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

Report No. 50-325/88-14 and 50-324/88-14

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

Docket No. 50-325 and 50-324 License No. DPR-71 and DPR-62

Facility Name: Brunswick 1 and 2

Inspection Conducted: March 1 - 31, 1988

Inspector: R L Carroll for 5/5/88
W. H. Ruland Date Signed

Accompanying Personnel: S. M. Shaeffer
R. B. Latta
E. H. Girard

Inspector: William J. Tobin 5/5/88.
(Paragraph 10) W. J. Tobin Date Signed

Approved By: P. E. Fredrickson 5/5/88
P. E. Fredrickson, Section Chief Date Signed
Reactor Projects Section 1A
Division of Reactor Projects

SUMMARY

Scope: This routine safety inspection by the resident inspector and others involved the areas of followup on previous enforcement matters, maintenance observation, surveillance observation, operational safety verification, in-office Licensee Event Report (LER) review, followup on inspector identified and unresolved items, plant modifications, allegation followup, and unusual events.

Results: Three Unit 1 violations were identified: Failure to determine reactor vessel pressure and shell temperature every 30 minutes during plant heatup; failure to verify position of an unlocked manual CAD valve in its correct position every 31 days; and failure to perform a 10 CFR 50.59 evaluation.

A potentially significant safety issue was identified by the licensee regarding operator inattention during shutdown conditions. A licensed operator unknowingly allowed reactor coolant system temperature increase 90 degrees. This item was identified as a licensee-identified violation.

REPORT DETAILS

1. Persons Contacted

Licensee Employees

W. Biggs, Principal Engineer
*E. Bishop, Manager - Operations
*J. Brown, Res. Engineer - Engineering
*S. Callis, Onsite Licensing Engineer - Licensing & Nuclear Fuel
T. Cantebury, Mechanical Maintenance Supervisor (Unit 1)
G. Cheatham, Manager - Environmental & Radiation Control
R. Creech, I&C/Electrical Maintenance Supervisor (Unit 2)
*C. Dietz, General Manager - Brunswick Nuclear Project
W. Dorman, Supervisor - QA
*R. Eckstein, Manager - Technical Support
*K. Enzor, Director - Regulatory Compliance
R. Groover, Manager - Project Construction
W. Hatcher, Supervisor - Security
A. Hegler, Superintendent - Operations
R. Helme, Director - Onsite Nuclear Safety - BSEP
J. Holder, Manager - Outages
*P. Howe, Vice President - Brunswick Nuclear Project
*L. Jones, Director - Quality Assurance (QA)/Quality Control (QC)
*M. Kesmodel, Supervisor - Document Control
R. Kitchen, Mechanical Maintenance Supervisor (Unit 2)
J. Moyer, Manager - Training
G. Oliver, Manager - Site Planning and Control
J. O'Sullivan, Manager - Maintenance
B. Parks, Engineering Supervisor
*R. Poulk, Senior NRC Regulatory Specialist
*D. Queener, Principal Engineer - Environmental & Radiation Control
J. Smith, Manager - Administrative Support
*J. Titrington, Principal Engineer - Operations
V. Wagoner, Director - IPBS/Long Range Planning
*R. Warden, I&C/Electrical Maintenance Supervisor (Unit 1)
D. Warren, Acting Engineering Supervisor
B. Wilson, Engineering Supervisor
*T. Wyllie, Manager - Engineering and Construction

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

*Attended the exit interview

2. Exit Interview (30703)

The inspection scope and findings were summarized on March 31, 1988, with those persons indicated in paragraph 1. The inspectors described the areas inspected and discussed in detail the inspection findings listed below. Dissenting comments were not received from the licensee. Proprietary information is not contained in this report.

Item Number	Description/Reference Paragraph
325/88-14-01	VIOLATION - Missed Surveillance of Reactor Vessel Temperature and Pressure During Inadvertent Heatup (paragraph 8.a). (A licensee identified violation was also reviewed.)
325/88-14-02	VIOLATION - Surveillance Procedure Failed to Check a CAD Valve Position (paragraph 8.c).
325/88-14-03	VIOLATION - Failure to Perform 10 CFR 50.59 Evaluation On Service Water System (paragraph 8.b).
325, 324/88-14-04	IFI - Correction of CAD System Hardware Problems (paragraph 8.c).

Note: Acronyms and abbreviations used in this report are listed in paragraph 12.

