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

July 27, 2000

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

SUBJECT: BRUNSWICK - NRC INTEGRATED INSPECTION REPORT NOS. 50-325/00-03 AND 50-324/00-03

Dear Mr. Keenan:

On July 1, 2000, the Nuclear Regulatory Commission (NRC) completed an inspection at your Brunswick facility. The enclosed report presents the results of that inspection which were discussed on July 11, 2000, with Mr. J. Lyash and other members of your staff.

The inspection was an examination of 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. Within these areas, the inspection consisted of a selective examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, the NRC identified two issues that were evaluated under the significance determination process and were determined to be of very low safety significance (Green). These issues have been entered into your corrective action program and are discussed in the summary of findings and in the body of the attached inspection report. The two issues were determined to involve violations of NRC requirements, but because of their very low safety significance the violations are not cited. If you contest these non-cited violations, 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 NRC Resident Inspector at Brunswick Nuclear Power Plant; and the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Brian 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

cc w\encls:

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REGION II

Docket Nos:	50-325, 50-324
License Nos:	DPR-71, DPR-62
Report No:	50-325/00-03, 50-324/00-03
Licensee:	Carolina Power & Light (CP&L)
Facility:	Brunswick Steam Electric Plant, Units 1 & 2
Location:	8470 River Road SE Southport, NC 28461
Dates:	April 2 - July 1, 2000
Inspectors:	T. Easlick, Senior Resident Inspector E. Brown, Resident Inspector E. Guthrie, Resident Inspector J. Coley, Reactor Inspector (Section 1R07) F. Wright, Senior Radiation Specialist (Sections 2OS1; 2OS2)
Approved by:	B. Bonser, Chief, Projects Branch 4 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

Brunswick Steam Electric Plant, Units 1 & 2
NRC Inspection Report 50-325/00-03, 50-324/00-03

The report covers a 13-week period of resident inspection. In addition, it includes the results of announced inspections by a regional radiation specialist and a regional reactor inspector. The significance of issues is indicated by their color (green, white, yellow, red) and was determined by the Significance Determination Process (see Attachment).

Cornerstone: Mitigating Systems

- Green. A non-cited violation of the fire protection program was identified for a failure to establish an adequate procedure to demonstrate the operability of the engine driven fire pump (EDFP) 24 volt battery charger and battery. This failure resulted in the inability of the engine driven fire pump to start when called upon to accomplish its fire or risk-related function. The licensee performed satisfactory troubleshooting, timely repair of the damaged battery charger, and replacement of the dedicated fire batteries. The motor driven fire pump and jockey pumps were unavailable for a short time while the EDFP was considered inoperable; therefore, the issue is considered to be of very low safety significance (Section 1R19).

Cornerstone: Occupational Radiation Safety

- Green. A non-cited violation of Technical Specification requirements was identified for the licensee's failure to provide occupational radiation workers with functioning personnel radiation monitoring dosimetry. Technicians entering the Unit 2 drywell on May 6, 2000, were provided electronic dosimeters that were not properly configured to measure the worker's personnel radiation exposure. This issue was characterized as having very low safety significance because the ability to assess dose was not compromised, and no over-exposure occurred (Section 2OS1).

Report Details

Unit 1 began the report period operating at 100 percent rated thermal power (RTP). On June 23, power was reduced to 55 percent RTP for control rod improvements and valve testing. The unit was returned to 100 percent RTP the following day. On June 28, the 1A reactor feedwater pump tripped on low suction pressure and power was reduced to 60 percent RTP. The 1A feedwater pump was restored to operation and the unit was returned to 100 percent RTP on June 29. The unit operated at or near full RTP for the remainder of the inspection period.

Unit 2 began the report period operating at 100 percent RTP. On May 5, power was reduced to 20 percent RTP for deep/shallow control rod exchanges, valve and scram time testing, and oil addition to the reactor recirculation pump motors. Power was restored to 100 percent RTP on May 7. On June 30, reactor power was reduced to 55 percent RTP for control rod improvements and valve testing. The unit was returned to 100 percent RTP the following day. The unit operated at or near full RTP for the remainder of the inspection period.

