UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
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ATLANTA, GEORGIA 30303-8931

October 27, 2000

EA 00-234

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 STEAM ELECTRIC PLANT- NRC INTEGRATED INSPECTION REPORT 50-325/00-04 AND 50-324/00-04

Dear Mr. Keenan:

On September 30, 2000, the NRC completed an inspection at your Brunswick facility. The enclosed report documents the inspection findings which were discussed on October 11, 2000, with you and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, the inspectors identified an issue of very low safety significance (Green). This issue as well as an issue evaluated outside the Significance Determination Process were determined to involve violations of NRC requirements. However, because of their very low safety significance and because they have been entered into your corrective action program, the NRC is treating these issues as Non-cited violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny these Non-cited violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Brunswick Steam Electric Plant.

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

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REGION II

Docket Nos:	50-325, 50-324
License Nos:	DPR-71, DPR-62
Report No:	50-325/00-04, 50-324/00-04
Licensee:	Carolina Power & Light (CP&L)
Facility:	Brunswick Steam Electric Plant, Units 1 & 2
Location:	8470 River Road SE Southport, NC 28461
Dates:	July 2 - September 30, 2000
Inspectors:	T. Easlick, Senior Resident Inspector E. Brown, Resident Inspector E. Guthrie, Resident Inspector D. Thompson, Reactor Inspector (3PP1; 3PP2)
Approved by:	B. Bonser, Chief, Projects Branch 4 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS
Brunswick Steam Electric Plant, Unit 1 & 2
NRC Inspection Report 50-325/00-04, 50-324/00-04

IR 05000325-00-04, IR 05000324-00-04, on 07/02-09/30/2000, Carolina Power & Light, Brunswick Steam Electric Plant, Units 1 & 2. The following are areas where findings were identified: flood protection, and maintenance rule implementation.

The report covers a 13-week period of resident inspection. In addition, it includes the results of an announced inspection by a regional safeguards inspector. The inspection identified one Green finding and one no color finding. Both of the findings were Non-cited Violations. The significance of the issues is indicated by their color (green, white, yellow, red) and was determined by the Significance Determination Process (SDP) found in Inspection Manual Chapter 0609 (See Attachment). Findings for which the SDP does not apply are indicated by "no color" or by the severity level of the applicable violation.

Cornerstone: Mitigating Systems

- Green. A Non-cited Violation (NCV) was identified for the failure to promptly identify and correct conditions adverse to quality involving 57 underground safety-related manholes subject to flooding and containing safety-related alternating current and direct current cables. This was determined to be of very low safety significance because no operability problems on safety-related equipment were identified from an engineering review of the deficiencies (1R06).

- No Color. An NCV was identified for the failure to adequately monitor system unavailability hours and take appropriate corrective actions, when the 1A safety-related battery exceeded the licensee established goal for unavailability. This was an isolated failure which did not result in any unidentified equipment failures and was dispositioned outside the SDP as a no color NCV (1R12).

Report Details

Unit 1 began the report period operating at 100 percent rated thermal power (RTP) and operated at or near full RTP for the inspection period.

Unit 2 began the report period operating at 100 percent RTP. On July 16, power was reduced to 84 percent RTP for removal of the 4A and 5A feedwater heaters from service. Power was restored to 100 percent RTP on July 17. On July 18, power was reduced to 80 percent RTP to return the feedwater heaters to service and was returned to 100 percent RTP the following day. On September 17, power was reduced to 56 percent RTP for special backwashing of the 2B-N circulating water system debris filters and was returned to 100 percent RTP the following day. On September 22, the unit scrammed due to a fire on the 2B main transformer. The unit was returned to 100 percent RTP by the end of the inspection period.

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

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors reviewed the licensee's preparations for severe weather as described in Administrative Instruction 0AI-68, "Brunswick Nuclear Plant Response to Severe Weather Warnings," Revision (Rev) 20 and Abnormal Operating Procedure 0AOP-13, "Severe Weather," Rev 28. The review verified that selected risk significant systems would remain functional when challenged by adverse weather, that the procedures would require system readiness and adequate staffing, and that operator actions required for those systems selected could be accomplished during severe weather. The systems selected for this review were:

- Residual Heat Removal (RHR) System
- Service Water (SW) System

b. Issues and Findings

No findings of significance were identified.

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 using the associated operating procedures:

- Unit 1 Reactor Core Isolation Cooling System (RCIC)
  - Operating Procedure 1OP-16, "Reactor Core isolation Cooling System," Rev 46

- Unit 2 High Pressure Coolant Injection (HPCI)
  - Operating Procedure 2OP-19, "High Pressure Coolant Injection System," Rev 93

- Unit 1 Residual Heat Removal Loop A
  - Operating Procedure 1OP-17, "Residual Heat Removal System," Rev 70

- Diesel Generators (DGs) 1, 3, and 4
  - Operating Procedure 0OP-39, "Diesel Generator Operating Procedure," Rev 88

In addition, the inspectors performed a detailed walkdown, of the Unit 2 Standby Liquid Control (SLC) system, to verify that the system was correctly aligned, and labeled. The power sources and support systems were verified to be available. Review of this system included outstanding design issues, maintenance work requests, and temporary modifications. The following documents were reviewed:

- Operating Procedure 2OP-05, "SLC System," Rev 47
- Piping and Instrumentation Diagram (P&ID) D-02547, Sheet (Sht) 36, Unit 2 Reactor Building Standby Liquid Control System Piping Diagram

b. Issues and Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors reviewed current Action Requests (ARs), work orders, and impairments associated with the fire suppression system. The inspectors reviewed the status of on-going 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 inspectors toured the following areas important to reactor safety and reviewed the associated documents:

- Unit 2 Reactor Building (9' and -17' elevations)
  - Plant Program Procedure 0PLP-01.2, “Fire Protection System Operability, Action, and Surveillance Requirements,” Rev 15
  - Fire Protection Procedure 0FPP-005, “Fire Watch Program,” Rev 19

- Unit 1 Reactor Building (-17' elevation)
  - Prefire Plan, 1PFP-RB, “Reactor Building Prefire Plans,” Rev 81

- Units 1 and 2 Cable Spreading Room
  - Prefire Plan, 0PFP-CB, “Control Building Prefire Plans,” Rev 58

b. Issues and Findings

No findings of significance were identified.

