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

April 25, 2000

Carolina Power and Light Company
ATTN: Mr. J. S. Keenan
Vice President
Brunswick Steam Electric Plant
P. O. Box 10429
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SUBJECT: NRC INTEGRATED INSPECTION REPORT NOS. 50-325/00-02 AND 50-324/00-02

Dear Mr. Keenan:

This refers to the inspection conducted on February 27, through April 1, 2000, at the Brunswick reactor facility. The enclosed report presents the results of this inspection.

During the inspection period, your conduct of activities at the Brunswick facility was generally characterized by safety-conscious operation, sound engineering and maintenance practices, and careful radiological work controls.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be placed in the NRC Public Document Room (PDR).

Sincerely,

/RA/

Brian R. Bonser, Chief
Reactor Projects Branch 4
Division of Reactor Projects

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

Enclosure: NRC Inspection Report

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REGION II

Docket Nos:	50-325, 50-324
License Nos:	DPR-71, DPR-62
Report No:	50-325/00-02, 50-324/00-02
Licensee:	Carolina Power & Light (CP&L)
Facility:	Brunswick Steam Electric Plant, Units 1 & 2
Location:	8470 River Road SE Southport, NC 28461
Dates:	February 27 - April 1, 2000
Inspectors:	T. Easlick, Senior Resident Inspector E. Brown, Resident Inspector E. Guthrie, Resident Inspector J. Coley, Reactor Inspector (Sections M2.1, M7.1) F. Wright, Reactor Inspector (Sections R1.1, R1.2, R5.1) E. Girard, Senior Reactor Inspector (Section E8.1)
Approved by:	B. Bonser, Chief, Projects Branch 4 Division of Reactor Projects

Enclosure

EXECUTIVE SUMMARY

Brunswick Steam Electric Plant, Units 1 & 2
NRC Inspection Report 50-325/00-02, 50-324/00-02

This integrated inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers a 5-week period of resident inspection; in addition, it includes the results of maintenance and radiological protection inspections by regional inspectors.

<u>Operations</u>

- Operations personnel generally demonstrated strong command and control of control room activities during startup activities on Unit 1, following a scheduled refueling outage. Procedural requirements for command, control, and communications were met. Overall, the startup was completed effectively and efficiently. Senior site management, as well as department and first line supervisors, demonstrated strong supervisory oversight of observed activities (Section O1.1).

- The licensee's response to the loss of offsite power (LOOP) and subsequent diesel generator (DG) 2 excitation transformer failure was satisfactory. The licensee correctly classified the LOOP as an Unusual Event. Proper communications, satisfactory use of procedures, and appropriate concern for reactor safety was observed. The licensee addressed an inspector question regarding the extent of DG lubrication oil contamination during diesel operations in a halon environment. The licensee determined that the DG oil was contaminated and needed to be replaced. The corrective actions proposed by the licensee were determined to be appropriate (Section O1.2).

- A pre-startup inspection of the drywell found that general housekeeping and material condition were adequate to ensure the drywell was free of foreign material and ready to support an operational cycle. Health physics support and coordination for this activity were effective in reducing personnel exposure (Section O2.1).

- The restart review conducted by the Plant Nuclear Safety Committee adequately evaluated the overall site organization's readiness to restart Unit 1. The maintenance rule impact of various component failures was discussed (Section O7.1).

<u>Maintenance</u>

- Inservice examination activities observed were performed using approved procedures by certified skilled examiners. The inspection results were properly recorded and evaluated in accordance with the appropriate test procedures. The repair package reviewed was detailed and complete (Section M1.2).

- A limitation in the test methodology for secondary containment was identified by the inspectors. Secondary containment leakage was quantified without consideration of all the operating configurations. The licensee revised the periodic test to address testing both reactor building railroad doors to assure that leakage through either boundary was maintained less than Technical Specifications requirements. Review of the maintenance history and subsequent testing during the recent refueling outage demonstrated that secondary containment integrity had been maintained, despite the test methodology limitations (Section M3.1).

Engineering

- During the Unit 1 refueling outage, a modification allowing large motor loads on the 4160 volt (V) balance of plant bus to be shed in the event of a loss of coolant accident, was installed. This modification was necessary due to a potentially large load demand on the offsite grid, which would have an effect on the 4160V emergency bus (e-bus) voltage. The probabilistic safety analysis and maintenance rule reviews for this modification had not been completed before the modification to the DG logic circuitry was implemented. Subsequent reviews by the licensee determined that no significant increase in risk was demonstrated (Section E2.2).