1. REACTOR SAFETY
Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignment

a. Inspection Scope

The inspectors reviewed plant documents to determine correct system lineup, and observed equipment to verify that the systems were correctly aligned while the other train or system was inoperable or out of service. The inspectors verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact mitigating system availability. The following system equipment alignments were verified:

- Motor Driven Fire Pump
- Units 1 and 2 Residual Heat Removal (RHR) System
- Unit 1 Reactor Core Isolation Cooling (RCIC) System
- Unit 2 High Pressure Coolant Injection (HPCI)System

In addition, the inspectors performed a detailed walkdown to verify that the 2A core spray (CS) system was correctly aligned during a maintenance outage on the 2B CS system. This review included outstanding design issues, maintenance work requests, and temporary modifications.

b. Issues and Findings

There were no findings identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors reviewed current Action Requests (ARs), work orders, and deficiency reports associated with the fire suppression system. The inspectors reviewed the status of ongoing surveillance activities to determine whether they were current to support the operability of the fire protection system. In addition, the inspectors observed the fire protection suppression and detection equipment to determine whether any conditions or deficiencies existed which would impair the operability of that equipment. During this inspection period the resident inspectors toured the following areas important to reactor safety:

- Diesel Building Basement
- Emergency Diesel Generator (DG) Rooms
- Emergency Switchgear Rooms (4 KV)
- Emergency Switchgear Rooms (480 V)
- Unit 1 Reactor Building 20 foot Elevation
- Unit 2 Reactor Building 20 foot Elevation
- Battery Rooms and Cable Spread Areas

b. Issues and Findings

There were no findings identified.

1R07 Heat Sink Performance

a. Inspection Scope

The inspectors reviewed procedures and documentation with the service water system engineer to ensure that heat exchanger deficiencies which could mask or degrade performance were identified. Selected components examined consisted of the residual heat removal heat exchangers (RHR HXs), the DG jacket water coolers, and the RHR and CS room coolers. The inspectors conducted walkdown inspections of the selected components as well as the chlorine injection facility and the service water pump building. Procedures for inspection and cleaning of each of the heat exchangers selected for inspection were reviewed. Inspection records dated from 1991 to the date of the inspection were reviewed for the components selected. Maintenance rule documentation for the service water system and eddy current test data for the RHR HXs was reviewed. Chemical treatment, sampling, and tube leak monitoring documentation was reviewed with chemistry department personnel. Service water system hydraulic performance test data, and the criteria used to determine heat sink heat transfer performance was also reviewed. Cleaning and inspection activities on the Unit 1 reactor building component cooling water 1B heat exchanger were observed. A nuclear assessment performed on the service water system in 1994 and five recent significant

AR's were also reviewed. This review included the following procedures and ARs:

- 0PM-ACU500- Inspection and Cleaning of The RHR/CORE SPRAY Room Aerofin Cooler Air Filters And Coolers, Revision 6

- 0MST-DG500R - Emergency Diesel Generators 24 Month Inspection, Revision 17

- 0MST-STU500 - Service Water Intake Structure Inspection and Cleaning, Revision 3

- 1SPP-MEC506 - Nuclear Service Water Header Inspection, Revision 4

- 0ENP-2704 - Administrative Control Of NRC Generic Letter 89-13 Requirements, Revision 6

- ENP-2705 - Performance Trending Of RHR Heat Exchangers, Revision 0

Corrective action documents reviewed included:

- Nuclear Assurance Section (NAS) Report B-SP-94-03, Service Water Operational Performance

- AR 00-007225
- AR 00-009592
- AR 00-007323
- AR 00-018150
- AR 00-007482
- AR 00-006539

b. Issues and Findings

There were no findings identified.

1R11 Licensed Operator Requalification Program

a. Inspection Scope

The inspectors observed licensed operator performance during simulator training for cycle 2000-02. This included observation of general emergency operating procedure and abnormal operating procedure scenarios to verify that the requalification program for licensed operators ensured safe power plant operation by evaluating how well the individual operators and crews have mastered training objectives. The scenarios tested the operators' ability to respond to a loss of feedwater, a reactor scram, and the emergency removal of a recirculation motor generator set due to high oil temperatures. Additionally, the scenarios included lessons learned from previous plant experiences.

b. Issues and Findings

There were no findings identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

For the equipment issues described in work orders, condition reports, and 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 a(1) or a(2) classification, and the appropriateness of either the associated a(2) performance criteria or the associated a(1) goals and corrective actions:

- ARs 0018573/ 0018789 — DG 4 Aux. Lube Oil Pump Breaker Found In Tripped Condition/ DG 4 Aux. LO Pump Breaker Trip
- AR 00009592 — Service Water pump strainer through wall leaks
- WR/JO 00-AAYD1 — DG 4 Manual Voltage Adjust Rheostat Erratic Operation
- AR 00019917 — Engine Driven Fire Pump FF
- AR 00020008 — Ground On The 1-E11-F048A-MO (Residual Heat Removal Heat Exchanger 1A Bypass Valve)

b. Issues and Findings

There were no findings identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

For the following work weeks, work tickets, or procedures, the inspectors reviewed the effectiveness of risk assessment before maintenance was conducted (planned and emergent), and verified that upon unforseen situations the licensee had taken the necessary steps to plan and control the resultant emergent work activities:

- 00-ADZF1 — Reviewed the emergent work activity associated with the emergency DG 2 which would not load properly during PT-12.2B, "No. 2 Diesel Generator Monthly Load Test," Rev. 67.
- 0PLP-17 — Observed the emergency work associated with the tightening of the connection for the bushing on the 1C main power transformer.

•	00-AAED1	Unit 2 RHR Service Water Dual Train Outage
•	2OP-5	Unit 2 2A Battery Ground Hunting
•	Week 21	Reviewed and observed scheduled work activities during week 21, with major work activities scheduled as DG 1/core spray/standby liquid control.
•	Week 23	Reviewed the planned work for the emergency DG 4 work week outage on June 5, 2000
•	Week 27	Unit 1 HPCI - Nuclear Service Water Pump 2A - Unit 1 Condensate Booster Pump System outages performed at the same time with high incremental conditional core damage probability values.

b. Issues and Findings

There were no findings identified.

1R14 Personnel Performance During Nonroutine Plant Evolutions

.1 Unit 2 Down Power Evolution

a. Inspection Scope

The inspectors reviewed the operating crew's performance in coping with a scheduled Unit 2 downpower to 24 percent RTP. Additionally, the inspectors observed evolution planning meeting and briefings, and reviewed applicable operating procedures.

b. Issues and Findings

There were no findings identified.

.2 Unit 1 Reactor Feedwater Pump Turbine (RFPT) Trip

a. Inspection Scope

Personnel performance was evaluated following a June 26 event in which the Unit 1, 1A RFPT tripped while the unit was operating at 100 percent RTP. Unit 1 reactor plant power was reduced to approximately 60 percent RTP as expected, following a reactor recirculation pump runback to the number 1 limiter. The cause of the 1A RFPT trip was a corroded suction pressure switch, which was replaced and the RFPT subsequently returned to service.

The inspectors reviewed operator logs, plant computer data, and strip charts to determine what had occurred and how the operators responded; determined if operator responses were in accordance with the response required by procedures and training; and evaluated the occurrence and subsequent personnel response using the significance determination process.

b. Issues and Findings

There were no findings identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed selected operability evaluations affecting risk significant mitigating systems, listed below, to assess, as appropriate: (1) the technical adequacy of the evaluations; (2) whether continued system operability was warranted; (3) whether other existing degraded conditions were considered as compensating measures; (4) where compensatory measures were involved, whether the compensatory measures were in place, would work as intended, and were appropriately controlled; (5) where continued operability was considered unjustified, the impact on Technical Specification (TS) limiting conditions for operations (LCOs) and the risk significance in accordance with the SDP. These reviews were performed for the following:

- 1-E41-LSH-N010-1 — RCIC(HPCI) Supply Drain Pot Level Switch, failure of the switch
- 2A-2-125VDC-BATT — 2A-2 250/125VDC Station Battery, battery cell leak
- 2-E51-C002-VAC-PMP — Unit 2 RCIC Barometric Condenser Vacuum Pump, failure of the vacuum pump
- 1-E11-F048A — RHR Heat Exchanger 1A bypass valve, T1 motor lead ground

b. Issues and Findings

There were no findings identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors performed a design, implementation, and testing review of Engineering Service Request (ESR) 00-00131, Replace DG Excitation Potential Transformers, Revision 0. The inspectors verified the permanent modification for all four DGs was in accordance with design requirements and licensing bases. The inspectors also verified that the modifications to be performed would not place the plant in an unsafe condition.

b. Issues and Findings

There were no findings identified.

