AR 77992, Lack of Trending Program for Snubbers (self-assessment AR)
AR 77988, Training of Maintenance and Engineering Personnel in Functional Testing of Snubbers (self-assessment AR)
AR 77985, Snubber Drag Testing (self-assessment AR)
AR 80162, Uncertainty Regarding Code Compliance for Dissimilar Metal Welds
AR 87205, No Fluid in Snubber 2-B21-1SS227 on Feedwater System
AR 87706, Low Fluid in Snubber 2-B32-SSA2
AR 106596, Assess Inconel Weld Inspections for Possible Augmentation
AR 108003, The ISI program Plan Lists Two Different Calibration Standards
AR 120537, Low Fluid Level in Snubber 1-B21-3SS13
AR 120855, Water and Rust Found in Penetration x-13B
AR 121086, 1-X-2 Personnel Airlock Penetration Sleeve Below Min Wall
AR 120997, Bolting Failure on Limitorque Actuator
AR 121142, Personnel Airlock Thickness not per Drawing
Technical Requirements Manual, TRMS 3.21 and B 3.21, Snubbers

Section 1R11: Licensed Operator Requalification

POM, Vol. IV, Rev. 162, 0GP-01, Prestartup Checklist
POM, Vol. IV, Rev. 75, 0GP-02, Approach to Criticality and Pressurization of the Reactor
POM, Vol. XVI, Rev. 47, 2APP-A-05, Annunciator Procedure for Panel

Section 1R12: Maintenance Effectiveness

Nuclear Condition Report (NCR), AR 116354
WO 488458, leak on jacket cooling water to EDG 3

Section 1R13: Maintenance Risk Assessments and Emergent Work Evaluation

AR 20277, Basis for crediting DC chargers in PSA

Section1R14: Operator Performance During Non-Routine Plant Evolutions and Events

POM, Vol. XIII, Rev. 48, Plant Emergency Procedure 0PEP-02.1, Initial Emergency Actions
POM, Vol., VII, Rev. 55, Operating Instruction 0-OI-01.08, Control of Equipment and System Status

Section 1R19: Post Maintenance Testing

POM, Vol. X, Rev. 112, 0PT-09.2, HPCI System Operability Test

Section 1R20: Refueling and Outage Activities

POM, Vol. I, Rev. 16, Admin. Proc. 0AP-022, BNP Outage Risk Management
POM, Vol. III, Rev. 79, 1OP-17, Residual heat Removal System Operating Procedure
POM, Vol. IV, Rev. 107, 0GP-05, Unit Shutdown

POM, Vol. III, Rev. 46,1OP-13, Fuel Pool Cooling and Cleanup System Operating Procedure
POM, Vol III, Rev. 16, 0OP13.1, Supplemental Spent Fuel Pool Cooling System Operating Procedure
POM, Vol. IX, Rev. 77, Fuel Handling Procedure 0FH-11, Refueling
POM, Vol. IX, Rev. 53, Fuel Handling Procedure 0FH-11A, Refueling Platform Operations

Section 1R22: Surveillance Testing

POM, Vol. X, Re. 76, 0PT-12.2C, No. 3 Diesel Generator Monthly Load Test
POM, Vol. X, Rev. 1, 0MST-DG11R, DG-1 Loading Test
POM, Vol. XX, Rev. 18, 0ENP-16.4, Use of Leak Test Equipment
POM, Vol. X, Rev. 56, 0PT-20.3, Local Leak Rate Testing
AR 120803, Weight test failure of 1-CAC-X20B

Section 2OS1: Access Control To Radiologically Significant Areas

Procedures, Guidance Documents, and Manuals

Environmental and Radiation Control Procedures (E&RC):

0E&RC-0040, Administrative Controls For High Radiation Areas, Locked High Radiation Areas, And Very High Radiation Areas, Revision (Rev.) 23
0E&RC-0100, Radiation Surveys Methods, Rev. 32
0E&RC-0120, Routine/ Special Airborne Radioactivity Survey, Rev. 19
0E&RC-0175, Radiological Controls For Diving Operations, Rev. 2
0E&RC-0220, Respiratory Protection Program, Rev. 42
0E&RC-0230, Issue And Use of Radiation Work Permit, Rev. 43