- Fuel rod replacement activities were conducted consistent with regulatory requirements and site administrative controls. Associated safety screens were determined to adequately analyze the safety significance of the maintenance activity consistent with 10 CFR 50.59, Changes, tests and experiments (Section E3.1).

Plant Support

- Licensee radiation surveys, postings, access controls, and radiological work controls were effective and performed in accordance with regulatory requirements. Good health physics planning and implementation in capturing and encasing a highly radioactive source range monitor guide tube was observed (Section R1.1).

- The Brunswick as low as reasonably achievable program was effective in reducing site collective personnel radiation doses (Section R1.2).

- The inspectors reviewed the resumes of the senior vendor technicians and found the technicians met the licensee’s TS requirements (Section R5.1).

Report Details

Summary of Plant Status

Unit 1 began the report period shutdown for scheduled refueling activities. The unit was returned to 100 percent rated thermal power (RTP) on March 27 at 1:41 a.m., where it remained for the remainder of the inspection period. The refueling outage lasted 27 days.

Unit 2 began the report period operating at 100 percent RTP. On March 3 power was reduced to 60 percent RTP due to a loss of offsite power (LOOP) on Unit 1 and the loss of one diesel generator (DG). The unit was returned to 100 percent RTP on March 4 where it remained through the end of the inspection period.

I. Operations

O1 Conduct of Operations

O1.1 Unit 1 Control Room Activities During Startup

a. Inspection Scope (71707)

The inspectors observed operator performance during preparations for, and activities during, the Unit 1 startup following a scheduled refueling outage. The inspectors reviewed procedures and assessed supervisory oversight, command and control, and communications during control room activities.

b. Observations and Findings

The inspectors observed generally strong command and control of control room activities. The startup activities following a scheduled refueling outage on Unit 1 were controlled by the unit senior control operator (SCO) and a dedicated senior reactor operator (SRO) responsible for the startup. Clear lines of responsibility were delineated between these two individuals. Three reactor operators were assigned to Unit 1 during the startup. Operators used appropriate procedures and control rod withdrawal sequence sheets. The operators effectively controlled routine control room activities as well as startup surveillance activities. Additionally, maintenance, instrumentation and control, and engineering support activities during testing and maintenance were the subject of close attention to detail and oversight. Reactor engineering personnel provided comprehensive direction and oversight throughout the startup. The inspectors observed that extra personnel were assigned to the shift and generally performed their assigned tasks without error.

The inspectors observed that the control room activities were monitored and observed by a member of site management. The plant general manager was in the control room on a daily basis to discuss plant and equipment status. The operations manager or his designee was also observed in the control room for the duration of the startup and participated in briefings and routine meetings.

c. Conclusions

Operations personnel generally demonstrated strong command and control of control room activities during startup activities on Unit 1, following a scheduled refueling outage. Procedural requirements for command, control, and communications were met. Overall, the startup was completed effectively and efficiently. Senior site management as well as department and first line supervisors, demonstrated strong supervisory oversight of observed activities.

O1.2 Loss of Offsite Power and DG Failure on Unit 1

a. Inspection Scope (71707, 37551, 93702)

The inspectors responded to a Unit 1 LOOP and subsequent DG 2 excitation transformer failure on March 3. The inspectors observed operator response and restoration of the plant. The inspectors reviewed the transient data, the licensee's site investigation review, root cause investigations, and the appropriateness of actions taken by operations and engineering to restore the plant.

b. Observations and Findings

On March 3 Brunswick Unit 1 was in cold shutdown for a refueling outage. Unit 2 was operating at 100 percent RTP. Testing was in progress on the 230 kilovolt (KV) electrical system for Unit 1. Transmission workers, not directly employed by the Brunswick Plant, were performing the 230 KV breaker and relay testing. The licensee determined that a transmission worker manipulated the wrong test switch which caused a LOOP to Unit 1. Reviews conducted by both the licensee and the inspectors found that the electrical distribution system functioned correctly. The LOOP to Unit 1 started all four emergency DGs as expected. When the LOOP occurred the licensee classified the event as an Unusual Event.

The inspectors observed satisfactory operator response following the LOOP. The inspectors observed proper communications, satisfactory use of procedures, and appropriate concern for reactor safety.