3. Followup on Previous Enforcement Matters (92702)

(CLOSED) Violation 325/86-29-01; Improper Temporary Revision to OP-03 When Operating EPA Breakers, response dated January 2, 1987. The inspectors reviewed OP-03, Revision 8 for Unit 1 and OP-03, Revision 19 for Unit 2 and determined that these procedures had been modified to provide for the opening of the EPA breakers prior to opening the MG set output breaker. These revisions adequately address the issue identified in the violation. This item is closed.

(CLOSED) Violation 325/87-31-01 and 324/87-35-01; Inadequate DG Surveillance Procedure - TCV Not in Restoration Lineup, response dated November 25, 1987. The inspectors reviewed PT-12.2A, Revision 32, PT-12.2B, Revision 31, PT-12.2C, Revision 31, and PT-12.2D, Revision 33 and determined that these procedures had been modified to include the subject temperature control valves in the system restoration valve lineup. Additionally, these procedures have been revised to provide operational information regarding normal and abnormal valve operation and required corrective action. The revised PTs adequately address the issue identified in the violation. This item is closed.

4. 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. Numerous maintenance items were reviewed throughout the reporting period. The inspectors observed/reviewed, in detail, those portions of the following maintenance activities:

86-BAFF1	1-CAC-V10 Outboard Drywell Purge Exhaust Valve
88-AIYB1	DG No. 3 Oscillations/Troubleshooting

No significant safety matters, violations, or deviations were identified.

5. Surveillance Observation (61726)

The inspectors observed surveillance testing required by Technical Specifications. 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 independently verified selected test results and proper return of equipment to service.

The inspectors witnessed/reviewed portions of the following test activities:

1MST-APRM12W	APRM Channel B, D, and F Functional Test
1MST-BATT11Q	Batteries, 125 VDC Quarterly Operability Test
1MST-HPCI22M	Steamline Low Pressure Channel Calibration
1MST-HPCI26M	HPCI Suppression Pool High Level Instrument Channel Calibration
2MST-HPCI27M	HPCI and RCIC CST Low Water Level Instrument Channel Calibration

2MST-RPS23R RPS High Reactor Pressure Instrument Channel Calibration

2MST-SEIS21R SMA-3 Seismic Monitoring System Channel Calibration

No significant safety matters, violations, or deviations were identified.

6. 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 and 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 Specifications Limiting Conditions for Operations. Direct observations were conducted of control room panels, instrumentation, and recorder traces important to safety to verify operability and that operating parameters were within Technical Specification limits. The inspectors observed shift turnovers to verify that continuity of system status was maintained. The inspectors 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 an initiation signal; the RHR subsystem cross-tie valve for each unit was closed with the power removed from the valve operator; there was no leakage of major components; there was proper lubrication and cooling water available; and a condition did not exist which might 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.

The inspectors verified that the licensee's health physics policies/procedures were followed. This included observation of HP practices and a review of area surveys, radiation work permits, posting, and instrument calibration.

The inspectors verified 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 protected area; vehicles were properly authorized, searched and escorted within the PA; persons within the PA displayed photo identification badges; personnel in vital areas were authorized; and effective compensatory measures were employed when required.

The inspectors also observed plant housekeeping controls, verified position of certain containment isolation valves, checked a clearance, and verified the operability of onsite and offsite emergency power sources.

No significant safety matters, violations, or deviations were identified.

7. In Office Licensee Event Report Review (90712)

The listed LER was reviewed to verify that the information provided met NRC reporting requirements. The verification included adequacy of event description and corrective action taken or planned, existance of potential generic problems and the relative safety significance of the event.

(CLOSED) LER 2-88-03; Auto-Isolation of Reactor Water Cleanup System Inlet Outboard Isolation Valve 2-G31-F004 With Reactor Defueled. The inspector reviewed the event and had no questions. This item is closed.

No significant safety matters, violations, or deviations were identified.

8. Followup on Inspector Identified and Unresolved Items (92701)

a. (CLOSED) Unresolved Item 325/88-01-03; Inadvertent Heatup During Cold Shutdown. The inspector reviewed the OER which described the event and documented the licensee's root cause determination and corrective actions. The licensee found two causes for the event: (1) operator inattentiveness; (2) the SRO directed the control operator to throttle the heat exchanger outlet valve, 1-E11-F003B, to control temperature. This valve has a key lock switch on the main control board that only has open and closed positions. The operator had to place the switch in a "dead" position to throttle the valve. This action, as well as throttling the F003B valve, was not authorized by the system operating procedure.