1R06 Flood Protection Measures

.1 Electrical Site Manhole Degradation and Corrective Action

a. Inspection Scope

The inspectors conducted a design review of flood protection measures and observed the licensee’s equipment and mitigation plans to verify that they were consistent with design requirements. The inspectors concentrated inspection efforts on sealing of equipment and electrical conduit below the floodline, holes and unsealed penetrations in walls between flood areas, and operable sump pumps and level alarm circuits. The inspectors inspected underground safety-related manholes subject to flooding, which contained safety-related alternating current (AC) and direct current (DC) cables. The inspectors reviewed and observed the condition and adequacy of sump pumps, operable level alarm circuits, cables/splices qualified for submergence, and adequate drainage from manholes.

b. Issues and Findings

The inspectors found that the licensee had scoped 57 site manholes including the underground ductbank and cable conduits in the Maintenance Rule (MR). The inspectors concentrated inspection efforts on those manholes. The inspectors found that the licensee had classified the entire manhole system as MR a(1) on February 1, 1999. A significant root cause investigation, which was conducted as part of the MR a(1) classification, was completed on March 1, 1999. The inspectors determined that a manhole designated SY2 was the manhole, inspected on January 21, 1999, that caused the licensee to place the entire manhole system in MR a(1) status, based on the condition of the SY2 manhole and the expectation that similar conditions would be found in the rest of the site manholes. The MR a(1) classification status was based on the identified adverse conditions in the SY2 manhole not meeting the system performance criteria for functional failure. The MR a(1) status required goals and corrective actions be established to return the system to a MR a(2) status. The inspectors found that the licensee inspected the other 56 MR scoped manholes from January 21, 1999, to July

28, 2000. The conditions of the site manhole system were assessed by direct inspector observations and review of digital pictures taken by the licensee during their inspections. The inspectors noted numerous conditions adverse to quality during the review. Deficiencies noted were cable jacket tears, cable supports corroded and broken, submerged cabling and cable splices, leaking ductbanks, and sump pumps and level control circuits inoperable. The inspectors determined on August 30, 2000, that the licensee had not generated any specific corrective action items for conditions adverse to quality identified by the licensee during site manhole, cable raceway, and ductbank inspections, since their initial inspection efforts began on January 21, 1999.

The inspectors found following a review of the Significant Root Cause Investigation completed on March 1, 1999, Engineering Service Request (ESR) 99-00445, AI #4 Response .... Cables Located in Manholes, was generated to disposition concerns identified during the SY2 manhole inspection on January 21, 1999. The ESR discussed cable bending radius limitations due to broken cable supports, sump pump inoperability and design capacity, DC grounds caused by cable water submergence, and cable water submergence. The inspectors found that the ESR resolved only the cable bending radius concern. The other issues were only discussed with statements that said removing the water from the manholes would solve the immediate problems. The inspectors noted that the Significant Root Cause Investigation completed on March 1, 1999, used the discussion items in the ESR 99-00445 (not completed until October, 1999) to disposition the manhole condition and operability questions imposed by the poor condition of the manholes. This was discussed with the licensee. On September 14, 2000, the licensee issued ESR 00-00381, Cabling Operability Within Manholes, and voided ESR 99-00445. ESR 00-00381 adequately dispositioned the operability issues raised from the total site manhole inspections, approximately one year and nine months after the original manhole inspection. Since no operability problems on safety-related equipment were identified from the engineering evaluation of the deficiencies this finding is considered of very low safety significance and was therefore characterized as Green by the Significance Determination Process (SDP).

10 CFR 50, Appendix B, Criterion XVI, Corrective Actions, states that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition. The identification of the significant condition adverse to quality, the cause of the condition, and the corrective action taken shall be documented and reported to appropriate levels of management.

Contrary to the above, on August 30, 2000, the inspectors identified that the licensee had not promptly identified and corrected conditions adverse to quality nor had they determined and taken corrective action to preclude repetition when the inspectors determined that no deficiency, work order, condition report, or corrective action was taken throughout licensee inspection efforts of 57 site safety-related manholes during which numerous deficiencies including cable jacket tears, cable supports corroded and broken, submerged cabling and cable splices, sump pumps and level control circuits inoperable, and leaking ductbanks were identified. Because this finding is considered of very low safety significance (Green) and the licensee has this problem identified in their corrective action program (AR 00024597) this finding is considered a Non-cited violation (NCV), consistent with section VI.A.1 of the NRC Enforcement Policy dated May 1, 2000. This NCV is identified as NCV 50-325(324)/00-04-01, Site Manhole Corrective Actions. This NCV closes unresolved item (URI) 50-325(324)/00-07-01, Site Manhole Corrective Actions.

.2 **Medium Voltage/DC/Instrumentation and Control Cable Water Submergence Issues**

a. Inspection Scope

The inspectors observed the licensee's equipment and conducted a review of the licensee's evaluation of the affects of water submergence on safety-related cables, splices, and cable supports.

b. Issues and Findings

During the Flood Protection Measures inspection, the inspectors found that most of the Brunswick site safety-related cables and splices in the manholes were submerged under water for significant periods of time. The ductbanks in the manholes were not sealed to prevent significant amounts of water in-leakage and the sump pumps in many of the manholes were not operable. The flooded conditions caused a significant amount of cable support corrosion. The inspectors found that two of the conductor cable jackets were damaged exposing the separator tape and copper shielding tape to a water submerged environment. The safety-related battery system background insulation ground resistance had been effected by cable submergence, and was contributing to the licensee performing ground hunting at an increased frequency.