Nuclear Generation Group Common Dosimetry Procedures (DOS-NGGC)

DOS-NGGC-0002, Dosimetry Issuance, Rev.19
DOS-NGGC-0003, Xe-133 Skin Dose Calculation, Rev. 3
DOS-NGGC-0004, Administrative Dose Limits, Rev. 7
DOS-NGGC-0005, Skin Dose From Contamination, Rev. 6
DOS-NGGC-0007, Internal Dose Calculations, Rev. 8

Administrative Instructions (AI)

0AI-112, Control of Materials in Spent Fuel Pool, Rev. 15
0AI-122, Pre-job Briefings & Post-job Critiques, Rev. 8
0AI-131, Conduct of Diving Operations, Rev. 6

Nuclear Generation Group Common Health Physics Standard Procedures (HPS-NGGC)

HPS-NGGC-0003, Radiological Posting, Labeling And Surveys, Rev. 8
HPS-NGGC-0008, Performing Work in Radiation Control Areas, Rev. 2
HPS-NGGC-0014, Radiation Work Permits, Rev. 1
HPS-NGGC-0016, Access Control, Rev. 1
HPS-NGGC-0017, Total Exposure Source Management, Rev. 2

Nuclear Generation Group Common Maintenance Standard Procedures (HPS-NGGC)

MNT-NGGC-0004, Scaffolding Control, Rev. 3

Radiation Work Permits (RWP)

RWP Number (No.) 1794, Task 2 Management Tours/ Inspections/ Visitors, Rev. 1
RWP No. 1803, Task 5 Pre-Outage & Post- Outage Support Work (Scaffolding), Rev. 0
RWP No. 1825, Task 9 U/1 New Fuel Receipt/ Inspection/ Support Activities (Health Physics), Rev. 0
RWP No. 1825, Task 8 U/1 New Fuel Receipt/ Inspection/ Support Activities (New Fuel Inspections), Rev. 0
RWP No. 0967, Task 2, DW-Temperature Improvement Project (B115R1) (APLAN # 2725) (Drywell-Install N2 Window Package), Rev. 0
RWP No. 0963, Task 1, Drywell CRD Replacement/ Setup, Rev. 0
RWP No. 0942, Task 3, RPV Dis/Reassembly/ Refueling/ Sipping (B115R1) (APLAN #2702), Rev. 0
RWP No. 0929, Task 1, DW- SRVS / Includes Support Work (B115R1) (APLAN #2708), Rev. 0
RWP No. 0921, Task 3, DW/Cavity- Shielding (B115R1)(APLAN #2700) (DW-Permanent Shielding Installation), Rev. 0

Corrective Action Program (CAP) Documents

Action Request (AR) No. 87227, Adverse Condition Investigation Form, LHRA Barrier Not Controlled by Radiation Control, 3/11/2003
AR No. 92157, Adverse Condition Investigation Form, Observations Where the Control of Radiological Boundaries Was Not Maintained Appropriately, 5/1/2003
AR No. 92157, Adverse Condition Investigation Form, Personnel Observed Reaching Across Boundaries Without Approval from Radiation Protection, 5/1/20032

Section 2OS3: Radiation Monitoring Instrumentation and Protective Equipment

Procedures, Manuals, Guidance Documents, and Lesson Plans

Nuclear Generation Group Common Dosemetry Procedures:

DOS-NGGC-0016, MGP Electronic Personal Dosimeter (EPD) Configuration Control, Rev. 10
DOS-NGGC-0020, Whole Body Counter (WBC) System Calibration, Rev. 6
DOS-NGGC-0021, Whole Body Counter (WBC) System Operation, Rev. 11

Nuclear Generation Group Common Health Physics Standard Procedures:

HPS-NGGC-0003, Radiological Posting, Labeling, and Surveys, Rev. 8
HPS-NGGC-0005, Calibration of Portable Radiation/Contamination Survey Instruments, Rev. 4
HPS-NGGC-006, Quantitative Fit Testing, Rev. 3
HPS-NGGC-0009, Operation of Radiation/Contamination Survey Instruments, Rev. 1

Environmental and Control Procedures:

0E&RC-0136, Set Up and Use of Air Line Respiratory Protection Devices, Rev. 9
0E&RC-0220, Respiratory Protection Program, Rev. 42
0E&RC-0292, SCBA Use and Maintenance, Rev. 1
0E&RC-0310, Calibration of NMC Continuous Air Monitors (CAMS), Rev. 16
0E&RC-0344, Calibration and Use of APTEC Personnel Monitors, Rev. 6
0E&RC-339, Calibration of the SPM-904/904C Portal Monitors, Rev. 8
0E&RC-0500, Inventory Control and Leak Testing of Radioactive Sources, Rev. 23

Miscellaneous Procedures:

0 Fire Protection Procedure (FPP)-038, Operation of the SCBA Refill System, Rev. 3
0 Process Instrument Calibration (PIC)-RE004, GE Area Radiation Monitor 194X927G11, G12, G13, and G17 Sensor and Converter Calibration, Rev. 14

Progress Energy Nuclear Generation, Radiation Protection Training, Site Specific General, Student Handout, Respiratory Protection Training, Rev. 17

RWPs

RWP No. 00000579, Task 01, Drywell-CRD Replacement/Setup (B216R1) (APLAN #2651), Rev. 0
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RWP No. 00001814, Task 02, U/1 Turbine Building Hotside Work, Rev. 0

Records and Data

Certificate for Valley Safety Supply Company for Air-Pak II/IIA SCBA Maintenance and Overhaul, 07/01/03
DOS-NGGC-0021, Rev. 11, Quality Control Check Record for Chair, 8/12/03
DOS-NGGC-0021, Rev. 11, Quality Control Check Record for Stand-Up, 1/27/04
EVC-NGGC-0026, Rev. 1, Low-level Radioactive Waste Analysis Data Sheet, 4/8/03
Form HPS-NGGC-0005-5-3, Eberline RM-14 Calibration Record, 3/20/03 for Model No. RM-14, Serial Number (S/N) 5808
Form HPS-NGGC-0005-6-3, Eberline RO-2, RO-2A, & RO-20 Calibration Record, 08/21/03 for RO-2 S/N 4978
Form HPS-NGGC-0005-6-3, Eberline RO-2, RO-2A, & RO-20 Calibration Record, 10/20/03 for RO-20 S/N 1035
Form HPS-NGGC-0005-8-3, Eberline 6112B Calibration Record, 10/20/03 for Model No. 6112B, S/N 37376
Health Physics Job Standard-5.4.2, Pre-Job Briefing Checklist for RWP# 1814-02, Task 02, Location 1TB 20' 45', 1/29/04
HPS-NGGC-0009, Rev. 1, Instrument Source Check Failure Investigation Form for Meter Type 6112B, S/N 22672, 1/13/04

0E&RC-0115, Rev. 9, SAM Calibration Record for Model SAM, S/N 3, 2/21/03
0E&RC-0115, Rev. 10, SAM Calibration Record for Model SAM, S/N 192, 12/28/03
0E&RC-0214, Rev. 4, Conveyor Monitor Calibration, 7/28/03
0E&RC-0217, Rev. 14, Calibration Data Sheet for SAM9A Module and Vault Monitor, 1/23/03 and 3/13/03
0E&RC-0292, Rev. 0, Scott Air-Pak 4.5 Inspection Record, Undated
0E&RC-0310, Rev. 16, CAM Calibration Data Sheet for CAM No. 14, 10/2/03
0E&RC-0310, Rev. 16, CAM Calibration Data Sheet for CAM No. 15, 9/9/03
0E&RC-0310, Rev. 16, CAM Calibration Data Sheet for CAM No. 17, 10/08/03
0E&RC-0310, Rev. 16, CAM Calibration Data Sheet for CAM No. 18, 7/3/03
0E&RC-0343, Rev. 9, CM11 (DP11) Calibration Form for S/N 148, 8/27/03
0E&RC-0344, Rev. 6, APTEC PMW Calibration Data Sheer for PMW S/N 0012-006, 8/26/03
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0PIC-ES002, Rev. 4, 1(2)-D22-ES-K603A,B & C for Tag No. 2-D22-ES-K603A, 5/7/01
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0PIC-RE004, Rev. 12, Model 194X927G11 (Non Tech Spec) for Tag No. 2-022-RE-N001-2-19, 5/31/01
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Recertification Certificate for Valley Safety Supply Company for AIR-PAK 2.2/3.0/4.5/Fifty SCBA Maintenance and Overhaul, 06/03/03
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SCOTT PosiChek3, Visual/Functional Test Results for Model Air-Pak 4.5, ID 3860240, Reducer S/N 3860240, Regulator S/N 3880549, 8/14/02
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Analysis of Breathing Air, Firehouse SCBA Compressor, 1/30/03, 6/27/03, 8/6/03, 11/4/03, 1/28/04

