UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W., SUITE 2900
ATLANTA, GEORGIA 30323-0199

Report Nos. 50-325/94-04 and 50-324/94-04

Licensee: Carolina Power and Light Company
P. O. Box 1551
Raleigh, NC 27602

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

Facility Name: Brunswick 1 and 2

Inspection Conducted: February 5 - March 4, 1994

Lead Inspector: [signature] 3/22/94
R. L. Prevatte, Senior Resident Inspector Date Signed

Other Inspectors: P. M. Byron, Resident Inspector
M. T. Janus, Resident Inspector
R. Bernhard, Senior Resident Inspector - Grand Gulf
R. Musser, Resident Inspector - Browns Ferry
L. Wert, Senior Resident Inspector - Hatch
C. W. Rapp, Reactor Engineer
R. W. Wright, Project Engineer

Approved By: [signature] 3/22/94
H. O. Christensen, Chief Date Signed
Reactor Projects Section 1A
Division of Reactor Projects

SUMMARY

Scope:

This routine safety inspection by the resident inspectors involved the areas of Unit 1 startup/power ascension, operations, maintenance and surveillance, engineering support, plant support, and other areas. Inspections were conducted during normal working hours, on back shift, deep back shift, holidays, and weekends.

Results:

In the areas inspected, one violation was identified involving the failure to perform adequate post modification and subsequent surveillance testing on the hardened wet well vent installation, paragraph 3.c.

The startup/power ascension plan and its implementation on Unit 1 was identified as a strength, paragraph 2.b.
Unit 1 was restarted on February 1, achieved full power on February 18, and was released for normal power operation on February 23, 1994.

Unit 2 was operated at essentially full power during the reporting period.

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REPORT DETAILS

1. Persons Contacted

Licensee Employees

K. Ahern, Manager - Operations Support and Work Control
R. Anderson, Vice-President - Brunswick Nuclear Project
*G. Barnes, Manager - Operations, Unit 1
M. Bradley, Manager - Brunswick Project Assessment
*J. Cowan, Acting Director - Site Operations
G. Honma, Supervisor - Regulatory Compliance
*N. Gannon, Manager - Maintenance, Unit 1
*R. Grazio, Manager - Brunswick Engineering Support Section
*J. Heffley, Manager - Maintenance, Unit 2
*G. Hicks, Manager - Training
P. Leslie, Manager - Security
*W. Levis, Acting Plant Manager - Unit 1
*R. Lopriore, Manager - Regulatory Affairs
*C. Pardee, Manager - Technical Support
*C. Robertson, Manager - Environmental & Radiological Control
*J. Titrington, Manager - Operations, Unit 2
*C. Warren, Plant Manager - Unit 2
G. Warriner, Manager - Control and Administration
*E. Willett, Manager - Project Management

Other licensee employees contacted included construction craftsmen, engineers, technicians, operators, office personnel and security force members.

*Attended the exit interview.

Acronyms and initialisms used in the report are listed in the last paragraph.

2. Operations

a. Operational Safety Verification (71707)

The inspectors verified that Unit 1 and Unit 2 were operated in compliance with Technical Specifications and other regulatory requirements by direct observations of activities, facility tours, discussions with personnel, reviewing of records and independent verification of safety system status.

The inspectors verified that control room manning requirements of 10 CFR 50.54 and the Technical Specifications were met. Control operator, shift supervisor, clearance, STA, daily/standing instructions and jumper/bypass logs were reviewed to obtain information concerning operating trends and out of service safety systems to ensure that there were no conflicts with Technical Specification Limiting Conditions for Operations. Direct observations of control room panels, instrumentation and recorded traces important to safety were conducted to verify operability and that operating parameters were within Technical Specification limits. The inspectors observed shift turnovers to verify that system status continuity was maintained. The inspectors also verified the status of selected control room annunciators.

Operability of a selected Engineered Safety Feature division was verified weekly by ensuring that: each accessible valve in the flow path was in its correct position; each power supply and breaker was closed for components that must activate upon initiation signal; the RHR subsystem cross-tie valve for each unit was closed with the power removed from the valve operator; there was not leakage of major components; there was proper lubrication and cooling water available; and conditions did not exist which could prevent fulfillment of the system's functional requirements. Instrumentation essential to system actuation or performance was verified operable by observing on-scale indication and proper instrument valve lineup, if accessible.

No violations or deviations were identified. The licensee's performance in this area was satisfactory.

b. Unit 1 Startup and Power Ascension (71710)
Sustained Control Room and Plant Observation (71715)

The licensee developed and implemented a comprehensive startup and power ascension plan for Unit 1 to ensure the unit was returned to service in a safe, controlled, and deliberate manner with all equipment tested and verified to be functioning correctly. The plan consisted of the following major elements:

- a startup organization and staffing
- defined command and control responsibilities
- guidance for resolution of emergent issues
- assessment performance objectives
- power ascension plan schedule
- startup test plan requirements
- system walkdown requirements

A manager was assigned to develop and implement this plan well in advance of its required date. Specific staff personnel were assigned on a full time basis to support the plan. A schedule was developed that provided for a 50 day startup, which included two contingency outages if needed to perform emergent repairs. The schedule included assessment hold points where an evaluation and a deliberate decision must be made by the Site Vice President to continue the startup sequence or shutdown and perform repairs. The decision to continue was based on an assessment of plant operations, plant material condition, personnel performance, organizational responsiveness, schedule adherence, and the functioning of administration and work control processes. Assessment holdpoints were pre-established prior to startup, prior to exceeding 40% power, and prior to returning the unit to normal operation at 100% power.