The inspectors reviewed the licensee's specification requirements for cable and cable splicing and found that the specification was written to meet 10 CFR 50.49 environmental qualification (EQ) requirements. The cable specifications stated that the EQ qualification was lost when the cable or splices were submerged in water. The licensee's interpretation of the original design specifications for qualification requirements on the splices was that the cables were qualified by the cable manufacturer to be buried underground and were therefore qualified to be submerged in water. The inspectors reviewed several industry cable failures in medium voltage cable and a recently performed cable fault root cause evaluation from another facility. The inspectors reviewed cable water submersion and water treating issues. Also, the inspectors reviewed documentation on water submergence qualification testing that occurred in the industry. The industry has not qualified cables for long term submergence because the cable submergence testing was done for 14 days.

Based on the inspectors findings and disparities between the industry and the licensee on the acceptance of long term cable submergence at the Brunswick Facility, this issue is an URI pending further inspection. The URI is 50-325(324)/00-04-02, Brunswick Cable Submergence Issues. The following documents were reviewed:

- P&ID F-03343, Sht 1, East Yard Area, Units No. 1 & 2, Electrical Underground Duct-Banks Plan
- AR 00005664, Manhole Functional Failure
- AR 00006610, 1B Battery BW Gnd
- AR 00023242, Inadequate Manhole Inspection PT-34.15.9.9 Documentation
- AR 00023498, Manhole Safety Classification Discrepancies
- AR 00023495, Material Applied to Duct-Bank Seal
- AR 00023663, Manhole Components Omitted From A-46 Scope
- ESR 95-00372, “Manhole Water Intrusion and Effects”
- ESR 99-00445, CR 99-00313, AI #4 Response Cables Located in Manholes
- ESR 00-00381, Rev 0, “Cabling Operability Within Manholes”
- Condition Report (CR) 98-01663, Storm H_20 Input to Radwaste
- CR 98-00155, Water Leakage in DG Basement
- CR 98-00651, Manhole Cables Shorted
- First Energy, Davis-Besse Nuclear Generating Station Root Cause Analysis Report, #2 CCW Pump Trip, Rev 1, CR 1999-1648
- Updated Final Safety Analysis Report (UFSAR), Section 3.2, “Classification of Structures, Components, and Systems”
- System Description, 0SD-51, “DC Distribution,” Rev 2
- Licensee Event Report (LER) 1-98-006, “Engineered Safety Feature Actuations Due to Main Stack Wide Range Gas monitor Failure”
- Specification 048-012, “Installation of Electric Cables,” Rev 21
- Operation Log Entries 8/26/97 - 8/25/00

1R11 <u>Licensed Operator Requalification</u>

a. <u>Inspection Scope</u>

The inspectors observed licensed operator performance during simulator training for cycle 2000-04 for two crews. This observation included emergency operating procedure and abnormal operating procedure scenarios. The inspectors verified that the licensee’s requalification program for licensed operators ensures safe power plant operation by adequately evaluating how well the individual operators and crews have mastered the training objectives, including training on high-risk operator actions. The scenarios tested the operators’ ability to respond to a loss of the uninterruptible power supply (UPS) and a DC distribution panel. Additionally operator performance was observed for the response to flooding in the turbine building, trips of a circulating water intake pump (CWIP) and nuclear service water (NSW) pump, and an anticipated transient without scram (ATWS). The inspectors verified consistent clarity and formality of communication, conservative decision-making by the crew, appropriate use of procedures, proper alarm response, and high-risk reactor turbine gage board manipulations. Group dynamics and supervisory oversight, including the ability to properly identify and implement appropriate Technical Specification (TS) actions and regulatory reports and notifications, were observed. The following documents were reviewed:

- Brunswick Steam Electric Plant Units 1 and 2 Individual Plant Examination, Volume 1, August 1992
- Abnormal Operating Procedure, 0AOP-2.0, “Control Rod Malfunction/Misposition,” Rev 9
- Abnormal Operating Procedure, 0AOP-12.0, “Loss of Uninterruptible Power Supply,” Rev 13
- Abnormal Operating Procedure 0AOP-18.0, “Nuclear Service Water System Failure,” Rev 15
- Abnormal Operating Procedure, 0AOP-31.0, “Flooding in Turbine Building Condenser Pit or Pipe Tunnel,” Rev 7
- Abnormal Operating Procedure, 0AOP-37.0, “Low Condenser Vacuum,” Rev 12
- Abnormal Operating Procedure, 0AOP-39, “Loss of DC Power,” Rev 12
- Annunciator Panel Procedure, 2APP-UA-23, “Annunciator Procedure for Panel UA-23,” Rev 39
- Operating Instruction , 0OI-50.5, “120V UPS Bus 1-1A and 2-2A Electrical Load List,” Rev 11
- LOI and LOCT Core Simulator Scenario, LOT-EOP-030, “Loss of UPS, ATWS,” Rev 1
- LOI and LOCT Core Simulator Scenario, LOT-AOP-118, “Loss of DC Panel 12A, NSW Pump Trip with Failure of Standby Pump to Auto Start, Circulating Water Pump Trip, Circulating Water Leak, Turbine Building Flooding,” Rev 1

b. Issues and Findings

No findings of significance were 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 MR (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:

- 125/250V DC Station Batteries

  The following documents were viewed:
  - Maintenance Rule Unavailability Log Report - 125V DC Battery & Distribution, System 5245