The actions initiated by the licensee to prevent recurrence included disciplinary action for the operators involved, on-shift training, evaluation of switch replacement for the E11-F003B valve, development of a routine pre-outage training package, proceduralization of key parameter monitoring frequency, and evaluation of the need for appropriate alarms during shutdown conditions.

The inspector reviewed real time training package 88-1-8 prepared for the event and found it acceptable. The licensee identified the failure to follow the RHR operating procedure and met the requirements of 10 CFR 2 Appendix C, for licensee identified violations. Therefore, no violation will be issued for the failure to follow procedure. However, during the heatup, the licensee failed to determine that the reactor vessel pressure and shell temperature were within limits every 30 minutes per TS 4.4.6.1.1. Since the licensee failed to identify the missed surveillance, a Violation is being issued on this particular item: Missed Surveillance of Reactor Vessel Temperature and Pressure During Inadvertent Heatup (325/88-14-01).

b. (OPEN) Unresolved Item 325/88-05-01 and 324/88-05-01; Service Water System Operating Mode Concerns. Further inspector review revealed that no 10 CFR 50.59 review was performed as a result of the problem with the V106 valve prior to Unit 1 start-up on February 20, 1988. FSAR table 9.2.1-1, Service Water Flow Distribution - One Reactor Plant, lists Reactor Building CCW Heat Exchanger flow rate from the nuclear service water header during the first 10 minutes following a Loss of Coolant Accident as zero gallons per minute. Based on the information that the licensee had received regarding how the service water system could operate during the above time, it was possible with V106 failed open, that flow through the RB CCW HX would be greater than zero gallons per minute. Although no mechanical or electrical components were modified, this information did constitute a change to the Service Water System in that failure of the V106 valve would cause the system to not perform as described in the FSAR. This change necessitated a 10 CFR 50.59 safety evaluation, prior to restart of the unit, to determine whether an unreviewed safety question existed. The licensee first became aware of this change in the facility in February 1988. A 10 CFR 50.59 evaluation was not performed until March 22, 1988, after being identified by the inspection in February 1988. Failure to perform a written 10 CFR 50.59 safety evaluation is a Violation: Failure to Perform 10 CFR 50.59 Evaluation On Service Water System (325/88-14-03).

The licensee had reviewed the issue at PNSC meeting 88-018 held on February 12, 1988, and concluded that an operability concern did not exist. Action items were established to resolve the issue permanently. After questions by the inspector, the licensee completed a JCO with a 10 CFR 50.59 evaluation (EER-88-167) for the SW V106 issue on March 23, 1988. The EER took credit for the licensee's interim action of limiting RB CCW SW flow to 5,000 GPM. Based on the SW pump curve, with SW flow assumed at 8,600 GPM, the total dynamic head of the pump would be about 108 feet, equivalent to 48 PSIG. This is above the 40 PSIG low header pressure autostart signal for another NSW pump. This portion of the item remains unresolved pending review by the licensee and inspector of the SW design data and a permanent fix, if required.

c. (CLOSED) Unresolved Item 325/88-05-05 and 324/88-05-05; CAD System Discrepancies. The associated issues were identified during a CAD system walkdown, and are addressed below.

The inspector found valve CAC-V168 unlocked in the open (required) position. All unlocked valves in the system's flow path shall be demonstrated to be operable at least once per 31 days by verifying valve position. Failure to observe the valve position is a violation of TS 4.6.6.2.a.2. The missed surveillance was due to an inadequate procedure, in that valve CAC-V168 does not appear on PT-16.1, Rev. 12, Section 7.1.3, System Operability Verification Checklist. This valve is a flow path valve in that closure of the valve could prevent the system from injecting nitrogen into the drywell or torus after a single valve failure. The inspector reviewed the history of

all periodic test procedures associated with the CAD system for accuracy and timeliness. No other surveillance problems were noted. This is a Violation: Surveillance Procedure Failed to Check a CAD Valve Position, (325/88-14-02).