In addition to the above assessment hold points, four decision points were established (i.e., at cold shutdown, 165 psig, 15% and 60% power) which required an evaluation and determination by the Unit 1 Plant Manager to proceed with the startup. This decision was based on plant operation, material condition as determined by system walkdowns, the completion of all needed maintenance, and the satisfactory completion of scheduled tests.

The above plan was placed in operation in late January prior to Unit 1 restart on February 1. The inspector did a detailed review and evaluation of this plan prior to its implementation. He held discussions with management personnel involved in developing and implementing the plan. The inspector determined that the plan was well developed, had extensive management input, and provided highly effective controls and evaluation processes to ensure a successful startup. The inspector also attended each decision and assessment meeting and found that they were very thorough and detailed. The inspector determined that the licensee's startup and power ascension plan was well planned and effectively implemented.

As a part of NRC action plan for unit restart, the resident staff was augmented to support startup inspection activities. This provided for 24-hour shift coverage with an observer stationed in the control room to monitor the conduct of control room activities and other important startup and test activities. The 24-hour shift coverage started on January 30 and continued until the plant achieved 60% power on February 14. The resident staff, with selected assistance, provided general coverage of plant activities and direct observation of important tests until the unit achieved full power and was released for unrestricted operation on February 23. The following is a listing by power plateau of inspectors' observations including strengths, weaknesses, and equipment problems that were identified during the startup:

Cold Shutdown to Reactor Critical

The preparations for reactor startup were completed and rod pull commenced on January 31, 1994. Control rods were pulled to within two steps of the Estimated Critical Position (ECP) and it became evident that the reactor would not go critical at the calculated ECP. Rods were driven in and nuclear engineering began consultation with the fuels section in the corporate office to determine why the reactor did not go critical at or near the ECP. After extensive communications between the site nuclear engineers and the fuels section, it was determined that the ECP was overly conservative and a new ECP was developed. Rods were pulled again and the reactor achieved criticality at 11:35 p.m., on February 1.

Other than poor communications between the site and the fuels section, no deficiencies were observed.

Reactor Critical to 15% Power

During power ascension from startup to 15% power, testing was conducted on nuclear instrumentation, HPCI, RCIC, the recently installed digital feedwater control system, the reactor water level reference leg backfill modification, the reactor feedpumps, and the EHC system. In addition to the above testing, numerous PMTRs were conducted to verify equipment operability and an entry was made into the drywell to identify any leakage at full system operating pressure. The following problems were identified or occurred during this power plateau:

- air trapped in reactor water level reference leg required venting
- a drywell temperature detector required replacement
- minor weeping past SRV J - this stopped when pressure and temperature increased
- the air operator on a recirculation pump seal staging valve required repacking
- an electrical short in BOP MCC 1 TA required repairs
- several problems involving the overspeed and backup mechanical overspeed devices on the reactor feedwater pump delayed placing these units in operation. Some inadequacies were also identified involving the procedures used to startup and test the reactor feedwater pumps. Vendor assistance was obtained in this area, but the licensee was able to accomplish the required repairs without extensive assistance.

During this period of the startup and power ascension, the licensee's performance was very positive. Evolutions in the control room were performed in a very controlled manner with nuclear safety being the first priority. Prior to the performance of any activity, a pre-job briefing was conducted for all individuals involved in the execution of the task. In general, the briefings were thorough and described the task in detail. Infrequently performed evolutions, such as rod withdrawal and SRV testing were further briefed in accordance with the licensee's procedure, PLP-17, Identification, Development, Review, and Conduct of Infrequently Performed Tests or Evolutions. These briefings conducted by operations management emphasized strict adherence to procedures, the licensee's self-checking STAR technique, and a discussion of problems experienced by CP&L and other licensees performing similar evolutions.

During the startup, the inspector observed the operations shift turnover on a daily basis. During turnover, most critical activities were stopped so that a thorough turnover could be conducted between the on-coming and off-going shifts. The turnovers observed by the inspector were very thorough and detailed. This was particularly the case for turnover at the unit senior reactor operator position. Plant status, work activities, LCOs, past problems, startup schedule, and other matters affecting the unit were discussed in detail.

Equipment performance during the initial phases of the startup for a plant that had not operated for approximately 21 months was good. The digital controls for the Startup Level Control Valve, a newly installed modification, performed well during the startup. However, the licensee did experience some equipment problems. For instance, a control operator experienced difficulty in withdrawing numerous control rods. This was most likely due to venting problems with the control rod drives. Additionally, a water level instrument, N004C, did not reflect the correct water level when compared to other instrumentation. This problem resulted in a delay in the startup process so that the instrument's reference leg could be backfilled and vented. Overall, the plant's equipment operated properly and allowed for safe power ascension.

The operators performance during the startup was good. All observed evolutions were performed in a careful and deliberate manner. Senior reactor operators controlled all evolutions ensuring that the ROs and AOs understood their duties and responsibilities prior to commencing important activities. Problems that arose were quickly handled in a conservative and safe manner. For example, on February 2, when water level instrument N004C was discovered to be outside its acceptable operating band, all power ascension activities were stopped and the proper TS LCO entered. Working as a team, the operators took the correct actions within the required time frame as specified in the plant's technical specifications. Additionally, the control room maintained a professional work atmosphere throughout the startup. Control room access was controlled by the SROs and ROs as specified by plant procedures.