- Maintenance Rule Unavailability Trend - 125V DC Battery & Distribution, System 5245
- Maintenance Rule Scoping and Performance Criteria - 125 V DC Battery & Distribution, System 5245
- Maintenance Rule Event Log Report 1/01/99-8/29/00 - 125V DC Battery & Distribution, System 5245
- Work Request/Job Order (WR/JO) 00-AFBI4 - 125V DC Battery & Distribution, System 5245
- Harris Nuclear Plant, System Scoping Review, 125V DC Electrical Distribution (Class 1E), System 5230
- Limiting Condition for Operation (LCO), Operations Log Entries 1/20/99-9/05/00, 125 V DC Battery & Distribution, System 5245
- Preventive Maintenance (PM) WR/JO ALKX and 125V DC Battery and Distribution, System 5245
- AR 17254, Battery Discharge List Interruption
- Maintenance Surveillance Test 0MST-BATT11R, "Batteries, 125V DC, Capacity Test," Rev 16
- WR/JO 00-AFBI2, 125V DC Battery & Distribution, System 5245
- WR/JO 00-AFFQ1, 125V DC Battery & Distribution, System 5245
- WR/JO 00-AFBI1, 125V DC Battery & Distribution, System 5245
- WR/JO 00-ALAJ25, 125V DC Battery & Distribution, System 5245
- PM WR/JO ALKY4, 125V DC Battery & Distribution, System 5245
- PM WR/JO ALKZ4, 125V DC Battery & Distribution, System 5245
- PM WR/JO ALKW4, 125V DC Battery & Distribution, System 5245
- PM WR/JO ALKX5, 125V DC Battery & Distribution, System 5245
- LCO A2 00-311
- LCO A2 00-304
- LCO A2 00-1408
- Maintenance Surveillance Test 0MST-BATT11Q, "Batteries, 125V DC, Quarterly Operability Test," Rev 3
-Nuclear Energy Institute, NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," April 1996, Rev 2
- Administrative Instruction 0AI-115, "125/250V DC System Ground Correction Guidelines," Rev 4

• Unit 2 RCIC Unavailability

The following documents were viewed:
- Maintenance Rule Unavailability Report
- Maintenance Rule Scoping and Performance Criteria - RCIC System
- WR/JO 00-ADPX1, Reactor Core Isolation Cooling System 2100
- Periodic Test 0PT-10.1.8, "RCIC System Valve Operability Test," Rev 9

• Unit 1 Condensate Booster Pump C Seal Failure

The following documents were viewed:
- Maintenance Rule Scope and Performance Criteria - Condensate System 3700
- Maintenance Rule Event Log Report - Condensate System 3700
- AR 00023563, 2C CBP Aux. Oil Pump

- Brunswick Steam Electric Plant, Units 1 and 2 Individual Plant Examination, Volume 1, August 1992

- Engine Driven Fire Pump Battery Charger MR Scoping

  The following documents were viewed:
  - AR 00019917, Significant Root Cause Investigation, Rev 1
  - Maintenance Rule Unavailability Log Report 7/21/99 - 7/20/00
  - Maintenance Rule Scoping and Performance Criteria - Fire Protection, System 6175
  - Maintenance Rule System Functions for System 6175
  - Catawba Maintenance Rule Inspection Report, March 20, 1997

b. Issues and Findings

On September 11 the inspectors reviewed the unavailability associated with battery cell replacement and equalization maintenance activities on the Unit 1 station batteries. During review of the operator's logs from January 1 to September 8, 2000, the inspectors noted an unsatisfactory test result for the Unit 1 Division 1 battery (1A). Further investigation by the inspectors revealed that during battery discharge testing for the 1A battery on March 16, in accordance with Maintenance Surveillance Test 0MST-BATT11R, the test equipment's timed sequence erroneously stopped the discharge test early. The inspectors determined that 21 hours and 40 minutes of unavailability time had been accrued due to the test equipment malfunction. The inspectors' review of the System 5245 Maintenance Rule Event Log Report for the 1A battery and the System 5245 Maintenance Rule Unavailability Report for the 1A battery revealed that the unavailability time accrued during the test failure had not been reported. This time added to the pre-existing 69 hours resulted in approximately 91 hours of unavailability which exceeded the licensee established MR performance monitoring group (PMG) unavailability criteria of 84 hours. The licensee initiated AR 00017254, Battery Discharge Test Interruption, to deal with the test equipment failure; however, there was no identification of MR applicability in the AR.

The inspectors discussed the observation with the licensee. After further review, the licensee verified that the 21 hours and 40 minutes of unavailability time had been missed. The licensee determined after accounting for the additional hours that the PMG criteria was exceeded on March 16, 2000, due to improperly reported additional testing on the 1A battery. AR 00023552, Missed Unavailability Event and a(1) Classification, was initiated. The inspectors noted that throughout the March 2000 maintenance activity, the inoperability and loss of functionality of the battery was properly controlled using the appropriate TS and with consideration for the risk of the plant configuration.

This failure was isolated and did not result in any additional unidentified equipment failures. This issue was evaluated using NRC Manual Chapter 0609, Attachment 0609.02. The review determined that the issue should be dispositioned outside the SDP because it was an issue which did not increase any initiating event frequency or affect any reactor safety mitigating system or barrier integrity equipment.

10 CFR 50.65 a(1), requires, in part, that the licensee shall monitor the performance or condition of structures, systems, or components (SSCs) within the scope of the rule as defined by 10 CFR 50.65 (b), against licensee-established goals, in a manner sufficient to provide reasonable assurance that such SSCs, are capable of fulfilling their intended functions.

10 CFR 50.65 a(2) states, in part, that monitoring as specified in 10 CFR 50.65 a(1) is not required where it has been demonstrated that the performance or condition of an SSC is being effectively controlled through the performance of appropriate preventive maintenance.