CAP Documents and Audits

AR 000671006, Training for SCBA Bottle Changeout, 7/25/02
AR 00089177, Individual Exited Portal Area with 223 nanocuries of Co-60, 3/27/03
AR 00099660, Unauthorized Vacuum Found in the RCA, Undated
AR 00100334, Change Management - E7RC Has Not Effectively Managed Change in the Respiratory Protection Program, Undated
AR 00100335, Site Training/Documentation May Not Be Adequate in Supporting the Needs of the Respiratory Program, Undated
AR 00100337, Lack of Program Owner for Respiratory Protection Program, Undated
AR 00100915, Health Physics Instrument with Improper Calibration Label, 8/4/03

Section 2PS1:Radioactive Gaseous and Liquid Effluent Treatment and Monitoring

Systems

Procedures, Manuals, and Guidance Documents

Brunswick Steam Electric Plant Off-site Dose Calculation Manual (ODCM), Rev. 26
Brunswick Steam Electric Plant, Unit Nos. 1 And 2 Radioactive Effluent Release Report For 2002, April 28, 2003

Records and Data

Compensatory Sampling summaries for CY2000, 2001, 2002 and monthly summaries from January 2003 through December 2003

CAP Documents and Audits

AR 82300, Adverse Condition Investigation Form, Fission and activation product releases from the Unit 1 Reactor Building Roof Vent monitor (1-CAC-AQH-1264) are elevated, 1/22/2003
AR 88304, Extract, During B2116R1 Refueling outage, WO 131031 and associated clearance on 2-OG-FY-245 required isolation of instrument air –rendering SJAE monitor actuated isolations inoperable.
BNAS 03-029, Radiation Protection Assessment (Nuclear Assessment Section, May 29, 2003
Assessment 81344, Self Assessment Report, February 10-13, 2003

Section 2PS3: Radiological Environmental Monitoring Program

Procedures, Manuals, and Guidance Documents

Brunswick Steam Electric Plant Off-Site Dose Calculation Manual, Rev. 26
CAP-NGGC-0200, Corrective Action Program, Rev. 9
0E&RC-0215, Removal of Materials from the Radiological Control Area
0E&RC-0216, Control and Monitoring of Nonradioactive Plant Waste and Scrap
OE&RC-3101, Radiological Environmental Monitoring Program, Rev. 23
OMST-MET21SA, Met Tower Equipment Calibration and Functional Test, Rev. 12
EVC-NGGC-0001, Operation And Calibration of HNP Environmental Air Samplers, Rev. 4
EVC-NGGC-0002, Operation of The HNP Portable Water Samplers, Rev. 2
EVC-NGGC-0012, Preparation and Counting of Samples for Determination of Gamma Activity
EVC-NGGC-0031, Calibration/Operation of the Canberra Nuclear 9900 Spectroscopy System

Records, Data, and Annual Reports

BSEP 03-0004, Annual Radiological Environmental Operating Report for 2002
2003 Land Use and Garden Census
Results of Environmental Cross Check Program, 4th Quarter, 2001; 1st Quarter, 2002; 2nd Quarter, 2002; 3rd Quarter, 2002
Met Tower Equipment Calibration and Functional Test, 1/21/03, 6/30/03, 7/11/03
Certification and Review Form, Upper Wind Sensor replacement, 9/22/03
Met Tower Bi-weekly Check, 12/16/03, 12/29/03