On February 24, 1988, the inspector observed that the loop A and B CAD vessel pressure indicators, PI-2703 and PI-2704, exceeded the maximum allowable working pressure of the CAD vessel which is 100 PSIG. The pressure indicators were above their maximum gauge graduations (greater than 100 PSIG). The process requirement for the CAD system is between 80 and 90 PSIG. The CAD vessel had exceeded these values due to the evaporation of liquid nitrogen within the vessel. The design of the system incorporates three separate pressure relief mechanisms: (1) two pressure control valves (PCV-2706 and 2707) with build-up coils designed to maintain 85 psig in the CAD tank; (2) PCV-2705 relieves pressure to atmosphere at 90 psig; and (3) pressure safety valve PS-V3 relieves to atmosphere at 100 psig. There was no evidence that any of these functions were operating within their setpoints. The licensee's engineering group has recommended that the relief setpoints be calibrated, and if not within tolerance, the valves should be rebuilt or exchanged.

The licensee showed the inspector the vendor's hydrostatic test results. The test was performed at 176 PSIG. Thus, no safety concern exists regarding the CAD tank integrity. Since the licensee had work requests written on the above problems and the safety significance is small, no violation occurred. The CAD system, although addressed in Technical Specifications, is not necessary to prevent any explosive mixture in a post-accident containment environment. This was shown by generic analysis done by BWROGs for the licensee's hydrogen recombiner rule response. The CAD system would only be needed post-LOCA if the containment had not been inerted. This occurs only during reactor startups and shutdowns.

Minor support problems were identified on the liquid nitrogen lines surrounding the CAD vessel loop B end. Apparently the icing of the nitrogen lines has caused two wood block supports to deteriorate and become non-functional. The inspector observed that another support was left disconnected near valve BYCV-2716. The licensee has issued work requests to correct the deficiencies. The licensee has also been informed of various housekeeping items such as ground cables draped over nitrogen lines and debris in the control cabinets.

Due to the extent of the deficiences, the inspector will followup on the licensee's corrective actions. This is an Inspector Followup Item (Inspection of CAD System Hardware Problems (325/88-14-04 and 324/88-14-04).

d. (CLOSED) Inspector Followup Item 325/86-24-04 and 324/86-25-04; OP-19 Revisions for HPCI CST Recirculation Mode. The licensee performed SP-86-080, HPCI System Operability Test, on October 9, 1986. The test checked the operation of the HPCI turbine control system during three conditions: operating in CST full flow test mode; only on the minimum flow valve; and a dual path mode. Problems with the governor control were noted during certain step changes in the flow demand signal made by the operator. The turbine RPM was hunting with 1,000 RPM swings with a reduced flow demand signal of 3,000 GPM. A portion of the governor, designated EGM, was recalibrated and SP-86-080 re-performed with no anomalies noted. On November 8, 1986, the licensee replaced, under DR-86-0140, the EGM with a newer design, further improving the control system.

The inspector reviewed SP-86-080, interviewed plant personnel, and reviewed the current revision (13) of 1-OP-19, HPCI System Operations Procedure. The licensee has added sections to describe the reactor pressure control mode of the HPCI system. Also, while switching modes, the procedure requires the operator to take the HPCI flow controller to manual. Based on the inspector's review of the licensee's actions, this item is closed.

e. (CLOSED) Inspector Followup Item 325/87-42-08 and 324/87-43-08; Submission of TS Amendment Request for RWCU Isolation Response Time. The licensee submitted the request for TS amendment on February 29, 1988. This item is closed.

f. (CLOSED) Inspector Followup Item 325/88-05-03 and 324/88-05-03; Sand Introduced Into Torus/Vacuum Breakers From Failed Inerting Line. The inspectors reviewed the circumstances of the line break and the adequacy of the corrective actions taken by the licensee. The review was conducted through visual observation of the vacuum breakers and the replacement inerting line, discussions with the Mechanical Discipline Group Project Engineer, and review of the following Work Request and LLRT records:

Unit 1 Work Requests

WR/JO 88-AELT1
WR/JO 88-AEIW1
WR/JO 88-AEIU1

Unit 2 Work Requests

WR/JO 88-AENW1
WR/JO 88-AFAC1
WR/JO 88-AAUH1
WR/JO 88-AAUI1

Unit 1 LLRTs

PT-20.3.71 performed 1/10/88 on V17
PT-20.3.70 performed 1/10/88 on V16
PT-20.3.71 performed 2/10/88 on V17
PT-20.3.70 performed 2/08/88 on V16

Unit 2 LLRTs

PT-20.3.71 performed 2/11/88 on V17
PT-20.3.70 performed 2/11/88 on V16

Relevant information and findings obtained by the inspectors:

(1) The inerting line failure was apparently the result of excessive cooling and resultant stress in the line. The excessive cooling was due to the introduction of insufficiently heated nitrogen from a malfunctioning vaporizer.