Contrary to 10 CFR50.65 a(1), on March 16, 2000, the licensee failed to adequately monitor system unavailability hours and take appropriate corrective actions, when the Unit 1A battery exceeded the licensee established goal for unavailability of 84 hours. This failure showed that the licensee had not demonstrated that the performance or condition of the 1A battery was being effectively controlled through the performance of appropriate preventive maintenance. This failure is considered a violation of 10 CFR 50.65 a(1) and is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy dated May 1, 2000. This NCV is identified as NCV 50-325/00-04-03, Failure to Consider Unit 1 Battery Unavailability. This violation is in the licensee’s corrective action program as AR 00023552.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

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 that the licensee had taken the necessary steps to plan and control the resultant emergent work activities:

- Work Week 30

  The following documents were reviewed:
  - BNP Risk Profile - Week 30
  - BNP1 Equipment Out Of Service (EOOS) Schedule Risk Profile - Week 30
  - BNP2 EOOS Schedule Risk Profile - Week 30
  - Progress Status Report

- Work Week 31

  The following documents were reviewed:
  - BNP Risk Profile - Week 31
  - BNP1 EOOS Schedule Risk Profile - Week 31
  - BNP2 EOOS Schedule Risk Profile - Week 31
  - Progress Status Report

- Work Week 36

  The following documents were reviewed:
  - BNP Risk Profile - Week 36
  - BNP1 EOOS Schedule Risk Profile - Week 36
  - BNP2 EOOS Schedule Risk Profile - Week 36
  - Progress Status Report

- 1A-2 Battery, Cell 35 Replacement

  The following documents were reviewed:
  - ESR 00-00345, Rev 0, "Online Battery Cell Replacement"
  - WR/JO 00-AFBI2, Battery Cell Replacement

- Unit 2 HB3 Panel refurbishment

  The following document was reviewed:
  -AR 23950, Unit 2 B Phase Main Power Transformer (MPT) Failure

b. Issues and Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine Plant Evolutions

.1 Unit 2 Feedwater Relief Valve Lift

a. Inspection Scope

The inspectors reviewed the operating crew's performance during an unscheduled Unit 2 downpower to 85 percent RTP. The downpower was necessary to accommodate the isolation of the 4A and 5A feedwater (FW) heater string in order to repair a malfunctioning FW heater relief valve. The inspectors observed corrective maintenance planning meeting and briefings, and reviewed applicable procedures, and work tickets. Additionally, the inspectors observed personnel performance from the control room during the restoration of the 4A FW heater. The operators utilized the condensate and FW system operating procedure, "Operating Procedure 2OP-32, Condensate and Feedwater System," Rev 103, for placing the bypassed FW heater back in service from the non-routine condition.

b. Issues and Findings

No findings of significance were identified.

.2 Unit 2 Reactor Scram Due to B Phase Main Power Transformer Failure

a. Inspection Scope

Personnel performance was evaluated following a September 22 event in which a fire in the Unit 2 B phase main power transformer caused a turbine trip followed by a reactor scram. The fire on the main power transformer was quickly extinguished by the transformer deluge system and plant fire brigade, and did not threaten any safety-related equipment. The cause of the transformer fire was a loss of cooling due to a power supply breaker tripping. Plant personnel had no indication of the loss of transformer cooling because the main transformer trouble annunciator was in an alarm condition (as a result of maintenance work) and compensatory measures had not been established. Offsite power remained available throughout the event and all emergency diesel generators were also available.

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 procedures and training; and evaluated the occurrence and subsequent personnel response using the significance determination process. The following documents were reviewed:

- Emergency Operating Procedure 2EOP-01-RSP, “Reactor Scram Procedure” Rev 4
- Operating Instruction 0OI-01.08, “Control of Equipment and System Status,” Rev 27
- Nuclear Generation Group Standard Procedure OPS-NGGC-1301, “Equipment Clearance,” Rev 5

b. Issues and Findings

No findings of significance were 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; and (5) where continued operability was considered unjustified, the impact on TS limiting conditions for operations (LCOs) and the risk significance in accordance with the SDP. These reviews were performed for the following:

- Unit 1 Residual Heat Removal Service Water Booster Pump D - high vibration

- Emergency Diesel Generator Operability - equipment clearance adequacy

  The following documents were reviewed:
  - Clearance No. 2-00-00-656
  - Clearance No. 2-00-00-656A
  - Clearance No. 2-00-00-656B
  - Administrative Instruction, 0AI-122, “Pre-job Brief /Post-job Review Checklist,” Rev 4
  - LCO A1-00-1160
  - LCO A2-00-1161
  - P&ID F-9345 Sht 1, Diesel Generator No. 1 Circuits Control Wiring Diagram
  - P&ID F-9347 Sht 1, Diesel Generator No. 3 Circuits Control Wiring Diagram
  - P&ID F-9348 Sht 1, Diesel Generator No. 4 Circuits Control Wiring Diagram
  - Operating Instruction, 0OI-50, “125/250V DC Electrical Load List,” Rev 19

- Emergency Diesel Generator Operability - Manual Voltage Regulator Failures

  The following documents were reviewed:
  - Operating Procedure 0OP-39, “Diesel Generator Operating Procedure,” Rev 88
  - P&ID F-09348, Sht 1, Units 1 & 2 Diesel Gen. No. 4 Circuits Control Wiring Diagram

- Emergency Diesel Generator Operability - Injector Failure

  The following documents were reviewed:
  - Operating Instruction 0OI-1.08, “Control of Equipment and System Status,” Rev 7
  - WR/JO 00-AFKW1, Diesel Generators System 5095
  - AR 00023344, DG#1 Cylinder #7 Low Temperature
  - Corrective Maintenance 0CM-ENG526, “Diesel Engine Fuel Injectors,” Rev 8