While restoring the electrical distribution system, following the LOOP, the DG 2 excitation transformer failed. The diesel with the failed excitation transformer tripped off and locked out on an overcurrent protective action. All of the DG excitation transformers are located in the basement of the diesel building. The halon fire suppression system, located in the basement of the diesel building, actuated when the area filled with smoke from the failed transformer. The halon discharge created enough pressure to unexpectedly actuate a tornado damper, which was located between the basement and upper elevations of the diesel building. Halon was free to escape into the upper elevations of the building with the tornado damper door open. The three other operating DGs ingested a halon and air mixture through their individual air intakes located on the upper elevation of the diesel building. The air intakes for all of the diesel generators are located on one elevation of the diesel building with the same air space communicating to the entire elevation.

The inspectors observed the licensee’s response to the failure of the DG 2 excitation transformer. When the transformer failed and DG 2 subsequently tripped the emergency bus that was powered by DG 2 lost power. The loss of power resulted in a second loss of shutdown cooling (SDC) to the reactor vessel. The first loss of SDC occurred following the LOOP. In both cases operators restored SDC promptly to the reactor vessel, with minimal temperature increases in the reactor vessel both times.

The inspectors noted that the exhaust from the running DGs was orange in color, during the LOOP. The inspectors questioned the licensee regarding contamination of the lubrication oil in the operating DGs due to halon being ingested into the engine during operations. The licensee sampled the oil and found that all three diesels halon concentrations were above the vendor specified limits for halons of 50 parts per million (ppm). DG 1 had concentrations of 211 ppm, while DG 3 and DG 4 had concentrations of 53 ppm. The licensee scheduled the three DGs for the oil to be changed during upcoming scheduled preventive maintenance. The concern with halon contamination was degraded alkalinity allowing corrosion and viscosity changes. The changes in the oil properties were long term effects on the DG and therefore not immediate operability concerns.

Unit 2 commenced a Technical Specification (TS) required shutdown after DG 2 failed due to degraded redundant emergency alternating current power sources. The licensee activated their emergency operations facilities to assist with the complex recovery of the plant. This was not required by licensee procedure. The inspectors observed those activities and found no deficiencies. The Unusual Event was terminated when electrical busses were inspected, re-energized, and the cause of the LOOP was known.

The inspectors reviewed the licensee’s significant root cause investigation and determined the results to be appropriate. The inspectors additionally found that the corrective actions proposed by the root cause investigation were appropriate. The licensee scheduled replacement of the other three DG excitation transformers before the end of the year 2000.

c. <u>Conclusions</u>

The licensee’s response to the LOOP and subsequent DG2 excitation transformer failure was satisfactory. The licensee correctly classified the LOOP as an Unusual Event. Proper communications, satisfactory use of procedures, and appropriate concern for reactor safety was observed. The licensee addressed an inspector question regarding the extent of DG oil contamination during diesel operations in a halon environment. The licensee determined that the DG oil was contaminated and needed to be replaced. The corrective actions proposed by the licensee were determined to be appropriate.

O2 Operational Status of Facilities and Equipment

O2.1 Drywell (DW) Closeout Inspection

a. Inspection Scope (71707)

On March 21 the inspectors accompanied the plant manager on an inspection of the Unit 1 DW in preparation for closeout of the DW.

b. Observations and Findings

The inspectors toured all elevations of the DW to verify that the material condition supported plant startup. Each downcomer connecting the torus and DW was also inspected. The areas were found to be generally clean with the exception of several small pieces of debris including plastic tiewraps, metal labels, wire, and tape. These items were later removed from the DW. No deficiencies were noted. Health physics support for this activity was excellent. A detailed pre-job briefing was conducted that included personnel safety, foreign material exclusion area, and radiological information.

c. Conclusions

A pre-startup inspection of the DW found that general housekeeping and material condition were adequate to ensure the DW was free of foreign material and ready to support an operational cycle. Health physics support and coordination for this activity were effective in reducing personnel exposure.

O7 Quality Assurance in Operations

O7.1 Startup Plant Nuclear Safety Committee (PNSC) Review Meeting

a. Inspection Scope (71707)

The inspectors observed startup assessment activities conducted by the PNSC for the Unit 1 refueling outage which began on February 25.

b. Observations and Findings

On March 18 and 20 the inspectors attended the PNSC restart review for Unit 1. The readiness of each organization to support restart activities was discussed by the PNSC and affirmed by the each responsible supervisor. Areas addressed during the meeting included the completion of emergent repairs and remaining scheduled work activities. Any activities identified that were not already in the plant schedule were flagged as exceptions. The meeting generated several exception items for completion. The inspectors noted that the maintenance rule impact of various component failures identified during the outage or as a result of the LOOP event were discussed. In addition PNSC attention was placed on safety-related work that was deferred from the outage to discuss the possible impact of performing some of the activities online. The inspectors determined that the meeting was adequate to assess restart readiness.