1R19 Post Maintenance Testing

a. Inspection Scope

For the post-maintenance tests listed below, the inspectors reviewed the test procedure and either witnessed the testing and/or reviewed test records to determine whether the scope of testing adequately verified that the work performed was correctly completed and demonstrated that the affected equipment was capable of performing it's intended function and was operable in accordance with TS:

- 0PT-8.1.4.b, Rev. 40 "RHR Service Water System Operability Test"
- 2MST-RHR27Q, Rev. 8 "RHR Shutdown Cooling Rx Press Inst Chan Cal"
- 0PT-10.1.1, Rev. 78 "RCIC System Operability Test"
- 0PT-34.1.1.0, Rev. 14 "Fire Pump Test (Motor-Driven and Engine-Driven)"
- 0PT-07.2.4a, Rev. 46 "Core Spray System Operability Test-Loop A"

b. Issues and Findings

On May 22 the licensee performed Periodic Test 0PT-34.1.1.0, "Fire Pump Test (Motor-Driven and Engine-Driven)," Rev. 14. During this test, the engine driven fire pump (EDFP) failed to start after six attempts, subsequently the EDFP was declared inoperable. Licensee troubleshooting determined that the EDFP failed to start due to insufficient charge from the two dedicated EDFP battery banks needed to start the EDFP. The insufficient charge was a result of the failure of the battery charger timer. The function of the timer was to alternate the battery charger from one battery bank to the other. As a result of the timer failure the battery charger overcharged one battery bank and failed to charge the other. When the EDFP tried to start both battery banks were not functional. In addition to providing motive force for the fire suppression system, the EDFP is credited in the licensee's probabilistic risk assessment for late inventory makeup for the loss of offsite power accident and as inventory makeup for several internal flooding events.

The inspectors performed a walkdown of the affected equipment, reviewed applicable procedures, work request/job orders, operator's logs, and root cause analysis. The inspectors noted that the dedicated fire pump battery charger had failed previously in April. The inspectors' review included 0PLP-20, "Post Maintenance Testing," Rev. 22, Periodic Test 0PT-34.1.1.0, and recent weekly performances of Maintenance Surveillance Test 0MST-BATT10NA, Rev. 4, "Fire Pump Diesel Starting Batteries," which was used as the battery charger post maintenance test for the April failure. Since the April failure the inspectors identified that the weekly performances of 0MST-BATT10NA were completed satisfactorily and the batteries met the acceptance criteria; however, the batteries were found to be nonfunctional during the testing performed on May 22. Through review of the pertinent documentation and discussions with the licensee the inspectors determined that the surveillance failed to adequately verify battery and battery charger operability. This failure led to the loss of the capability of the 24 volt batteries to start the EDFP.

TS 5.4.1.d requires that written procedures shall be established for the Fire Protection

Program. Plant Program Procedure 0PLP-1.2, “Fire Protection System Operability, Action, and Surveillance Requirements,” Rev. 15, states in section 6.1.3.3 that the EDFP starting 24-volt battery charger shall be demonstrated operable. The licensee failed to establish an adequate procedure to demonstrate the operability of the battery and battery charger when on May 22, the EDFP failed to start after six attempts. This violation of TS 5.4.1.d is being treated as a Non-Cited Violation consistent with Section VI.A.1 of the NRC Enforcement Policy, (NCV 50-325(324)/00-03-01) failure to adequately establish a procedure to demonstrate the operability of the EDFP 24 volt battery charger and battery. This violation is in the licensee’s corrective action program as AR 00-19917.

Significance Determination

The inspectors reviewed this issue with the assistance of an NRC senior reactor analyst using the Significance Determination Process (SDP) and found the significance of this event was minimal because redundant equipment was available for most of the time the EDFP was inoperable. The licensee performed satisfactory troubleshooting, timely repair of the damaged battery charger, and replacement of the dedicated fire batteries. The motor driven fire pump and jockey pumps were unavailable for a short time while the EDFP was considered inoperable; therefore, the issue is considered of very low risk significance and was therefore, characterized as Green by the SDP.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors examined the test procedure and/or witnessed the testing, and reviewed test records against the Updated Final Safety Analysis Report and TS to determine whether the scope of testing adequately demonstrated that the affected equipment was capable of performing it’s intended function and was operable in accordance with TS:

- 1MST-RHR25Q, Rev. 2 — RHR Pump Disch Press ADS Permissive Inst Chan Cal
- 0PT-07.2.4a, Rev. 46 — Core Spray Operability Test- Loop A
- 0MST-ATWS21Q, Rev. 2 — ATWS Reactor Water LL2 Trip Unit Channel Calibration
- 2MST-RPS23Q, Rev. 3 — RPS High Reactor Pressure Trip Unit Channel Calibration
- 0PT-07.2.4b, Rev. 47 — Core Spray Operability Test- Loop B (Inservice Test)

b. Issues and Findings

There were no findings identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation

a. Inspection Scope

The inspectors observed emergency response training drills conducted on June 8 and June 22 to evaluate drill conduct and the adequacy of the licensee's critique of performance utilizing the emergency plan and the plant emergency procedures (PEPs). The drills were conducted using the plant simulator and emergency facilities. The inspectors verified licensee self-assessment of classification, notification, and protective action recommendation development.

b. Issues and Findings

There were no findings identified.

2. RADIATION SAFETY
Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiological Significant Areas

a. Inspection Scope:

The inspectors reviewed licensee TS requirements for entry into areas having general area dose rates greater than 1Rem and licensee procedures for drywell entries while the reactor is at power. The inspectors reviewed records and interviewed personnel making a drywell entry at power on May 6.

b. Issues and Findings:

On the evening of May 5, the licensee sent two health physics technicians (HPTs) into the Unit 2 drywell to check the oxygen levels and make radiation surveys for a planned entry at reduced power level (<25 percent). The purpose for the entry at power was to add oil to the A and B recirculating pumps. The drywell general area radiation dose rates varied from 300 to 2,000 mrem/hr from gamma radiation and 300 to 600 mrem/hr neutron radiation for the areas the workers would pass through and work in.

Three maintenance workers and two HPTs made a second drywell entry to add oil to the recirculating pumps. Each worker was to have a direct reading dosimeter equipped with a vibrating alarm. When only three external vibrating alarm dosimeters could be found the HPTs on shift assembled two other dosimeters with vibrating alarms. They were provided to the maintenance workers. During the second drywell entry one of the HPTs providing job coverage found the maintenance worker he was monitoring had zero dose on his direct-reading dosimeter when there should have been some measurable dose. The HPT told the maintenance worker to exit the drywell immediately. The other maintenance worker also exited the drywell when his dosimetry was also found to not be measuring dose. The two dosimeters the HPTs had

assembled were not measuring the workers dose.

The dosimeters issued to maintenance workers on May 6, were configured to work with an external detector they did not have. When the vibrating alarm was plugged into the dosimeter it recognized the vibrating alarm as an external detector, which disabled the dosimeter. The dosimeter did not have a functioning detector making it impossible for the dosimeters to measure radiation exposure or reach an alarm setpoint.

The inspectors found that the technicians preparing and issuing the dosimeters were not qualified to configure the dosimeter for use with a vibrating alarm; the licensee did not have written procedures to describe the proper dosimetry configuration; and the licensee failed to make corrections to the dosimetry program that would ensure the dosimeters equipped with a vibrating alarm were properly configured following a warning issued by the dosimeter vendor and received by the licensee prior to the May 6 event.

TS 5.7.2 prescribes licensee requirements for personnel entering high radiation areas with dose rates greater than 1.0 rem/hour (at 30 centimeters from the radiation source or from any surface penetrated by the radiation), but less than 500 rads/hour (at 1 meter from the radiation source or from any surface penetrated by the radiation). Section TS 5.7.2.d prescribes acceptable monitoring requirements for personnel entering such an area.

The inspectors found the licensee had met part of the monitoring requirements of TS 5.7.2.d.3 during the drywell entry, in that, the licensee had assigned HPTs to provide surveillance as specified on the RWP. The HPTs were qualified in radiation protection procedures, equipped with a radiation monitoring and indicating device, and were responsible for controlling personnel exposure within the drywell. However, TS 5.7.2.d.3 also requires functioning direct reading dosimetry.

TS 5.7.2.d prescribes requirements for personnel entering high radiation areas. The licensee failed to issue functioning dosimeters, in accordance with the TS requirements for two maintenance workers entering the Unit 2 drywell at power on May 6. This violation of TS 5.7.2.d is being treated as a Non-Cited Violation consistent with Section VI.A.1 of the NRC Enforcement Policy (50-324/00-03-02) failure to issue functioning dosimeters in accordance with TS requirements. This violation is in the licensee's corrective action program as AR 00-19289.

The inspectors screened this finding using the Occupational Radiation Safety SDP and determined that the ability to assess dose was not compromised, and no over-exposure occurred. This issue was determined to have very low safety significance and was therefore characterized as Green by the SDP.