- RCIC System Water Contaminated Oil

  The following documents were reviewed:
  - Periodic Test 0PT-10.1.1, “RCIC System Operability Test,” Rev 78
  - Plant Document - Unit 2 RCIC Turbine Lube Oil Moisture Intrusion Action Plan
  - ESR 00-0037, “2-E51-C002, Lube Oil Clr Rear Bonnet Drain Leak,” Rev 0
  - ESR 00-0038, “RCIC Operability Eval for Water in Lube Oil,” Rev 0
  - ESR 00-00338, “RCIC Operability Eval. For Water in Lube Oil,” Rev 1
  - Plant Document - Fault Tree & Fault Matrix
  - System Description 0SD-16, “Reactor Core Isolation Cooling System Description,” Rev 1

- Units 1 and 2 Drywell Dome Studs

  ESR 00-00360, “Operability of Prim. Contmt. With Reduced Dome Stud Preload,” Rev 0

b. Issues and Findings

No findings of significance were identified.

1R16 Operator Workarounds

a. Inspection Scope

The inspectors reviewed degraded or non-conforming conditions that could complicate the operation of plant equipment and required compensation by the operators. The following mitigating systems were reviewed to determine if system function or the operator's ability to implement abnormal and emergency operating procedures were affected:

- RHR system venting due to unidentified system inleakage

  The following documents were viewed:
  - P&ID D-2526 Shts 2A and 2B, Reactor Building Piping Diagram Residual Heat Removal System Unit 2

The inspectors reviewed the cumulative effects of operator workarounds. The inspectors reviewed the workarounds on reliability, availability, and potential mis-operation of the system involved. The inspectors reviewed whether the operator workarounds on Unit 1 and Unit 2 could increase an initiating event frequency or could affect multiple mitigating systems. The inspectors also reviewed the cumulative effects of operator workarounds on operator correct and timely response to plant transients and accidents. Items reviewed:

- Workarounds dated Friday, September 8, 2000, on Unit 1 and Unit 2.

b. Issues and Findings

No findings of significance were 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, demonstrated that the affected equipment was capable of performing it's intended function, and was operable in accordance with TS:

- Unit 2 Uninterruptible Power Supply

  The following document was reviewed:
  - Operating Procedure 2OP-52, "120 Volt AC UPS, Emergency and Conventional Electrical Systems Operating Procedure," Rev 46

- Battery 1A-2, Cell 54

  The following document was reviewed:
  - Maintenance Surveillance 0MST-BATT11Q, "Batteries, 125V DC, Quarterly Operability Test," Rev 3

- DG-2 Potential Transformer Modification

  The following documents were reviewed:
  - Operating Procedure 0OP-39, "Diesel Generator Operating Procedure," Rev 88, Section 8.10 - Diesel Generator Operation for Maintenance or Testing
  - Periodic Test 0PT-12.2B, "No. 2 Diesel Generator Monthly Load Test," Rev 66
  - Post Maintenance Test Requirements - Report PMTR4-WR/JO 00-ABQI1
  - ESR 00-00131 Attachment G - Test Data Sheet, Rev 0

- Battery 1A-2, Cell 35, Replaced

  The following documents were reviewed:
  - Maintenance Surveillance Test 0MST-BATT11Q, "Batteries, 125V DC, Quarterly Operability Test," Rev 3
  - Preventive Maintenance 0PM-BATT04A, "125V DC, Plant Battery Link Cleaning," Rev 7
  - Plant Program Procedure 0PLP-20, "Post Maintenance Program," Rev 22

- Unit 1 HPCI Minor Valve Repairs and Flow Instrument Calibration

  The following documents were reviewed:
  - WR/JO ALPO 004, Calibrate 1-E41-FT-N008
  - WR/JO ADRM 027, Perform 0PT-09.2 HPCI Full Flow Test and Valve Operability at Rated Pressure

- WR/JO ARKW 013, Perform 0PT-9.7 to Determine the Operability of the HPCI

  System Valves
- WR/JO 98-AEZU3, Minor Packing Leak on 1-E41-F001
- Periodic Test 0PT-09.7, “HPCI System Valve Operability Test,” Rev 7
- Periodic Test 0PT-0.2, “HPCI System Operability Test,” Rev 105

• Unit 2 RCIC Valve Maintenance

  The following documents were reviewed:
  - Periodic Test 0PT-10.1.8, “RCIC system Valve Operability Test,” Rev 9
  - WR/JO 00-ADPX1, RCIC System 2100

b. Issues and Findings

No findings of significance were identified.

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 (UFSAR) 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:

• Unit 2 HPCI

  The following documents were reviewed:
  - Periodic Test 0PT-9.2, “HPCI System Operability Test Unit 2,” Rev 105
  - Periodic Test 0PT-11.4, “Turbine Exhaust Check Valve,” Rev 1

• Unit 1 Core Spray A

  The following document was reviewed:
  - Periodic Test 0PT-7.2.4a, “Core Spray System Operability Test - Loop A,” Rev 46

• DG-2 Monthly Load Run

  The following document was reviewed:
  - Periodic Test 0PT-12.2B, “No. 2 Diesel Generator Monthly Load Test,” Rev 66

• Unit 2 RCIC Inservice Testing (IST) Surveillance

  The following documents were reviewed:
  - Periodic Test 0PT-10.1.1, “RCIC System Operability Test,” Rev 78
  - ANSI/ASME, OM-1987, “Operation and Maintenance of Nuclear Power Plants,”

Oma-1088 Addenda Part 6 & 10
- System Description 0SD-16, “Reactor Core Isolation Cooling System Description” (Design Basis Document)