c. Conclusions

The restart review conducted by the PNSC adequately evaluated the overall site organization's readiness to restart Unit 1. The maintenance rule impact of various component failures was discussed.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Maintenance Activities (61726, 62707)

The inspectors reviewed all or portions of the following surveillance test and/or work activities:

- Periodic Test 0PT-11.1.2, Automatic Depressurization System and Safety Relief Valve Operability Test, Revision (Rev.) 32; and
- Periodic Test 0PT-15.4, Secondary Containment Integrity, Rev. 19.

The inspectors attended pre-job briefings for the surveillance tests. The participants in the briefings discussed human error precursors, operating experience and verified that no other testing that could interfere with the activities was in progress. In the case of the safety relief valve testing, practice runs were performed to ensure that proper communications and test responsibilities were established. During the activities, effective shift supervisory oversight was present, and procedures used were of the proper revision. Technicians were knowledgeable of the evolutions and expected instrument responses and used satisfactory three-part communications. The testing was completed satisfactorily in accordance with TS.

M1.2 Inservice Inspection (ISI) - Observation of Work Activities

a. Inspection Scope (73753, 57090) Unit 1 (except as noted)

The inspectors observed four methods of non-destructive examination, reviewed a code repair package, verified ISI program requirements for Class 2 pressure retaining piping, and reviewed outage documentation which included outage plans, examination procedures, and examiner certification documentation. These observations were performed to determine whether the ISI, repair, and replacement of Class 1, 2, & 3 pressure retaining components at the Brunswick facility were performed in accordance with TS, the American Society of Mechanical Engineers (ASME) Code (1989 Edition, Sections XI & V), and correspondence between NRC staff and the licensee.

b. Observations and Findings

The inspectors observed the manual ultrasonic examinations of one reactor core isolation cooling system (RCIC) weld, two feedwater system welds and one weld repair overlay on the recirculation system. The component identifications for these welds were 1E513-4-3-FWRCICB6A, 1B21-1-1-FWRFWA8, 1B21-1-1-SWB and 1B32RECIRC-4-B-1. No defects were observed during these examinations.

Automated ultrasonic examinations and review of examination data for the portion of the reactor core shroud H-1 and H-5 welds in the first 90 degree quadrant were also observed by the inspectors.

A completed Unit 2 outage repair and replacement activity package was reviewed for repairs to the standby liquid control system (SLC) completed during the last Unit 2 refueling outage. Review of radiographic film for six Unit 2 SLC system welds revealed that radiographic film quality and weld quality were satisfactory. The component identifications for these welds were FW-2-C41-54, FW-2-C41-56, FW-2-C41-58, FW-2-C41-59, FW-2-C41-63, and FW-2-C41-65. No findings were identified during the examinations observed or as a result of the repair and replacement review.

Pulsed eddy current acquisition activities and analyses of the data were observed for the 1FWH-3A feedwater heat exchanger outer shell. No significant erosion wear of the heat exchanger outer shell was identified during the activities observed.

The inspectors held discussions with the ISI program engineer, reviewed drawings, and documentation of the ISI program to determine if ISI program requirements for class 2 piping above 3/8 inch wall thickness had been implemented in accordance with the 1989 ASME Code. The inspectors found that the licensee had properly implemented the code.

A repair package which was required to permanently repair a temporary non-code repair of the conventional service water six inch supply to vital header was reviewed. The licensee had followed Generic Letter (GL) 90-05, “Guidance for Performing Temporary Non- Code Repair of ASME Code Class 1, 2, and 3 Piping,” when applying the non-code repair criteria. An engineering evaluation was performed and code relief for the non-code repair was obtained.

c. Conclusions

Inservice examination activities observed were performed using approved procedures by certified skilled examiners. The inspection results were properly recorded and evaluated in accordance with the appropriate test procedures. The repair package reviewed was detailed and complete.