Good command and control was also exhibited during a minor fire which occurred on February 4, at approximately 5:00 a.m., due to a phase to phase short in MCC 1TA (see paragraph 3.a.). Following the initial report, the control room immediately notified and dispatched the fire brigade to the scene. During this time, the use of the plant page was restricted for emergency use only. Operators concentrated on their entry into the required procedures and the assessment of damages and recovery of lost systems. Effects of this fire were felt on both units, as Hydrogen Water Chemistry was lost on Unit 2 as a result of the fire. Unit 2 response and subsequent recovery/stabilization efforts were conducted in a smooth and efficient manner. The overall response to this event was well coordinated and controlled, demonstrating good use of command and control and a familiarity with emergency response actions.

Overall, this portion of the startup was performed in a cautious and deliberate manner. Although meeting their schedule was important, the safe operation of the unit was overwhelmingly the most significant objective during the startup.

15% To 35% Power

The licensee's efforts at this plateau consisted of inerting the drywell testing the main turbine and its associated protective devi synchronization to the grid, performing system walkdowns, feedwater testing, and other performance and operational tests.

The equipment problems experienced at this power level included the main turbine stop and control valves closing prior to achieving rated speed. It was determined that this had been caused by a faulty diode which was replaced. An incorrectly adjusted limit switch on the turbine control valve resulted in an automatic actuation of all four diesel generators while performing turbine overspeed trip testing. A steam leak was identified and repaired in a drain line in the MSIV pit area. All the above items were correctly diagnosed and repaired in a timely manner.

The inspectors continued to identify excellent pre-job briefings and exceptionally good internal and external communications in the control room and between plant operations and other supporting sections. Shift turnover continued to be detailed and professional. All observed testing was performed in a deliberate and controlled manner with good support provided by all units. Management continued to provide good oversite and direction.

One personnel error was identified when a vendor technician inadvertently pushed the wrong button on the digital feedwater control system in the back panel area. This resulted in a rapid change in reactor water level from 187 inches to 173 inches and the system transferred from three element to single element control as designed. The level was restored to normal and all testing was stopped to address this item. It was determined that operations was not aware that the individual was entering data into the system controls. The individual did not believe that his actions would have any effect on system operations. Overall performance at this plateau was good.

35% to 60% Power

This plateau consisted of placing the second feedwater pump in service, performing low power testing, LPRM calibration, performing system walkdown of the MSRs and heater drains, and the completion of performance, maintenance, and surveillance testing. No significant personnel or equipment problems were identified at this plateau. Operations and the support organizations continued to perform well in a controlled and deliberate manner. After the unit achieved 60% power, NRC control room staffing was reduced to the observation of special tests.

60% to Full Power

This plateau consisted of additional testing of the DFW system at 75% power, turbine valve testing at 80% power, followed by a period of fuel preconditioning at 80% power. This preconditioning was accomplished to reduce the potential for damage from any debris that may exist in the RCS and fuel area. After fuel preconditioning, power was reduced to 65% for a rod pattern change and power was then raised to 98% to attain Xenon equilibrium perform core parameter checks, and complete DFW system final acceptance tests. These tests included a reactor feedwater pump trip and a recirculating pump runback from full power.

Prior to performing the final acceptance test, the operators were given specialized training on these transients in the simulator. This training included exercises with and without faults. The inspector observed this training and the plant acceptance testing. The tests were performed satisfactorily on February 22 and 23. After the above testing and an assessment by the unit manager and Site Vice President, the unit was released for unrestricted operations on February 23.

The above startup activities were conducted in accordance with a well organized, planned, and developed startup and power ascension plan. Inspections of equipment and spaces prior to and during the unit startup indicated that significant improvements had been made in the areas of plant cleanliness, preservation, and equipment maintenance and upgrading. Operator and support organizations morale and attitudes appeared to be positive and well focused on the unit restart. Management involvement and oversite had significantly increased and provided very positive results. Considering that the plant had been shutdown for 21 months with a large amount of work performed, the plant restarted and performed well with very few equipment and personnel problems. The startup and power ascension plan, including the performance of the plant staff during the unit 1 restart, is considered a strength.

c. Review of Operations LERs (92700)

(Closed) LER 1-91-27, Two Inoperable Control Rod Accumulators Result in Entry into Technical Specification 3.0.3. This event occurred when a CRD accumulator low nitrogen pressure alarm was received on HCU 46-27 while an operator was recharging the HCU 34-19 accumulator. With two inoperable HCU accumulators, TS 3.0.3 was entered when the Control Room declared the second HCU inoperable. The AO recharging HCU 34-19 was dispatched to verify low nitrogen pressure on the second HCU and to continue recharging 34-19.

HCU 34-19 was recharged and returned to service, thus exiting the TS LCO. The licensee recharged the accumulator for HCU 46-27 and on January 25, 1993, submitted a TS change that would eliminate the reportability of this event.

(Closed) LER 1-91-15, Two Inoperable Control Rod Accumulators Result in Entry Into Technical Specification 3.0.3. This event occurred when CRD accumulator low nitrogen pressure alarm was received on HCU 30-47. Maintenance was in progress on HCU 30-19 to repair a nitrogen leak at the same time. With two inoperable HCU accumulators, TS 3.0.3 was entered. An AO was dispatched to recharge the accumulator for HCU 30-4 The accumulator was recharged and returned to service, thus exiting the TS LCO. The licensee completed the maintenance on HCU 30-19 a day later. As previously indicated, a TS change was submitted on January 25 1993, that will eliminate the reportability of this event.