(2) Sand did not reach the Unit 2 vacuum breakers, but rather was found on the inboard side of closed butterfly valves (V16 and V17) located inboard of the vacuum breakers. Also, the sand probably did not contribute to the LLRT failures. The butterfly valves are normally closed and sand could not easily get onto the seals. The project engineer stated that the butterfly valves had a past history of excessive leakage and that they had actually leaked less than in past tests. As of March 4, maintenance on one valve had resulted in acceptable leakage and the other showed leakage just in excess of the specified limit. The identical Unit 1 valves passed the LLRTs and required no repair.

(3) The inerting line failure did not occur on January 4, as initially thought. In order for the sand to have reached Unit 2 valves V16 and V17, the failure would have had to occur on or before the Unit 2 shutdown (about January 1, 1988).

(4) An attempt to repair the underground inerting line proved unsuccessful. When an identical break area was repaired the line continued to exhibit leakage; indicating an additional break or breaks. Most of the underground line has now been bypassed. A temporary replacement line installed above ground is now being used. (This is a non-safety-related line.)

(5) The sand that entered the Containment Atmosphere Control System through the break in the inerting line was found at the following locations:

- Almost 2 quarts in the Unit 1 drywell near the 18-inch inerting line opening (underground line was 8-inch size).

- Blown into torus of Unit 1 (from 20-inch line).
- None found in drywell or torus of Unit 2.
- Approximately 2 cups of sand/water slurry was found in the Unit 2 pipe between V16 and V17.
- Approximately 2 gallons of slurry was found between Unit 1 valves V16 and V17.
- About a handful of dry sand near V5 (downstream side) on Unit 1.

The inspectors had no further questions. This item is closed.

g. (OPEN) Inspector Followup Item 325/88-05-04 and 324/88-05-04, Hydrogen Leak in Turbine Building Pipe Tunnel. The inspectors reviewed Operating Experience Report No. 88-024 and conducted an inspection of the rad-waste pipe tunnel and the electrical tunnel located above the pipe tunnel. The inspectors determined that the pre-installation pressure test requirements had been modified to preclude pressure testing of valves on their back seat, that temporary hydrogen monitors will be installed above the subject hydrogen valves per SP-88-01 (not yet accomplished as of March 29), and that routine system checks for hydrogen leaks were being conducted.

The potential for hydrogen intrusion into safety-related equipment areas does not represent a hazard in the pipe tunnel as currently configured. This item remains open pending the inspector's review of other possible hydrogen interactions with safety-related equipment.

One potentially significant safety matter regarding operator inattention during plant shutdown conditions was identified. Three violations, and no deviations were identified.

9. Plant Modifications - Unit 2 (37700)

The inspector reviewed PM-87-128, Weld Overlay Repair, Field Revisions 30 and 31, Repair of CRD Line Leaks, to verify compliance with ENP-03, Rev. 35, Plant Modification Procedure. The inspector observed the pipe replacement and repair, which used cryogenic couplings to connect the new pipe. The inspector also reviewed purchase orders and specifications related to the cryogenic fittings as well as ALARA pre-job reviews conducted for the job. Personnel interviews of selected engineering ALARA personnel were also conducted.

The licensee replaced the CRD lines after leaks were found on insert and withdraw lines inside containment during the reactor coolant system hydrostatic test. Small weeping leaks were found on 5 lines on March 19, 1988. Additional CRD line leaks were found on subsequent inspections; all leaks were repaired. A brown discoloration on the lines was found, by CP&L's laboratory, to contain chlorides and sulfur. Two regional inspectors reviewed the licensee's metallurgical investigation at the lab (see report 325,324/88-17). The inspector will continue to follow the licensee's resolution as part of the LER closeout.

No violations or deviations were identified.

10. Allegation No. RII-88-A-8011

This allegation was received by the Senior Resident Inspector on February 24, 1988, and later, on February 29, was repeated by the alleger to the Region II Allegation Coordinator in Atlanta, Georgia. On March 23, an inspector conducted an announced inspection of the allegation which involved a Shift Operations Supervisor allegedly not being fit for duty. The inspector reviewed several pertinent documents to include the Quality Check Report No. B-88-01-10/QCR No. 14221, and met with the following individuals.