M3 Maintenance Procedures and Documentation

M3.1 Secondary Containment Test Methodology Limitations

a. Inspection Scope (61726)

On February 14 the inspectors reviewed risk assessments and periodic tests (PTs) related to secondary containment for conformance to applicable TS, regulatory guidance, and updated final safety analysis report (UFSAR) requirements. Periodic Test 0PT-15.4, Secondary Containment Integrity, Rev. 16, was used to verify that secondary containment integrity was operable. This was accomplished by verifying that one standby gas treatment (SBGT) subsystem could maintain secondary containment pressure at greater than 0.25 inches water gauge at 3000 cubic feet per minute (cfm). The margin above the TS limits obtained from this test was utilized in an engineering procedure to allow maintenance activities on secondary containment doors, hatches, and penetrations while at 100 percent RTP.

b. Observations and Findings

During observation of various maintenance activities including spent fuel cask movement and hurricane preparations, the inspectors noted that both reactor building railroad doors were relied upon during certain activities independently to maintain secondary containment integrity. The inspectors reviewed 0PT-15.4 and determined that only the outer railroad door was tested. The inspectors questioned why both doors were not independently verified during the PT to meet the surveillance requirement, since the amount of leakage through the interior railroad door would not be accounted for in the secondary containment integrity test. This omission could have allowed the leakage through the internal railroad door to exceed TS requirements or permitted maintenance to the secondary containment boundary while online that could have exceeded TS. The licensee indicated initially that any gross leakage was identified by visual inspections conducted prior to the surveillance test and was sufficient to determine secondary containment integrity. The inspectors questioned whether personnel performing the visual inspection could quantify the leakage through the inner railroad door. After additional review the licensee determined that the PT needed to be revised to address testing of both doors to assure that leakage through either boundary was maintained less than TS requirements.

The inspectors reviewed the maintenance records and noted that no work tickets identifying significant deficiencies in any railroad door were present. Additionally, testing performed on March 22 and 23 confirmed the integrity of secondary containment was satisfactory. As a result, the inspectors concluded that secondary containment integrity was maintained despite the limitations of the test methodology.

c. Conclusions

A limitation in the test methodology for secondary containment was identified by the inspectors. Secondary containment leakage was quantified without consideration of all the operating configurations. The licensee revised the periodic test to address testing both reactor building railroad doors to assure that leakage through either boundary was maintained less than TS requirements. Review of the maintenance history and subsequent testing during the recent refueling outage demonstrated that secondary containment integrity had been maintained, despite the test methodology limitations.

M7 Quality Assurance in Maintenance Activities

M7.1 Licensee Assessments of ISI Activities (73753)

The inspectors evaluated the effectiveness of licensee's controls for identifying, resolving and preventing problems in ISI by reviewing a significant adverse condition investigation report (AR 16029). The investigated concern identified missed pressure tests and VT2 visual examinations for the CL-2 SGT system. After thorough examination of the identified problem, the inspectors concluded that the licensee's controls were effectively identifying and resolving issues within the corrective action program.

III. Engineering

E2 Engineering Support of Facilities and Equipment

E2.1 Unit 1 Core Verification Inspection (37551)

The inspectors reviewed the underwater video tape inspection of the Unit 1 reactor core following the refueling activities. The inspectors verified fuel assembly location and orientation as required by the Unit 1 B1C13 Fuel Core Loading Pattern (FCLP) document. All of the fuel assemblies were in the required position with proper orientation. The fuel assembly identification numbers were checked against the FCLP document to verify position. Fuel assembly height was also verified to ensure that the fuel was properly seated in the reactor. No discrepancies were noted by the inspectors.

E2.2 Loss of Coolant Accident (LOCA) Load Shed Modification

a. Inspection Scope (37551, 62707)

The inspectors verified that required technical reviews and post maintenance testing were conducted for the LOCA load shed modification. The inspectors reviewed the applicable 10 CFR 50.59 screen, pending calculation updates, updated procedures, and unreviewed safety question (USQ) determination.

b. Observations and Findings

During the Unit 1 refueling outage, the licensee installed a modification which would allow large motor loads on the 4160 volt (V) balance of plant (BOP) buses to be shed in the event of LOCA. The modification was necessary due to a potentially large load demand on the offsite grid, which would have an effect on the 4160V emergency bus (e-bus) voltage. During a LOCA, the voltage on an e-bus could drop low enough, due to the large loading on the electrical bus to trip the undervoltage protective relays, which would result in the tripping of the master/slave breakers and a LOOP. The shedding of these BOP loads would prevent an undervoltage condition on the e-bus as a result of excessive grid load thus allowing accident mitigation loads to be powered from offsite power during a LOCA.