(Closed) LER 1-93-10, Hourly Fire Watch Technical Specification Surveillance Missed During Radiography. This event occurred when an assigned fire watch was unable to enter an area and perform the required hourly inspection due to radiographic activities. This occurred due to a breakdown in communications between the firewatch, health physics personnel, and radiography. Only one hourly round was missed. The individuals involved in this event were counselled and a faulty public address system phone that was a contributing factor was repaired under WR/JO 93-APLZ1. The inspector verified that the above actions had been completed.

d. Licensee Action on Previous Operations Inspection Findings (92701, 92702)

(Closed) Violation (325,324/93-19-01), Inadequate Corrective Action to Correct Deficiencies in Clearance Implementation, Tagout Audits, and Operator Shift Turnovers. On April 19,1993, clearance 2-93-1094 was hung which required the control switches for Containment Atmospheric Control Valves 2-CAC-V4, V55, V56, and V58 to be in the closed position. On April 21, the inspector found these switches to be in the neutral position. A tagout audit on April 20 and multiple control board walkdowns between April 19 and 21, 1993, failed to identify this discrepancy. This event was similar to that described in Notice of Violation (Violation B) dated August 25, 1992. The licensee responded to the violation in a letter dated June 25, 1993. The licensee's corrective actions were completed on April 1, 1993, and included:

- Counseling of involved individuals
- Senior Operations Management reviewing the event with each shift, re-emphasizing their expectations

- Each shift supervisor administering a control board awareness/walkdown checkout card to each operator assigned to his shift
- Operations management performing semi-weekly assessment/walkdowns of the control boards with the ROs for 10 weeks
- Implementing a new self-checking program called STAR (Stop, Think, Act, and Review)

The inspector reviewed the assessments that had been completed by operations management. He also reviewed the corrective actions for items identified during the assessments. The items identified by operations management were not significant and were similar to those observed by the inspector.

The licensee continually emphasizes the STAR process, and the inspector has observed a significant reduction in the number of personnel related issues. The licensee also continues to give control board walkdowns to the ROs. The inspector found the licensee's corrective actions for these issues to be effective.

(Closed) Violation 325,324/93-52-01, Inadequate Control Room Logs. This violation identified that an operator had failed to log when a control rod was found at position 46 instead of the required position 48. An investigation found that the rod had not been returned to position 48 after the performance of PT.14.1, Control Rod Operability Check, which exercised the rods weekly. This rod remained in the wrong position for 18 hours until identified by an oncoming nuclear engineer. This was identified during a shift turnover, and it appears that both operators thought the other one would make the log entry. The licensee responded to the violation in a letter dated January 28, 1994. The licensee corrective actions were to make a late log entry and counsel the applicable operators and discuss the item with all shift personnel. The inspector verified that these actions had been completed. The inspector also reviewed the licensee's existing guidance on log keeping (OI-71, Operations Shift Logs) and determined that it provides adequate detail and guidance on log keeping.

Violations and deviations were not identified.

3. Maintenance

a. Maintenance Observation (62703)

The inspectors observed maintenance activities, interviewed personnel, and reviewed records to verify that work was conducted in accordance with approved procedures, Technical Specifications, and applicable industry codes and standards. The inspectors also verified that: redundant components were operable; administrative controls were followed; tagouts were adequate; personnel were qualified; correct replacement parts were used; radiological controls were proper; fire protection was adequate; quality control hold points were adequate and observed; adequate post-maintenance testing was performed; and independent verification requirements were implemented. The inspectors independently verified that selected equipment was properly returned to service.

Outstanding work requests were reviewed to ensure that the licensee gave priority to safety-related maintenance. The inspectors observed/reviewed portions of the following maintenance activities:

94-ACTR1	Repair of Annunciator U4-23, Exhaust Hood A Vacuum Low
93-BCPN1	Adjust MGU/MSC overspeed stops, 1A RFPT
94-ACMS1	Repair DG No. 4 jacket water heating pump motor
93-BCPN1	Repair/adjust reactor feedwater pump 1A overspeed trip mechanism
94-ADAM1	Repair leak on B21 F038C steam line drain venturi
94-ACRV1	Repair damaged bus on MCC 1TA
93-IR9001	Lever adjustment for RFPT MGU
94-ADBI1	Troubleshoot spurious turbine trip and repair/ adjust limit switch on Unit 1 stop valve No. 1

MCC 1TA Repairs

Inspection Report 325,324/94-02 identified the occurrence of a short and resulting electrical fire in turbine building MCC-1TA that occurred on February 4. The main feeder breaker automatically tripped and extinguished the fire. The fire and damage to equipment was isolated to one cubicle in the MCC. The damaged area included arc damage to one phase of a 600 ampere vertical bus section adjacent to where the main power feeder cable connects to this bus. The bus loads are all balance of plant loads.

Plant maintenance removed the two cubicle buckets in the A section of the MCC and removed the damaged bus section. They did not have a 600 ampere replacement bus section in stock so they performed an engineering evaluation (EER 94-0038) that allowed them to install a modified 300 ampere bus assembly as a temporary replacement until parts could be obtained from the manufacturer. These actions allowed Unit 1 restart to continue.

The inspector reviewed the above EER and discussed it in detail with the engineers who performed the evaluation. The inspector found this repair to be acceptable until the correct replacement parts could be obtained and an outage of sufficient duration exists to permit replacement. The inspector observed the disassembly and inspection of the failed parts. No deficiencies were identified during the above repair activities.

RFP Overspeed Trip Repairs

The personnel working on this item experienced procedural problems that required a procedure revision. The lead mechanic who generally worked on this equipment was unavailable and it took a significant amount of time for the assigned people to familiarize themselves with the equipment and make the necessary repairs and adjustments. The individuals involved appeared to do an overall effective job and clearly gained confidence on this equipment.