- Unit 2 RHR Service Water (SW) Surveillance Test

  The following documents were reviewed:
  - Periodic Test 0PT-08-1.4a, “RHR Service Water System Operability Test - Loop A,” Rev 50
  - System Description 0SD-43, “Service Water System,” Rev 2

- Unit 1 SLC System

  The following documents were reviewed:
  - Periodic Test 0PT-06.1, “Standby Liquid Control System Operability Test,” Rev 55
  - UFSAR 9.3.4, “Standby Liquid Control System”
  - TS 3.1.7, Standby Liquid Control (SLC) System

b. Issues and Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed a modification to the main steam isolation valve closure logic as well as the associated 10 CFR 50.59 screening against the system design bases documentation, including UFSAR and TS. The review verified that configuration control of the modification was adequate by verifying that any affected plant documents, such as drawings and procedures were properly controlled. The following documents were reviewed:

- Nuclear Generation Group Standard Procedure REG-NGGC-0002, “10 CFR 50.59 and Other Regulatory Evaluations,” Rev 3
- P&ID 1-FP-55046, Shts 4,6,7, 8, 9, 14,and 15, Unit 1 General Electric Reactor Protection System Elementary Diagram

b. Issues and Findings

No findings of significance were identified.

3. Safeguards
Cornerstone: Physical Protection

3PP1 Access Authorization

a. Inspection Scope

The inspector's interviewed representatives of licensee management and escort personnel concerning their understanding of the Behavior Observation portion of the personnel screening and Fitness For Duty (FFD) program. In interviewing these personnel the inspectors reviewed the effectiveness of their training and abilities to recognize aberrant behavioral traits.

b. Issues and Findings

No findings of significance were identified.

3PP2 Access Control

a. Inspection Scope

The inspectors observed access control activities against the requirements of the Physical Security Plan on August 1,2, and 3, 2000, and the equipment testing conducted on August 2, 2000. In observing the access control activities the inspectors assessed whether officers could detect contraband before it was introduced into the protected area. Additionally, the inspectors assessed whether the officers were conducting access control equipment testing according to regulatory requirements.

b. Issues and Findings

No findings of significance were 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:

- Safety System Unavailability

- Emergency AC Power System
  The following documents were reviewed:

- Maintenance Rule Unavailability Log Report, 5/18/96 -12/20/99, System 5095 Diesel Generator
- Self-Assessment Report, Assessment number-9822, March 27-31, 2000, “NRC Performance Indicators”
- MR Unavailability Log Report, 5/18/96-12/20/99, System 4060 Service Water
- MR Unavailability Log Report, 5/18/96-12/20/99, System 5245, 125V DC Battery and Distribution
- MR Unavailability Log Report, 5/8/96-12/20/99, System 5230, 250V DC Distribution
- MR Unavailability Log Report, 5/18/96-12/20/99, System 5240 125V Battery Chargers
- MR Unavailability Log Report, 5/18/96-12/20/00, System 5112, Diesel Generator Starting Air

• High Pressure Injection System Unavailability

  The following document was reviewed:
  - MR Unavailability Log Report 1/01/00 - 7/01/00, System 2095 HPCI

• Heat Removal System Unavailability

  The following document was reviewed:
  - MR Unavailability Log Report 1/01/00 - 7/01/00, System 2100 RCIC

b. Issues and Findings

No findings of significance were identified.

4OA3 Event Follow-up

.1 Unit 2 Reactor Scram Due to a Failure of the 2B Main Transformer

a. Inspection Scope

On September 22 the licensee declared a Notification of Unusual Event (NOUE) at 4:00 a.m., for Unit 2 due to a fire in the 2B main power transformer that lasted greater than 10 minutes. The transformer failure caused a turbine trip followed by a reactor scram at 3:46 a.m. The fire was extinguished by the transformer deluge system and the site fire brigade at 4:02 a.m. The licensee terminated the NOUE at 5:12 a.m. The reactor scrammed in response to the main turbine trip, and all expected isolations and safety systems responded properly to the event. This event is documented in the licensee’s corrective action program as AR 00023950, Unit #2B Phase Main Power Transformer Failure.

The inspectors observed and reviewed plant parameters and status following the event. The inspectors verified that the licensee properly classified the event in accordance with emergency action level procedures and made timely notification to the NRC. The inspectors also evaluated performance of mitigating systems and licensee actions for proper response. The following document was reviewed:

-Plant Emergency Procedure 0PEP-02.1, “Initial Emergency Actions,” Rev 47

b. Issues and Findings

No findings of significance were identified.

.2 (Closed) Licensee Event Report (LER) 50-325(324)/00-003-00: Control Room Emergency Ventilation System Actuation During Chlorination System Breach. This LER was a minor issue and was closed.

.3 (Closed) LERs 50-325/98-003-00 and 98-003-01: Safety Relief Valve Exceeded Technical Specifications Setpoint Limits. The licensee found that four of the 11 main steam safety/relief valves (SRVs) exceeded lift settings in excess of the TS setpoint tolerance. The cause was attributed to valve disassembly and re-assembly practices related to cleanliness, application of lubricant, and pilot disc lapping. Although the SRV setpoint limits required by the TS were exceeded, the condition is bounded by a General Electric (GE) analysis, Evaluation of SRV Setpoint Drift Brunswick Steam Electric Plant Units 1 and 2, dated March 1986. The SRV vendor testing indicated that all 11 SRVs would have actuated with the lift setpoints at their as-found values and, therefore, the licensee concluded that no operational safety concern existed. There was, therefore, no loss of safety function. The SRV pilot valves were replaced with certified spares prior to startup of Unit 1. As a corrective action for this event, the licensee reviewed with the vendor procedural/practice changes to the vendor’s valve rebuilding process. In addition, maintenance and engineering personnel monitored vendor rebuild of the valves that were installed in Unit 2 during the last refueling outage to ensure that the lessons learned from evaluations of this issue were incorporated into the rebuild process. Because there was no firm evidence to establish the time of SRV inoperability, it was appropriately assumed that it was the time of discovery (i.e., during surveillance testing). Since the licensee had complied with the TS action statement requirements at the time of surveillance testing this issue did not constitute a violation of NRC requirements.