The inspectors verified that applicable drawings and procedures had been revised and updated in the licensee tracking system. The modification package stated that a known single-failure preventing the LOCA load shed scheme from stripping selected loads on a

LOCA signal could have resulted in a LOOP. The single-failure was identified by the licensee as the failure of the BOP loads to shed in the event of a LOCA with increased offsite loading. The inspectors questioned the risk impact this non-safety related single-failure would have on any safety-related component. The licensee indicated that a technical review prior to installation based on the risk impact had not been performed.

The inspectors reviewed the licensee ESR procedure for the screening criteria guidance for determining required technical reviews. The inspectors questioned the thoroughness of the technical reviews and determined that the probabilistic safety analysis (PSA), station blackout (SBO), as well as maintenance rule (MR) program reviews, should have been conducted during the initial modification review based on the screening criteria guidance. Discussions with the licensee revealed that the BOP load shed function needed to be reviewed for possible scoping into the MR. The licensee stated that the need for the PSA and MR program reviews was missed, however the electrical reviewers had considered SBO. Nuclear Condition Reports 00-18007, Missed Maintenance Rule Review on ESR 99-207, and 00-18485, ESR Potentially Missed PSA Review, were initiated to address the missed MR and PSA reviews. During subsequent discussions, the licensee indicated that the modifications would not significantly increase the risk impact or failure probability for any safety-related component.

c. Conclusions

During the Unit 1 refueling outage, a modification allowing large motor loads on the 4160 volt balance of plant bus to be shed in the event of LOCA, was installed. This modification was necessary due to a potentially large load demand on the offsite grid, which would have an effect on the 4160V emergency bus voltage. The probabilistic safety analysis and maintenance rule reviews for this modification had not been completed before the modification to the DG logic circuitry was implemented. Subsequent reviews by the licensee determined that no significant increase in risk was demonstrated.

E3 Engineering Procedures and Documentation

E3.1 Nonirradiated Bundle Fuel Rod Replacement

a. Inspection Scope (37551, 62707)

The inspectors reviewed the adequacy of the administrative controls for the replacement of 27 nonirradiated fuel rods that were potentially defective.

b. Observations and Findings

In mid-December, the fuel manufacturing vendor notified the licensee that the fuel assemblies received contained potentially defective fuel pellets. The potentially defective pellets were identified to be within 27 fuel rods in 24 new fuel bundles. The 24 fuel bundles were intended to be loaded into the Unit 1 core during the current refueling outage. The licensee developed a repair plan, wherein the fuel vendor would replace the potential defective fuel rods using the spent fuel pool (SFP) located in the Unit 1 Reactor Building (RB). The fuel rod replacement was completed without incident in early January 2000.

The inspectors reviewed the licensee's TSs, UFSAR, and associated 10 CFR 50.59 safety screens performed for this activity and questioned the adequacy of the assessment performed for this evolution. The licensee indicated that all activities were considered maintenance activities and were performed under existing procedures which had received a safety screen under 10 CFR 50.59. The safety screens indicated that these activities did not represent a change to the facility nor could have caused a previously unidentified indirect effect on safety related systems, structures, or components. This conclusion was based on the activity being bounded by the fuel handling accident (FHA) analysis, as described in chapter 15 of the UFSAR.

The inspectors reviewed the FHA analysis and concluded that maintenance on the fuel bundles was within the capabilities of associated RB systems to preclude a release in excess of 10 CFR 100 limits in the event of a dropped new fuel bundle or nonirradiated fuel rods in the SFP. This conclusion was based on the following: the fuel bundles for this activity not having been irradiated; the maximum drop height as analyzed being less than the 32 feet assumed in the FHA analysis; all equipment used being the same equipment used during refueling; and that the equipment was used in the same manner as during refueling or other fuel bundle maintenance activities.

c. Conclusion

Fuel rod replacement activities were conducted consistent with regulatory requirements and site administrative controls. Associated safety screens were determined to adequately analyze the safety significance of the maintenance activity consistent with 10 CFR 50.59, Changes, test and experiments.

E8 Miscellaneous Engineering Issues (92903)

E8.1 (Closed) Inspection Followup Item (IFI) 50-325(324)/98-03-01: Completion of MOV Program Followup Items.

a. Inspection Scope

This IFI tracked the licensee's implementation of commitments to address the findings of a Generic Letter 89-10 motor-operated valve (MOV) inspection. The commitments included modifications to enhance the performance of MOVs, differential pressure tests to evaluate the capabilities of motor-operated butterfly valves and other MOVs, and an industry survey to confirm the acceptability of valve factors used in globe valve calculations. The status of the commitments was last reviewed in Inspection Report 50-325(324)/98-10 (IR 98-10), which reported that some of the commitments were complete and that work was progressing on the others. IR 98-10 indicated that the IFI would remain open for further review of the licensee's implementation of the butterfly valve differential pressure testing and globe valve survey commitments. The butterfly valve testing was of particular interest because of the limited testing that had been completed at the time and because of a recent motor-operated butterfly valve failure. The globe valve survey was identified for further review because NRC inspectors found that the survey documents contained errors and/or unclear entries.