DG No. 4 Jacket Water Pump and Steam Line Drain

The repairs to the DG jacket water heating pump motor and the leaking steam line drain were performed in a timely manner without impact on the unit restart.

No deficiencies were identified on the other observed WR/JOs.

b. Surveillance Observation (61726)

The inspectors witnessed/reviewed portions of the test activities during Unit 1 restart. Through observation, interviews and record review, the inspectors verified that: tests conformed to Technical Specification requirements; administrative controls were followed; personnel were qualified; instrumentation was calibrated; and data was accurate and complete. The inspectors reviewed the test results and ensured that the equipment was correctly returned to service:

OPT-14.3.1	In sequence Critical Shutdown Margin
OPT-5C 2	SRM/IRM Overlap Verification and IRM Range 6 and 7 Continuity Check
OPT-50.12	Measurement of In Sequence Critical Data
1MST-IRM25NA	IRM Range Correlation Adjustment
PM-89-001	Digital Feedwater Testing (SULCV)
OPT-10.1.1A	RCIC Component Test
OPT-10.11.L, 10.12.L, & 10.13.L	RCIC ASD Test
OPT-9.3A	HPCI Component Test
OPT-09.10L	HPCI Component Local and ASSD Test
OPT-9.3	HPCI Operability Test
OSP 93-049	Tune HPCI/RCIC Controllers
PM 89-001	RFPT High Level Trip Test
OCM-TRB521	1st RFPT Overspeed Test
PM 89-001	1st RFPT MGU/MSC Functional Test
OPT-37.2.1 & OPT-37.2.3	1st RFPT PTs
PM 92-152	1A RFPT Logic Redesign Acceptance
PM 89-001	Digital Feedwater Functional Test When 1st RFPT Placed In Service
OCM-TRB521	2nd RFPT Overspeed Test

PM 89-001	2nd RFPT MGU/MSC Functional Test
OPT-37.2.1 & PT-37.2.3	2nd RFPT PTs
PM 92-152	1B RFPT Logic Redesign Acceptance Test
OPT-50.2	IRM/APRM Overlap Verification
OPT-11.1.2	SRV/ADS Test
PM 89-001	Digital Feedwater Functional Test When Control System Placed in Master Automatic
OPT-10.1.1	RCIC 1000 psig Operability Test
OPT-09.2	HPCI 1000 psig Operability Test
OPM-TRB507	HPCI Operational Inspection
1MST-HPCI39R	HPCI Response Test
OPT-01.9E	TIP Axial Alignment
1SP-93-071	Feedwater Valve Inspection
OPT-80.2	Drywell Entry and Class 1 Conditional System Leak Test
OPT-20.3C	Drywell Air Lock Leak Test
ENP-24.15	Full Core TIP Scan Before Exceeding 25% Power
OPT-13.1	Jet Pump Operability
OP-26	Main Turbine Startup Tests
OPT-40.2.11	Generator Voltage Regulator
OPT-01.11	Core Performance Parameter Check
OPT-40.2.6	Main Turbine Overspeed Test
OPT-40.2.8	MSIV Closure Test
OPT-26.8.5, 8.6,8.7, & 8.10	Main Turbine Valve Testing (SV, BPV, CIV, NRV)
PM 89-001	Digital Feedwater Control System Functional Test - When Placed in Three Element Control
OPT-01.11	Core Performance Parameter Check
OPT-14.2.1	Single Rod Scram Insertion Time Test if Required
OPT-1.9D	TIP System Calibration @ <40% Power
OPT-1.9	LPRM Calibration @ <40% Power
OPT-1.8D	Core Thermal Power Calibration @ <40% Power
OPT-50.13	APRM/LPRM Flux Noise Baseline Data @ <40% Power
PM 93-031	RPV Reference Leg Backfill Sensitivity Test
OPT-37.2.2	RFPT 1A and 1B Stop Valve Test
PM 89-001	Recirculating Pump Runback Test @ 45%
PM 89-001	Digital Feedwater Functional Test When Second RFPT Placed in Service
OP-26.8.16	Main Turbine Power/Load Unbalance Test
1MST-RPS28R	MSL Rad Monitor Setpoint at 60% Power
OPM-NE001	LPRM Detector Performance Evaluation
OPT-50.3	TIP Reproductibility and Uncertainty Determination @ 60% Power
OPT-01.9E	Axial Alignment of TIP @ 60% Power

OPT-01.9	LPRM Calibration @ 60% Power
OPT-1.8D	Core Thermal Power Calculation @ 60% Power
OPT-50.14	TIP Tube and LPRM Configuration Verification @ 60% Power
PM 89-001	Digital Feedwater Functional Test at 75% Power
OPT-40.2.5 & 40.2.9	Main Turbine Valve Testing
PM 89-001	Digital Feedwater Functional Test @ 100% Power
1MST APRM11W	APRM CH A, C, and E Channel Functional Test RDS Inputs
2MST RPS 27R	RPS Scram Discharge Volume Hi Water Level Channel Functional Test and Channel Calibration
PM 93-031	RPV Reference Leg Backfill Sensitivity Test - Rated Reactor Pressure.

RPV Reference Leg Backfill

The inspector observed the testing performed in the above RPV reference leg backfill modification (PM 93-031) for Unit 1 during power ascension testing on February 12,1994. The first phase of this test was performed to gather data and determine the sensitivity on the unit's reactor water level instrumentation over variable flow rates. Flow sensitivity tests were performed for each of the seven reference leg condensing pots at 920 psig reactor pressure with flows that ranged from 0.002 to 0.016 gpm (note 0.016 gpm is 200% normal flow). Time history plots of the archived data were recorded by ERFIS. All plant parameters for the injected loops appeared to be relatively constant when observed with increased flows and compared favorably with the non-injected loops resulting in a successful test.