2OS2 As Low As Reasonably Achievable (ALARA) Planning and Controls

a. Inspection Scope:

The inspectors reviewed policies, procedures, and records regarding plant ALARA activities. The inspectors focused on the ALARA program effectiveness during the 1999 refueling outage. Specific program elements reviewed included:

- The plant collective exposure history, current exposure dose trends, annual dose goals, and exposure tracking procedures;
- Source term reduction initiatives and trending data;
- Radiological work planning;
- Licensee outage reports and documentation of significant outage job evaluations and performance;
- Temporary shielding installation and removal, and schedules for scaffold erection and removal; and
- Corrective action program for problem identification and resolution.

b. Issues and Findings:

There were no findings identified.

4 OTHER ACTIVITIES

4OA1 Performance Indicator Verification

a. Inspection Scope

The inspectors reviewed the past Performance Indicator (PI) data reported upon initiation of the Revised Reactor Oversight Program. A sample of ARs, engineering databases and operator's logs were reviewed to validate the previously reported events. The inspectors reviewed the following PIs for the period of second quarter 1999 through second quarter 2000:

- Unplanned Scrams per 7,000 Critical Hours
- Scrams with Loss of Normal Heat Removal
- Unintended Power Reductions per 7,000 Critical Hours

b. Issues and Findings

There were no findings identified.

4OA3 Event Follow-up

a. Inspection Scope

Following a June 26 event in which the Unit 1, 1A RFPT tripped while the unit was operating at 100 percent RTP the inspectors evaluated the event for plant status and mitigating actions. Unit 1 reactor plant power reduced to approximately 60 percent RTP as expected, following a reactor recirculation pump runback to the number 1 limiter caused by the loss of the RFPT. The inspectors evaluated licensee actions following the event.

b. Issues and Findings

There were no findings identified. The inspectors determined the event to be of minimal risk significance, which met the performance indicator threshold as defined in the inspection attachment.

4OA4 Other

(Closed) Licensee Event Report (LER) 50-324/1999-001-00: Reactor Core Isolation Cooling System Inoperable Due to Valve Packing Leak. This LER was a minor issue and was closed. The licensee's corrective actions for this issue were adequate.

(Closed) LER 50-324/1999-002-00: Reactor Trip Due to Turbine Vibration Instrumentation Failure. The inspectors reviewed the circumstances associated with the event and documented the inspection findings in NRC Inspection Report (IR) 50-325(324)/99-03 dated June 7, 1999. The corrective actions for this event, which included the replacement of the failed vibration detector, were adequate.

(Closed) LER 50-324/1999-006-00: Automatic Reactor Shutdown Due to Condenser Low Vacuum Main Turbine Trip. The inspectors reviewed the circumstance associated with the event and documented the inspection findings in NRC IR 50-325(324)/99-05 dated August 27, 1999. As a result of this event, and previously identified concerns with potential circulating water intake pumps (CWIPs) trips due to fouling of the CWIP screens, the licensee has developed a continuous dredging maintenance plan for the intake canal. A multi-disciplinary task force was also established to review operational guidance and develop an operational strategy to mitigate CWIP trips.

4OA5 Performance Indicator Data Collecting and Reporting Process Review

a. Inspection Scope (TI 2515/144)

The inspectors reviewed the licensee's PI data collecting and reporting process to determine whether the data collecting and reporting methods for current PI data were

consistent with the guidance contained in Nuclear Energy Institute (NEI) 99-02, Revision 0, Regulatory Assessment Performance Indicator Guidelines. During this inspection the inspectors reviewed indicator definitions, data reporting elements, calculation methods, and clarifying notes for the following indicators:

- Unplanned Power Changes per 7000 Critical Hours;
- Safety System Unavailability and Safety System Failures;
- Emergency Response organization Drill Participation;
- Occupational Exposure Control Effectiveness; and
- Protected Areas Security Equipment Performance Index.

b. Issues and Findings

At Brunswick the task of monitoring and trending NRC PI data was assigned to the Regulatory Affairs staff. A PI responsibility matrix was established which assigns responsibility for the data collection to an individual along with that persons supervisor and manager. The individual then collects data for a specific PI and submits it to Regulatory Affairs once per month for tracking and trending. A database is used to compile and report the PI data to the NRC. The database provides for a monthly calculation of each PI which, in turn, allows more “real time” performance monitoring and trending. Additionally, the responsible managers review the PI data applicable to their area and presents the results to the Plant Nuclear Safety Committee on a monthly basis. While this process has not been proceduralized, the licensee is developing a corporate procedure to address the process used for acquisition of site-wide PI data.