In the current inspection, NRC inspectors examined the status of the butterfly valve testing and the globe valve survey. In addition, they examined the status of two other commitments that were not complete at the time of the last inspection. These were: (1) a commitment to modify MOVs to increase their thrust capabilities (margin enhancements), and (2) a commitment to reduce the seating torque for MOV 2-E11-F024B. The inspectors also questioned whether the licensee still planned to provide a final closure letter by January 31, 2001, confirming completion of all of the commitments.

b. Observations and Findings

BSEP 98-0058, Commitment 1 - Margin Enhancement Modifications

The licensee committed to implement modifications to increase the thrust capability margins of thirteen MOVs. Previously (IR 98-10), NRC inspectors verified completion of the modifications to four of the MOVs. In the current inspection, the inspectors reviewed the following Work Request/Job Orders and verified that the margin enhancement modifications on the remaining MOVs (nine) were completed (seven) or were scheduled to be completed (two) in accordance with the licensee's commitment: 98-ADYA1, 98-ADYF1, 98-AHHA1, 99-AETC1, 98-AHHB1, 99-AETD1, 98-AFNZ1, 98-AETP1, 98-AFPA1, 98-AETQ1, 98-AEUG1, 98-AEUR1, and 98-AEUS1.

BSEP 98-0058, Commitment 2 - Differential Pressure Tests of Gate, Globe, and Butterfly Valves

The licensee committed to conduct differential pressure tests of 27 valves to evaluate their capabilities. The majority (15) of the valves to be tested were butterfly valves. Previously (IR 98-10), NRC inspectors verified that the licensee had completed the tests on 7 of the 27 valves, including 2 butterfly valves. In the current inspection, the inspectors focused on the remaining 13 butterfly valve tests. The inspectors reviewed the following test evaluation reports and verified that all of the butterfly valve tests in the commitment had been completed: 1-SW-V680 (tested 11/9/98 and 8/16/99), 2-SW-V680 (tested 12/7/98), 1-SW-V294 (tested 12/18/98), 2-SW-V103 (tested 1/5/99), 2-SW-V106 (tested 1/5/99), 1-SW-V679 (tested 1/26/99), 2-SW-V679 (tested 1/26/99), 1-SW-V117 (tested 3/24/99), 2-SW-V3 (tested 5/8/99), 2-SW-V36 (tested 5/8/99), 2-SW-V13 (tested 6/3/99), 2-SW-V105 (tested10/14/99), and 2-SW-V102 (tested11/11/99). The tests determined that the valves were currently capable of performing their safety functions. However, in most instances, the seating torque obtained was greater than expected. The test evaluation reports indicated that the setup calculations for the valves would be revised to incorporate the results of the differential pressure tests.

BSEP 98-0058, Commitment 6 - Industry Survey of Globe Valves to Confirm Acceptability of Valve Factor Assumption

The licensee committed to perform an industry survey to confirm the adequacy of the valve factor applied in calculating globe valve thrust requirements. As noted previously, the inspection documented in IR 98-10 found that there were errors and/or unclear entries in the globe valve survey documents. In the current inspection, the inspectors found that the licensee had incorporated all of the results of the survey into calculation BNP-MECH-MOV-VF, Review of BNP As Tested Valve Factors & Determination of VF Values to be Used for BNP GL 89-10 Motor-Operated Valves, Revision 4. The inspectors reviewed this document and found that it was generally clear, error free, and included valve factor information to support acceptability of the licensee’s current assumptions.

BSEP 98-0058, Commitment 7 - Residual Heat Removal System Valve 2-E11-FO24B Torque Switch Adjustment

The licensee committed to adjust the torque switch for MOV 2-E11-F024B to reduce excess seating torque. In the current inspection the inspectors reviewed the Motor Operated Valve Trace Review Sheet, Valve 2-E11-F024B, performed 4/24/99, and verified that it documented appropriate adjustment of the torque switch.