The second phase of the sensitivity testing involved increasing the back flow rate to all seven reference leg condensing pots to 0.016 gpm and observing their level indication sensitivity effects for each of the following reactor perturbations:

- Start the standby CRD pump and stop the operating CRD pump
- Transfer of CRD pump suction filters
- Transfer of CRD pump drive filters
- Transfer of CRD pump suction source to the CST
- Return the CRD suction source to the pretest condition
- Continual withdrawal of a selected control rod (30-03) to position 24
- Continual insertion of control rod (30-03) to position 00
- Notch withdrawal of control rod (30-03) to position 24
- Notch insertion of control rod (30-03) to position 00

The third phase of testing the unit's reactor water level instrumentation sensitivity involved performing the same above listed reactor perturbations for all seven leg condensing pots using a 0.008 gpm (100% normal flow rate) backfill flow rate.

The fourth and final sensitivity testing phase involved isolating level transmitter 1-B21-LT-N027A and monitoring the backfill flow indicator while slowly decreasing the backfill flow rate to 0.008 gpm using the flow metering valve.

Once all testing was complete, the backfill flow was left in the normal operational alignment and the vessel level instrumentation system was turned over to Operations. Observation and review of the sensitivity measurements recorded by this acceptance test determined the Reference Leg Backfill Modification should have no adverse affect on the reactor vessel instrumentation.

The inspector noted that excellent pre-job briefings were conducted for involved personnel prior to the performance of the above remaining tests. These briefings were detailed and covered the tests, anticipated results, and acceptance criteria. Applicable plant and industry experience associated with the test was also discussed. The assignments of test supervisors, coordinators, and specific test personnel enhanced this process and provided more effective control. The inspector noted that support organizations responded in a timely manner to provide assistance when needed. The questioning attitude of test personnel led to the identification and resolution of several problems, such as the need to test the CAC-V216 valve (See paragraph 3.c). The most significant problems encountered during the above test involved the reactor feedwater pumps. The majority of these problems related to poor procedures, workmanship, and inadequate knowledge on the equipment. The maintenance organization was challenged, but was able to resolve these problems with only minor assistance from a vendor. The pre-startup tests performed on HPCI and RCIC using auxiliary steam significantly reduced the problems normally experienced on this equipment during startup.

Digital Feedwater

The testing on the digital feedwater system went well and provided the operators with added assurance of this new system's capability. Overall, the above testing went exceptionally well with significantly less than anticipated problems.

TSC/EOF Diesel Generator

During a routine review of corrective actions identified and committed to during 1993, the licensee identified a failure to schedule and perform preventive maintenance on the TSC/EOF diesel generator as identified in NRC Violation 93-04-03. The violation identified the fact that the TSC/EOF diesel generator did not have a scheduled preventive maintenance program which was contrary to the requirements of plant emergency procedure PEP-04.2, Emergency Facilities and Equipment.

The above routine review identified that all the corrective actions committed to in the Reply to Notice of Violation dated April 16, 1993, were not met. In the Reply to Notice of Violation, the licensee committed to developing, scheduling, and completing semi-annual and eighteen month maintenance prior to August 25, 1993 and 1994, respectively. The licensee's review on February 3, 1994, found that these actions had not been completed. Adverse Condition Report 94-058 was initiated to track this issue to resolution, as well as a root cause investigation to determine why the maintenance and testing had not been performed after procedure development. The inspector will review the results of the root cause analysis when completed. On March 2, 1994, the scheduled preventive maintenance and testing was completed satisfactorily utilizing procedures OPM-ENG-505, Maintenance Instruction for Covington Diesel Generator Model 7123-7305, Rev. 1, and OPM-GEN-008, Covington Diesel Generator Electrical Inspection, Rev.1. This work was scheduled and performed under preventive maintenance routes: 94-J01004, 94-JI3104, and 94-SA4001.

c. Licensee Action on Previous Maintenance Inspection Findings (92701, 92702)

(Closed) URI 325,324/94-02-01, Inadequate Surveillance Procedure. The Unit 2 Hardened Wet Well Vent Plant Modification PM 92-073 was completed in March, 1993. The modification included Hardened Wet Well Vent Outboard Isolation Valve 2-CAC-V216 which is listed in Appendix B of RCI-02.6, Cross Reference to Technical Specifications, as a Primary Containment Isolation System (PCIS) valve. This valve can be operated manually from the RTGB with an override switch or closed automatically by a LOCA signal provided by relay 3B (SK91001-Z-7007) in the Group 6 isolation logic.

The Nuclear Plant Modification Program (NPMP) Procedure, Section 4.3.6.1 requires that tests be included or identified to demonstrate that the changes made by a modification are satisfactorily implemented and to verify compliance with affected required surveillances. Technical Specification 4.6.3.1 requires that each PCIS valve specified in RCI-02.6 be demonstrated operable prior to returning the valve to service. Technical Specification 4.6.3.2 requires that each isolation valve be demonstrated operable at least once per 18 months by verifying that each PCIS valve actuates to its isolation position upon receipt of a containment isolation test signal. The licensee demonstrates this function for the CAC valves by performing 2-MST-CAC-41R, CAC PCIS Groups 2 and 6 Isolation Logic System Functional Test.