C. Dietz - Plant General Manager
A. Bishop - Plant Operations Manager
S. Choplin - Director of Personnel Relations
E. Enzor - Director of Regulatory Compliance
W. Ruland - Senior USNRC Resident Inspector

There is no USNRC regulation governing Fitness for Duty. However, the licensee's policy, Drug and Alcohol Abuse Statement of Practice, states... "The use of alcoholic beverages by an employee on or away from Company property that may adversely affect the employee's job performance, or that may reflect unfavorably upon public or governmental confidence in the manner in which the Company carries out its responsibilities may result in disciplinary action, including possible termination".

The inspector learned that a similiar allegation had been anonymously telephoned to the licensee's Quality Check Program during the evening hours of January 25. Within a few hours after receipt, the Plant General Manager instructed his operational staff to conduct an immediate investigation which included interviews of operation supervisors who had close contact with the Shift Operations Supervisor during the shift in question. The licensee concluded the allegation was without merit.

Based upon interviews and a review of documentation the inspector was unable to substantiate the allegation and, given the absence of an NRC regulation governing Fitness for Duty, found the licensee to have been responsive to this allegation.

No significant safety matters, violations, or deviations were identified.

11. Unusual Event/Fire (93702)

The licensee declared an Unusual Event for two minutes on March 9, 1988, at 2:40 p.m., after a fire could not be confirmed out within 10 minutes. The drywell chiller power cables (non-safety) on the Radwaste Building roof were reported on fire at 2:27 p.m. The cable insulation had chafed on the roof edge. The fire was announced at 2:28 p.m., and the fire brigade was dispatched. The cables were de-energized at 2:31 p.m., and the fire reported out at 2:36 p.m., by an I&C technician. Since the shift fire brigade commander did not report the fire out within 10 minutes, the licensee declared an Unusual Event per Emergency Procedure PEP-2.1, Revision 21, Initial Actions, which requires the declaration of an Unusual Event with a fire within the protected area lasting more than ten minutes. The shift fire commander reported the fire out at 2:42 p.m. No safety-related equipment was affected by the fire.

Based on interviews with plant personnel, record and log review, the inspector concluded that the licensee complied with their emergency procedures.

No significant safety matters, violations, or deviations were identified.

12. List of Abbreviations for Unit 1 and 2

ALARA	As Low As Reasonably Achievable
AO	Auxiliary Operator
APRM	Average Power Range Monitor
BWROGS	Boiling Water Reactor Owners Group
BSEP	Brunswick Steam Electric Plant
CAC	Containment Atmospheric Control
CAD	Containment Atmospheric Dilution
CCW	Closed Cooling Water
CRD	Control Rod Drive
CST	Condensate Storage Tank
DC	Direct Current
DG	Diesel Generator
DR	Direct Replacement
EER	Engineering Evaluation Report
ENP	Engineering Procedure
EPA	Electrical Protection Assembly
ESF	Engineered Safety Feature
F	Degrees Fahrenheit
FSAR	Final Safety Analysis Report
GPM	Gallons Per Minute
HP	Health Physics
HPCI	High Pressure Coolant Injection
HX	Heat Exchanger
I&C	Instrumentation and Control
IE	NRC Office of Inspection and Enforcement
IFI	Inspector Followup Item
JCO	Justification for Continued Operation
LER	Licensee Event Report

LLRT	Local Leak Rate Test
LOCA	Loss of Coolant Accident
MG	Motor Generator
MST	Maintenance Surveillance Test
NRC	Nuclear Regulatory Commission
NSW	Nuclear Service Water
OER	Operating Experience Report
OP	Operating Procedure
PA	Protected Area
PI	Pressure Indicator
PM	Plant Modification
PNSC	Plant Nuclear Safety Committee
PSIG	Pounds per Square Inch Gauge
PT	Procedure Test
QA	Quality Assurance
QC	Quality Control
RB	Reactor Building
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RPM	Revolutions Per Minute
RPS	Reactor Protection System
RWCU	Reactor Water Cleanup
SP	Special Procedure
SRO	Senior Reactor Operator
STA	Shift Technical Advisor
SW	Service Water
TCV	Thermostatic Control Valve
TS	Technical Specification
URI	Unresolved Item
USNRC	United States Nuclear Regulatory Commission
V	Volt
VIO	Violation
WR/JO	Work Request/Job Order