The inspectors determined that the process for data collecting and reporting was effective. Most plant personnel interviewed during this inspection were knowledgeable of the guidance in NEI 99-02 and were appropriately implementing that guidance. A review of the safety system unavailability data for emergency AC power identified an issue with the monthly load test of the emergency diesel generator (EDG) engines, which were barred during the test. The barring of the engine resulted in that engine being unavailable for approximately 30 minutes during the test. The licensee previously did not add this time to the EDG unavailability because the NEI guidance stated that planned unavailable hours are taken for testing, unless the function can be promptly restored either by an operator in the control room or by a dedicated operator stationed locally for that purpose. The guidance also stated that restoration action must be contained in a written procedure and must be uncomplicated. However, in NEI 99-02, Frequently Asked Questions, ID Question 150, the guidance states that engine barring is more complex than a few single operator actions that the current guidance allows to be excluded. Therefore, engine barring unavailability time should be included in the EDG unavailability. This issue was discussed with the licensee and future calculations of EDG unavailability will include the time for engine barring.

The inspectors reviewed the data collection process for the Physical Protection - Protected Area Security Equipment Index. Discussions with the responsible individual revealed a very conservative approach to determining compensatory hours. Procedural changes had been implemented to ensure that the security log reporting criteria adequately reflected any compensatory hours compiled. Minor issues were noted with the ability to account for multiple members of the security force providing compensatory

coverage concurrently. Additionally no formal means of second verification of the accuracy of the data collected was available.

For the Occupational Radiation Safety - Occupations Exposure Control Effectiveness indicator, the inspectors sampled possible performance events, reviewed applicable procedures, and discussed collection methodology with the responsible individual. The inspectors determined that the data collected was dependent on entry of an event into the Corrective Action Program (CAP). The inspectors noted that the threshold for occupational exposure events as defined in Nuclear Generation Group Procedure CAP-NGGC-0200, Rev. 1, “Corrective Action Program,” did not appear to be sufficient to ensure that all PI occurrences would be entered into the CAP. The information may be handled in some other program not related to the CAP, and as a result, the data used to compile the PI may not contain all of the relevant occurrences.

The licensee has performed two self-assessments of the NRC PI data reporting process in March 2000. One addressed the emergency preparedness cornerstone PI’s and the other addressed the process in its entirety. In their self-assessments, the licensee concluded that, although the process was generally effective, issues were identified involving technical inaccuracies related to historical data reported, which was not consistent with the NEI guidance. None of the inaccuracies significantly affected the overall PIs reported to the NRC. In response to the self-assessments, the licensee initiated 10 corrective actions, ARs to document the issues, weaknesses, and items for management consideration identified in the reports.

4OA6 Management Meetings

Exit Meeting Summary

The inspectors presented the inspection results to Mr. J. Lyash, Director of Site Operations, and other members of licensee management at the conclusion of the inspection on July 11, 2000. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

A. Brittain, Manager Security
N. Gannon, Plant General Manager
C. Patterson, Manager Nuclear Assessment (Acting)
J Gawron, Training Manager
W. Dorman, Manager Regulatory Affairs
J. Keenan, Site Vice President
E. O'Neil, Manager Site Support Services
J. Lyash, Director of Site Operations
J Franke, Manager Brunswick Engineering Support Section
W. Noll, Manager Operations
E. Quidley, Manager Maintenance
H. Wall, Manager Outage and Scheduling

NRC

B. Bonser, Chief, Reactor Projects Branch 4

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Opened and Closed During This Inspection

325(324)/00-03-01	NCV	Failure To Adequately Establish A Procedure To Demonstrate The Operability Of The EDFP 24 Volt Battery Charger And Battery (1R19)
324/00-03-02	NCV	Failure to Issue Functioning Dosimeters In Accordance With TS Requirements (2OS1)

Previous Items Closed

324/1999-001-00	LER	Reactor Core Isolation Cooling System Inoperable Due To Valve Packing Leak (40A4)

324/1999-002-00	LER	Reactor Trip Due To Turbine Vibration Instrumentation Failure (4OA4)
324/1999-006-00	LER	Automatic Reactor Shutdown Due To Condenser Low Vacuum Main Turbine Trip (4OA4)

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for

inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.