The licensee elected not to perform Group 6 isolation logic testing on CAC V216 after installation since it would have caused the other PCIS valves to isolate drywell ventilation. This action would have created a confined space and would have impacted the outage schedule. The licensee only tested CAC-V216 for valve stroke time and the operation of outboard isolation override logic and did not test the operation of the valve using a LOCA test signal. NPMP 4.3.6.2 states that those portions of acceptance tests which cannot be performed until after the unit is returned to service should be identified as startup tests. NPMP 4.3.7 also requires that documents requiring revision prior to operability be identified to support any surveillance and/or startup requirements. The engineer assigned to this project believed that the functional test described above was adequate to demonstrate operability. He therefore determined that 2-MST-CAC-41R, which demonstrates the operability of CAC-V216 was not required to be revised prior to operability. In addition, CAC-V216 was not incorporated into the monthly OPT-4.1.1, Reactor Building Vent Exhaust Monitoring System Functional Test, which would have demonstrated the valve's operability.

The licensee discovered this deficiency on January 31, 1994, when the Unit 1 STA noted that 1-CAC-V216 was not tested during the performance of OPT 4.1.1. Investigation revealed that the same valve for Unit 2 (2-CAC-V216) had not been adequately tested.

The inspector reviewed the modification package and determined that the licensee did not include a test to demonstrate that the CAC-V216 would close upon the actuation of a LOCA logic relay as required by Technical Specification 4.6.3.1. The licensee also failed to include this valve in their surveillance program. This is a Violation of Technical Specification 4.6.3.1 (50-324/94-04-01), Failure to Incorporate CAC-V216 into PCIS Test Procedures. This closes the URI. All corrective actions for this item will be tracked under this Violation.

The licensee documented the above event in ACR 94-052 and reported it to the NRC in LER 2-94-01. The immediate corrective action was to issue and perform a one-time only, temporary revision to OPT 4.1.1 which included valve CAC-V216. This test was performed on January 31, 1994. The valves for both units were tested satisfactorily. The licensee also plans to test these valves during the performance of MST-CAC41R, CAC PCIS Groups 2 and 6 Isolation Logic System Functional Test, which is performed each refueling outage. The licensee stated that OPT-4.1.1 will be revised to test the PCIS logic of these valves monthly.

4. Engineering Support

a. Installation/Testing of Modifications (37828)

The Plant Process Computer Replacement (PPCR) project (PM90-004) transferred the functions currently performed by the Plant Process Computer to a new advanced system with greater hardware and software capabilities, expandability, and reliability. The new system can be more easily maintained and supported. In conjunction with the installation of the new plant process computer system, the Nuclear Fuels services group updated and upgraded the core monitoring software. This modification expanded the capabilities and reliability of the existing system and provided a more efficient and user friendly system for the control room operators.

The PPCR project involved the removal of the existing system consisting of a Honeywell 4010 computer, analog and digital signal I/O cabinets, computer console, alarm typers and assorted printers. The system was replaced with new front end data acquisition equipment, data links, a high speed interface to existing VAX computers, additional VAX systems including CPUs, memory disks, controllers, special purpose interfaces to existing plant data systems, and new operator interface consoles. Associated with the hardware upgrade was an upgrade of the system software. This software upgrade includes new data acquisition and validation capabilities, a new core monitoring software package entitled POWERPLEX and system integration software to coordinate and monitor the entire system.

With the new system, POWERPLEX will be utilized to calculate core power distribution and margins to TS thermal limits. The program is a commercially available product of Siemens Nuclear Power Corporation which has been approved for use by the NRC and is currently in use at seven other BWRs around the country.

The installation of new equipment and new software required that training be conducted for the primary users of the system, control room operators and nuclear engineers. All primary users have been trained on the new system, and many have had actual experience using the system during the outage. Discussions with these individuals determined that the training was adequate to operate the system. Based on discussions with various system users, the upgrade was viewed as a useful improvement, providing increased monitoring capabilities over the existing system.

The inspector reviewed the scope of the project and discussed the various aspects of this modification with the responsible engineers and system users. These discussions included the various types of qualification and verification processes used in completing this modification, capabilities of the system, improvements over the previous system, and adequacy of training and support for use of the new system. The inspector did not identify any deficiencies in these areas. The inspector will review the final acceptance test package to verify no additional problems were identified. The modification completed on Unit 1 is identical to the modification which will be performed on Unit 2 during its upcoming refueling outage starting in March of this year.

Violations and deviations were not identified.

5. Plant Support

a. Radiological Controls (71707)

The inspectors verified that the licensee's HP policies and procedures were followed. This included routine observation of HP practices and a review of area surveys, radiation work permits, posting and instrument calibration. No deficiencies were identified.

b. Security (71707)

The inspectors verified by general observations that: the security organization was properly manned and security personnel were capable of performing their assigned functions; persons and packages were checked prior to entry into the PA; vehicles were properly authorized, searched and escorted within the PA; persons within the PA displayed photo identification badges; personnel in vital areas were authorized; effective compensatory measures were employed when required; and security's response to threats or alarms was adequate. No deficiencies were identified.

c. Review of Plant Support LERs (92700)

(Closed) LER 1-93-07, Sampling of Reactor Vessel Coolant Conductivity Not Performed. On April 22, 1993, while Unit 1 was in refueling with the reactor defueled and the fuel pool gates installed, the licensee terminated sampling of the reactor vessel water inventory and established chemical sampling of the fuel pool inventory. On April 24, 1993, Operations personnel recognized that sampling the fuel pool did not satisfy the intent of the Reactor Coolant System Chemistry Technical Specification. The intent of the requirement was to ensure the integrity of reactor materials which could be compromised by chloride induced stress corrosion cracking. Reactor coolant sampling was re-established on April 24, 1993, approximately 52.5 hours after it had been secured.