.4 (Closed) LERs 50-324/99-005-00 and 99-005-01: Safety Relief Valves Exceeded Technical Specification Setpoint Limits. The licensee found that two of 11 main steam SRVs exceeded lift settings in excess of the TS setpoint. The cause was attributed to a bent pilot rod on one SRV, and valve disassembly and re-assembly practices and pilot disc lapping on the other SRV, as stated above. This event was bounded by GE analysis, and all 11 SRVs would have actuated with the lift setpoints at their as-found values. There was, therefore, no loss of safety function. The SRV pilot valve assemblies were replaced with certified spares prior to startup of Unit 2. In addition to incorporating the lessons learned from LER 1-98-003, the licensee is planning to begin rebuilds of the SRV’s by site personnel in 2001. Site ownership of the rebuild process is expected to improve the quality of the process. Because there was no firm evidence to establish the time of SRV inoperability, it was appropriately assumed that it was the time of discovery (i.e., during surveillance testing). Since the licensee had complied with the TS action statement requirements at the time of surveillance testing this event did not constitute a violation of NRC requirements.

.5 (Closed) Inspection Follow-up Item IFI 50-325(324)/98-14-04: Consideration of Bypass Leakage in Control Room and Offsite Dose Calculations. In 1982 the drain lines from the HPCI and RCIC turbine drain pots were modified by rerouting them from the reactor building equipment drain tank to the main condenser. During review of these modifications (PM 82-137 and 82-138), NRC questioned whether the modifications circumvented the design intent in that it created what appeared to be new release paths that bypassed secondary containment. Discussions with licensee engineers and review of the UFSAR showed that the Brunswick licensing basis did not require accounting for leakage that bypassed the secondary containment and that bypass leakage was not considered in the licensee's analyses for offsite and control room accident doses. PM 82-137 and 82-138 were re-examined. The basis specified in the 10 CFR 50.59 evaluations for the rerouting the drain lines was consistent with the original design specified by GE. The original GE design for the drain lines shown on drawing numbers 729E600BB and 729E604BB indicated that these drain lines should be routed to the main condenser. This design is consistent with other Region II boiling water reactors. Subsequent to the inspection, a review was performed by NRC of the licensing basis for offsite dose analysis. This review was documented in a memorandum to R. Correia from M. Reinhart, dated March 10, 2000, Subject: Response to Region II Task Interface Agreement Regarding Licensing Bases for Control Room and Offsite Dose Calculations at Brunswick (TAC NO. MA6662). The review showed that the licensing basis was as stated by licensee engineers and as referenced in the licensee’s submittals for the power uprate amendment. Review of the UFSAR showed that the values for control room dose and the site boundary doses for design basis analysis loss-of-coolant accident exposures, had not been revised to reflect the power uprate dose calculations. The failure to amend the UFSAR was identified as a violation of 10 CFR 50.71(e). The violation was determined to have very low safety significance and was identified as a minor violation. This violation is in the licensee’s corrective action program as AR 00009136, UFSAR Section 15.6.4 Not Updated for Power Uprate.

4OA5 Other

.1 (Closed) NRC Temporary Instruction (TI) 2515/144: Performance Indicator Data Collecting and Reporting Process Review. The inspectors completed NRC TI 2515/144 and documented the results of that inspection in NRC Integrated Inspection Report Nos. 50-325/00-03 and 50-324/00-03, dated July 27, 2000.

4OA6 Management Meetings

.1 Exit Meeting Summary

The inspectors presented the inspection results to Mr. J. S. Keenan, Vice President, and other members of licensee management at the conclusion of the inspection on October 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.

.2 Revised Oversight Process Public Meeting

On July 11, 2000, the NRC held a public meeting at the Southport City Hall to discuss the new reactor oversight process which began at the Brunswick Plant in April. During the meeting NRC representatives presented highlights of the new process and provided an opportunity for members of the public to ask questions.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

A. Brittain, Manager Security
N. Gannon, Plant General Manager
J. Gawron, Training Manager
W. Dorman, Manager Regulatory Affairs
G. Johnson, Manager Nuclear Assessment (Acting)
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

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-325(324)/00-04-02	URI	Brunswick Cable Submergence Issues (1R06)

Opened and Closed During This Inspection

50-325(324)/00-04-01	NCV	Site Manhole Corrective Actions (1R06)
50-325/00-04-03	NCV	Failure to Consider Unit 1 Battery Unavailability (1R12)

Closed

50-325(324)/00-07-01	URI	Site Manhole Corrective Actions (1R06)
50-325(324)/00-003-00	LER	Control Room Emergency Ventilation System Actuation During Chlorination System Breach (4OA3)
50-325/98-003-00	LER	Safety Relief Valve Exceeded Technical Specifications Setpoint Limits (4OA3)
50-325/98-003-01	LER	Safety Relief Valve Exceeded Technical Specifications Setpoint Limits. (4OA3)
50-324/99-005-00	LER	Safety Relief Valves Exceeded Technical Specification Setpoint Limits (4OA3)
50-324/99-005-01	LER	Safety Relief Valves Exceeded Technical Specification Setpoint Limits (4OA3)
50-325(324)/98-14-04	IFI	Consideration of Bypass Leakage in Control Room and Offsite Dose Calculations (4OA3)
TI 2515/144	TI	PI Data Collecting and Reporting Process Review (4OA5)

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.