2000 Brunswick JWF Data Upper and Lower Points
2001 Brunswick JWF Data Upper and Lower Points
2002 Brunswick JWF Data Upper and Lower Points
2003 Brunswick JWF Data Upper and Lower Points
Environmental Air Sample Dry Gas Correction Factor calculations; BNP-1 through BNP-12 inclusive, 2/19/03; BNP-10, 3/18/03; BNP-11, 6/23/03; BNP-12, 6/23/03
SAM Calibration Record: S/N 3, 2/21/03; S/N 66, 3/19/03; S/N 67, 4/15/03, 3/28/03, 3/23/03; S/N 192, 11/10/03; S/N 194, 10/20/03; S/N 358, 2/7/03; S/N 362, 2/7/03
BM-285 Calibration Record: S/N 240, 10-16-03
Germanium Detector No. 1 Calibrations, Multiple Geometries, 5/22/03, 5/25/03, 5/27/03
Germanium Detector No. 2 Calibrations, Multiple Geometries, 5/20/03, 5/21/03, 5/22/03, 5/23/03, 5/28/03
Germanium Detector No. 5 Calibrations, Multiple Geometries, 5/21/03, 5/22/03, 5/23/03
Germanium Detector No. 6 Calibrations, Multiple Geometries, 5/21/03, 5/22/03, 5/23/03, 5/24/03
Fax dated 1/29/04 from Sharon Langdon demonstrating derivation of germanium detector LLD for I-131 using air cartridge geometry

CAP Program Documents

AR 50067, Radiochemistry Laboratory exceeded the 3-sigma level on 2 of 32 analyses
AR 58375, Radiochemistry Laboratory exceeded the 3-sigma level on 1 of 30 analyses
AR 64806, Missing quarterly environmental TLD
AR 79711, Environmental air sampler #204 filter bypass during sample period
AR 84208, Blown fuse on environmental air sampler (#203)
AR 84812, Environmental sample analysis results indicates activity
AR 104206, Failure of ERFIS weather data from Met Tower
AR 105027, Met Tower wind speed data during hurricane Isabel (upper wind speed sensor failure)
AR 107274, Met Tower to ERFIS Interface anomalies

Section 4OA1: Performance Indicator Verification

Occupational Radiation Safety Cornerstone

Procedures and Records

CAP-NGGC-0200, Corrective Action Program, Rev. 9
REG-NGGC-0009, NRC Performance Indicators, Rev. 3
Spreadsheet :NRC Performance Indicators Brunswick Nuclear Plant and BWR Quarterly Average, December 2003

CAP Documents

AR 87227, Adverse Condition Investigation Form, LHRA Barrier Not Controlled by Radiation Control, 3/11/2003
AR 92157, Adverse Condition Investigation Form, Observations Where the Control of Radiological Boundaries Was Not Maintained Appropriately, 5/1/2003

Public Radiation Safety Cornerstone

Procedures and Records

CAP-NGGC-0200, Corrective Action Program, Rev. 9
REG-NGGC-0009, NRC Performance Indicators, Rev. 3
Spreadsheet :NRC Performance Indicators Brunswick Nuclear Plant and BWR Quarterly Average, December 2003

CAP Documents

AR 82300, Adverse Condition Investigation Form, Fission and activation product releases from the Unit 1 Reactor Building Roof Vent monitor (1-CAC-AQH-1264) are elevated, 1/22/2003
AR 88304, Extract, During B2116R1 Refueling outage, WO 131031 and associated clearance on 2-OG-FY-245 required isolation of instrument air –rendering SJAE monitor actuated isolations inoperable

Section 4OA2: Problem Identification and Resolution

Work Orders 503125, 495917, and 488458
AR 114663, Missing Support Bracket on EDG 1 Jacket Water Piping to Turbocharger
AR 114946, EDG 3 Jacket Water Piping Leak Rework Issue
AR 114576, EDG 3 Past Operability Determination

Significance Determination Process Phase III Summary

SRA Analysis Number: Brun 2004-01
Analysis Type: SDP Phase III
Inspection Report # : 05000325,324/2004002
Plant Name: Brunswick
Unit Number: 2
Enforcement Action # : 04-076

I. Background

Performance Deficiency - Timely and appropriate corrective action, commensurate with the potential safety significance, was not taken for leakage identified from EDG #3 jacket water cooling system on December 7, 2003. Missing pipe supports on the jacket water cooling system resulted in misalignment of a system pipe coupling which caused system leakage. Maintenance practices and controls for repairs performed on December 8, 2003 to correct the deficiency caused the leakage to increase which was not detected due to the failure to perform appropriate post-maintenance testing. Operability assessments of system leakage, on both December 7, 2003, and initially on January 4, 2004, did not consider the potential impact of a loss of off-site power on the ability of the demineralized water system to make up to the EDG to compensate for the degraded condition (inspector identified). This performance deficiency resulted in Units 1 and 2 not meeting TS Limiting Condition of Operation 3.8.1, AC Sources-Operating.