Investigations performed by the licensee indicated that the event was caused by a misinterpretation of the sampling requirement by both Operations and E&RC personnel. Further investigation revealed that this misinterpretation had existed since the mid-1980s. The personnel involved failed to recognize that the intent was to protect the reactor materials, not just the fuel.

The licensee reviewed the conductivity and chloride levels of the last sample taken from the reactor vessel, and when proper sampling was re-established, determined that all levels were within TS limits. It was determined that no activities which would have increased these levels occurred during this time period. The conductivity and chloride levels of available sources of water to the vessel were also within TS limits. Based on these reviews, the licensee determined that this event was not safety significant.

In response to the event, the licensee implemented the following corrective actions: re-established reactor vessel coolant sampling; issued a Standing Instruction to ensure consistency in interpretation of the TS sampling requirement; revised the E&RC procedures to ensure future sampling was performed in accordance with the TS; and evaluated the issue for future training for OPs and E&RC personnel. The inspector reviewed these corrective actions and found them adequate to prevent recurrence of this event.

(Closed) LER 1-93-013, Main Stack Wide Range Gas Monitor Failure Results in Group 6 Isolation. This failure occurred due to a blown fuse. This resulted in a Group 6 isolation and all components functioned as designed. The licensee established auxiliary stack sampling within one hour. The licensee replaced the fuse and placed the system back in service under a system monitoring mode for 2 days. They were unable to determine the cause of the blown fuse. After 2 days of monitoring, the system was declared operable. The inspector reviewed licensee logs and verified that backup sampling had been initiated as required. This item has not been a recurring problem on this system and the licensee's corrective action appears to be appropriate for the event.

Violations and deviations were not identified.

6. Other Areas (76000)

a. Meetings with Local Officials (94600)

The Senior Resident Inspector (SRI) met with the Mayor and Commissioners of Kure Beach at a regularly scheduled meeting at 7:30 p.m., on February 15. The SRI made a formal presentation to the Mayor and City Council which included an update on the NRC's organization, mission, and responsibility. A summary of the recent plant history and current status, a brief resume of the assigned Resident Inspectors, and the telephone numbers and addresses of appropriate NRC contacts were provided. The SRI responded to several questions involving the shipment of spent fuel and radioactive waste. He also offered to respond to and/or provide assistance and coordination in answering any future questions or concerns the Mayor or Council members may have involving the NRC or the Brunswick Plant. This meeting concluded the bi-annual meetings with officials of communities in the vicinity of the plant.

b. Nuclear Safety Review Committee (40500)

The February 10, PNSC meeting discussed LER 1-94-02 involving the CBEAF system inoperability. A revision to 0-AP-010, Procedure Use and Adherence; the 1993 calendar year security program review; and a review of 2-SP-93-0073/0074, A & B Loop RHR Chemical Decontamination, which is planned to be done just prior to the Unit 2 refueling outage. The RHR decontamination plan was discussed extensively with numerous questions being asked by the PNSC. The team presenting this item appeared to have done an excellent job in planning the project. Several questions could not be conclusively answered and the project managers were asked to research these issues and respond to the PNSC at a later meeting. The minutes of all other meetings for the month of February were also reviewed. No deficiencies were identified.

Violations and deviations were not identified.

7. Exit Interview (30703)

The inspection scope and findings were summarized on March 4, 1994, with those persons indicated in paragraph 1. The inspectors described the areas inspected and discussed in detail the inspection findings listed below and in the summary. Dissenting comments were not received from the licensee. Proprietary information is not contained in this report.

Item Number	Description/Reference Paragraph
324/94-04-01	Violation: Inadequate post modification/ surveillance test involving valve CAC-V216 paragraph 3.c.

8. Acronyms and Initialisms

AO	Auxiliary Operator
BWR	Boiling Water Reactor
CAC	Containment Atmospheric Control
CPU	Central Processing Unit
CRD	Control Rod Drive
CBEAF	Control Building Emergency Air Filters

DFW	Digital Feedwater
DG	Diesel Generator
E&RC	Environmental & Radiation Control
ECP	Estimated Critical Position
EER	Engineering Evaluation Report
EHC	Electro Hydraulic Control System
ENP	Engineering Procedure
EOF	Emergency Operations Facility
ERFIS	Emergency Response Facility Information System
GE	General Electric Company
HCU	Hydraulic Control Unit
HP	Health Physics
HPCI	High Pressure Coolant Injection
LCO	Limiting Conditions for Operation
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
LPRM	Local Power Range Monitor
MCC	Motor Control Center
MSR	Moisture Separator Reheater
MST	Maintenance Surveillance Test
NPMP	Nuclear Plant Modification Program
NRC	Nuclear Regulatory Commission
OI	Operating Instruction
OP	Operating Procedure
PA	Protected Area
PCIS	Primary Containment Isolation System
PEP	Plant Emergency Procedure
PLP	Plant Procedure
PM	Preventive Maintenance
PM	Plant Modification
PMTR	Post Maintenance Testing Requirements
PNSC	Plant Nuclear Safety Committee
PPCR	Plant Process Computer Replacement
QA	Quality Assurance
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RO	Reactor Operator
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
SRI	Senior Resident Inspector
SRO	Senior Reactor Operator
SRV	Safety Relief Valve
STA	Shift Technical Advisor
STAR	Stop, Think, Act, and Review
TS	Technical Specification
TSC	Technical Support Center
URI	Unresolved Item
WR/JO	Work Request/Job Order