BSEP 98-0058, Commitment 9 - Commitment Status Submittal

The licensee committed to provide a full completion status submittal letter by January 31, 2001. In the current inspection, the inspectors verified that the commitment for a final closure letter was tracked by the licensee as Action Request 7576, Generic Letter 89-10 Final Closure Letter, with a due date of January 17, 2001.

c. Conclusions

Based on the commitments completed and the licensee's identification and tracking of the remaining commitment actions, this IFI is closed.

IV. Plant Support

R1 Radiological Protection and Chemistry (RP&C) Controls

R1.1 Radiological Work Controls

a. Inspection Scope (83750, 71750)

The inspectors observed and compared radiation protection (RP) activities against applicable RP program requirements and 10 CFR Part 20. The inspection included reviews of records and procedures, interviews with licensee personnel, and observations of work activities in progress.

b. Observations and Findings

Independent radiation surveys made by the inspectors were in agreement with licensee's radiation survey results. The radiological postings were adequate for areas surveyed. All locked high radiation area doors checked by the inspectors were secured properly.

There was good RP coverage at the main radiological control area entrance, the major job sites, and exit portals. The inspectors observed good interactions and communications between radiation workers and RP personnel.

During the outage the RP staff found a radioactive hot spot approximately 250 rad per hour on contact with a source range monitor guide tube after the cable and detector had been removed. The licensee removed the tube and shielded it until a recovery plan could be developed. The licensee health physics staff carefully developed a plan to cut and capture the portion of the tube having the high radiation to minimize occupational radiation dose. The inspectors observed the process which was performed as planned and with minimized collective dose.

c. Conclusions

Licensee radiation surveys, postings, access controls, and radiological work controls were effective and performed in accordance with regulatory requirements. Good health physics planning and implementation in capturing and encasing a highly radioactive source range monitor guide tube was observed.

R1.2 As Low As Reasonable Achievable (ALARA) Performance

a. Inspection Scope (83750, 71750)

This review was made to assess licensee performance with respect to maintaining collective radiation exposures ALARA during the Unit 1 refueling outage (RFO).

b. Observations and Findings

Licensee goals for the Unit 1 RFO included a collective radiation dose goal less than 225 person-rem. The annual collective dose goal for year 2000 was 350 person-rem. The licensee utilized a systematic process to prepare ALARA plans for specific RFO tasks. Selected ALARA plans were reviewed and the inspectors found the plans included processes and controls to reduce collective occupational radiation exposures. During the inspection, the licensee was meeting collective dose goals for most ALARA plans.

The licensee made good use of video monitoring, radios, and tele-dosimetry to monitor and control activities in high radiation areas. The licensee also cooled the Unit 1 reactor building aiding workers' efficiency and minimizing personnel contamination events.

c. Conclusions

The Brunswick ALARA program was effective in reducing site collective personnel radiation doses.

R5 Staff Training and Qualification in Radiological Protection and Chemistry (RP&C) Controls

R5.1 Health Physics Vendor Radiation Protection Personnel Qualifications (83750)

The inspectors reviewed the resumes of the senior vendor technicians and found the technicians met the licensee's TS requirements.

V. Management Meetings

XI Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on April 10, 2000. The licensee acknowledged the findings presented.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

A. Brittain, Manager Security
K. Crocker, Radiation Control Outage Manager
D. Dicello, Manager Radiation Protection
N. Gannon, Plant General Manager
J. Gawron, Training Manager
W. Dorman, Manager Regulatory Affairs
J. Franke, Manager Brunswick Engineering Support Section
D. Holder, Radiation Control Outage Manager
J. Johnson, Acting Environmental and Radiation Control Manager
J. Keenan, Site Vice President
J. Lyash, Director of Site Operations
W. Noll, Manager Operations
E. O'Neil, Manager Site Support Services
C. Patterson, Manager Nuclear Assessment (Acting)
E. Quidley, Manager Maintenance
S. Tabor, Project Analyst, Regulatory Affairs
H. Wall, Manager Outage and Scheduling

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 57090:	Nondestructive Examination Procedure Radiographic Examination Procedure Review/ Work Observation/ Record Review
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observations
IP 71707:	Plant Operations Program
IP 71750:	Plant Support Activities
IP 73753:	Inservice Inspection - Maintenance
IP 83750:	Occupational Radiation Exposure
IP 92903	Followup - Engineering
IP 93702:	Prompt Onsite Response To Events At Operating Power Reactors

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Closed

50-325(324)/98-03-01	IFI	Completion of MOV Program Followup Items (Section E8.1)

Discussed

None