Exposure Time - the exposure time includes three separate windows.

1. EDG 3 available with an operator action to refill the jacket water cooling system for 26 days,
2. EDG 3 unavailable for 37.5 hours because the demineralized water storage tank level was inadequate for gravity fill
3. EDG 3 was unavailable due to repairs for 28 hours (see below for discussion on recovery)

Date of Occurrence - 12/8/2003

II. Safety Impact: White for unit 2
Green for unit 1

III. Risk Analysis/Considerations
Assumptions:

1. EDG 3 available with an operator action to refill the jacket water cooling system for 26 days, operator action =0.1 based on SPAR HRA evaluation.

2. EDG 3 unavailable for 65.5 hours without recovery based on:

The demin water storage tank level was inadequate for gravity fill for 37 .5 hours.

The EDG was unavailable due to repair of the performance deficiency for a period of 28 hours.

3. PRA Model used for basis of the risk analysis:

The SPAR model, rev 3i, was used with several significant revisions. Revisions included:
the update for the NUREG 5496 loss of offsite power, recovery curves, and EDG mission time;
revisions to fault trees SDC, CSS, SPC-a, SPC-b to allow cross tie electrical power to these systems;
revised RSW-HXA,RHR-A-SS, and RHR-B-SS to allow credit for CSW/NSW cross tie;
revised CVS to remove the requirement for containment purge for containment success;

revised the failure probability of one SRV fails to close from 0.18 to .031 based on latest SPAR value;
revised the HPCI injection valve failure rate from 0.2 to 0.02 based on a discussion with INEEL.

4. Because the SPAR is a unit one model, EDG 1 was used to calculate the risk increase of EDG 3 on unit 2. EDG 3 was used to calculate the risk impact of this finding on unit 1.

The Phase Two SDP Notebook was used to screen the finding using the same logic as described for the SPAR evaluation. The phase Two notebook resulted in a white finding based on internal events.

Significant Influence Factor(s) [if any]:

LERF Evaluation: Because the dominant risk sequences involved the loss of offsite power and resulted in high pressure sequences, the impact on LERF must be analyzed. A typical factor of 1 would be used to calculate the delta LERF for a typical BWR with a Mark 1 containment based on MC 609, appendix H. However, because of the robust containment design at Brunswick, a factor of 0.1 was selected as the apropriate multiplier for the high pressure sequences with a dry containment. The 0.1 factor was provided by Mr. Bob Palla in the attached memo. The LERF sequences that were determined to result in vessel breach at high pressure were LOOP 49, LOOP 52-06, and LOOP 52-03. Therefore, only the results for these sequences were utilized in the LERF analysis.

External Events Evaluation: The effects of external events were considered for this deficiency and found to have a negligible contribution.

Fire - This finding is dominated by loss of offsite power and SBO sequences. Fires that impacted significant mitigation equipment did not significantly increase the initiating event likelihood. A switchgear based fire that increased the initiating event frequency was postulated with a frequency of 5e-3/yr and was used as a bounding initiating event frequency for a fire induced LOOP. This produced a delta CDF of 7.8e-8.

Earthquake and Tornado - The two issues relevant to this finding were the impact on the loss of the deminerialized water storage tank for filling the EDG head tank and the impact on the loss of offsite power frequency. Because the tank fragility was only a concern during a loss of offsite power, a bounding analysis was performed to evaluate a loss of offsite power due to earthquake and tornado with no recovery and no credit for recovery of the EDG. Based on an earthquake of 300cm/sec/sec being the minimum to cause a loss of offsite power, the frequency of interest was determined to be 6.58e-5/yr. Based on a review of the IPEEE, Tornados of F2 or greater are required to cause a loss of offsite power. The IPEEE provided a frequency of F2 or greater as 9.3e-5/yr. Therefore, an initiating event frequency of 1.58e-4/yr was used for this analysis. This resulted in a delta CDF of 1.48e-8 for a 28 day exposure period.

Huricanes - Plant procedures require a plant shutdown and other compensatory actions in the event of a pending huricane. Manual Chapter 609, Appendix G was utilized to estimate the risk contribution of a loss of offsite power during shutdown conditions. This result was a delta CDF of 1e-8.

Therefore, the results of this bounding analysis concluded that the maximum impact of external events would be less than 1e-7/year, which is not enough to change the conclusions of this analysis.

IV. Calculations

<u>BASE CASE</u> - CDF = 1.53E-5/yr

<u>NON-CONFORMING CASE</u> -

Evaluation for Unit 2:
EDG failed. Used T&M= 1 because there was no common cause connection, CDF = 1.05e-4/yr
EDG degraded with a failure of recovery of 0.1. Added 0.1 to the base T&M term to evaluate this condition. CDF = 2.16e-5/ yr
Evaluation for Unit 1:
EDG failed. Used T&M= 1 because there was no common cause connection, CDF = 2.3e-5/yr
EDG degraded with a failure of recovery of 0.1. Added 0.1 to the base T&M term to evaluate this condition. CDF = 1.59e-5/ yr

<u>DELTA CDF</u>

Evaluation for Unit 2:
EDG failed. Used T&M= 1 because there was no common cause connection, Delta CDF = 1.02e-8/hr
EDG degraded with a failure of recovery of 0.1.Delta CDF = 7.19e-10/hr

Evaluation for Unit 1:
EDG failed. Used T&M= 1 because there was no common cause connection, Delta CDF = 9.22e-10/hr
EDG degraded with a failure of recovery of 0.1. Delta CDF = 6.56e-11/hr

<u>DELTA CDF FOR EXPOSURE TIME</u>

Unit 2 Risk
EDG failed without recovery * 65.5 hours + EDG failed with recovery *26 days = ICDF
(1.02E-8 * 65.5) + (7.19e-10 * 24*26) = 1.1e-6 ICDF - White Finding

LERF Evaluation
The sequences were reviewd to determine those resulting in core damage at high pressure. Based on those sequences only the ICDF is (7.9e-9 *65.5) + (5.3e-10 *24 *26)= 8.5e-7
Internal events high pressure ICDF * .1 = 8.5e-7 * 0.1 = 8.5e-8 delta LERF = Green Finding

Unit 1 Risk
EDG failed without recovery * 65.5 hours + EDG failed with recovery *26 days = ICDF
(9.22E-10 * 65.5) + (6.56e-11 * 24*26) = 1.0e-7 ICDF - Green Finding

LERF Evaluation, (used all ICDF as screening measure) Internal events ICDF * .1 = 1.0e-7 * 0.1 = 1.0e-8 delta LERF = Green finding

V. Conclusions/Recommendations - Risk increase over the base case was White for the impact on unit 2 and Green for unit 1.

A phase 3 analysis was performed using the Brunswick SPAR model. Assumptions critical to this evaluation involved the ability of operators to maintain jacket water level by making up to the leaking system. This analysis assumed that for a period of 26 days, the #3 emergency diesel generator (EDG) could have been recovered by an operator refilling the jacket water cooling system. For a 37.5 hour period when the demineralized water tank level was too low to support gravity feed the EDG was unavailable with no recovery. The EDG was also unavailable due to repair of the performance deficiency for a period of 28 hours. The probability of the operator failing to recover the EDG was estimated at 0.1 using SPAR human reliability analysis (HRA) methods. The SPAR analysis identified the loss of offsite power with failures to supply power to the emergency busses as dominant risk sequences. Because some of the sequences also involve failures to depressurize the reactor, the analysis of large early release frequency (LERF) was also an important consideration. However, based on the unique containment structure of Brunswick, LERF was determined to not be a significant factor for the loss of power sequences. External events were considered but also determined not to be significant contributors to this evaluation because their initiating event frequencies were small compared to the loss of offsite power frequency. The SPAR model result for the change in core damage frequency (CDF) was 1.1e-6. Based on these results, this performance deficiency was found to be of low to moderate safety significance and therefore has been classified as White for unit 2. The safety significance of this performance deficiency for unit 1 was of very low safety significance and therefore has been classified as Green.

VI. References

Phase I Screening Sheets
Phase II sdp sheets
LERF Memo from Bob Palla
LERF Sequence analysis
Zip file of model
sequence and cut set files