UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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ARLINGTON, TEXAS 76011-4005

May 8, 2003

Paul D. Hinnenkamp
Vice President - Operations
River Bend Station
Entergy Operations, Inc.
P.O. Box 220
St. Francisville, Louisiana 70775

SUBJECT: RIVER BEND STATION - NRC INTEGRATED INSPECTION REPORT 50-458/03-03

Dear Mr. Hinnenkamp:

On April 12, 2003, the NRC completed an inspection at your River Bend Station. The enclosed report documents the inspection findings which were discussed on April 1, 2003, with you and other members of your staff.

This 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. Within these areas, the inspection consisted of selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, the NRC has identified issues that were evaluated under the risk significance determination process as having very low safety significance (Green). The NRC has also determined that violations are associated with these issues. These violations are being treated as noncited violations (NCV), consistent with Section VI.A of the Enforcement Policy, and are described in the subject inspection report. Additionally, one issue was identified regarding potential flooding of the G-Tunnel. The risk assessment for this issue is ongoing, and you will be notified when the significance is determined. If you contest the violation or significance of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the River Bend Station facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response will be made 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/reading-rm/adams.html (the Public Electronic Reading Room).

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

/RA/

David N. Graves, Chief
Project Branch B
Division of Reactor Projects

Docket: 50-458
License: NPF-47

Enclosure:
NRC Inspection Report
50-458/03-03

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-458
License:	NPF-47
Report No.:	50-458/03-03
Licensee:	Entergy Operations, Inc.
Facility:	River Bend Station
Location:	5485 U.S. Highway 61 St. Francisville, Louisiana
Dates:	December 29, 2002, through April 12, 2003
Inspectors:	P. J. Alter, Senior Resident Inspector M. O. Miller, Resident Inspector B. D. Baca, Health Physicist, Plant Support L. E. Ellershaw, Senior Reactor Inspector, Engineering and Maintenance R. E. Lantz, Senior Emergency Preparedness Inspector, Plant Support G. B. Miller, Reactor Inspector, Engineering and Maintenance L. T. Ricketson, P.E., Senior Health Physicist, Plant Support M. F. Runyan, Senior Reactor Inspector, Engineering and Maintenance W. C. Sifre, Reactor Inspector, Engineering and Maintenance
Approved By:	D. N. Graves, Chief, Project Branch B
ATTACHMENT:	Supplemental Information

SUMMARY OF FINDINGS

River Bend Station
NRC Inspection Report 50-458/03-03

IR 05000458/2003-003; 12/29/2002 - 04/12/2003; River Bend Station; Flood Prot Measures, Personnel Perf During Nonroutine Plant Evolutions, Access Ctrl to Rad Significant Areas, ALARA Planning & Controls, & Ident & Res of Problems.

The report covered a 15-week period of routine inspections by resident inspectors and announced inspections by regional emergency planning, engineering and maintenance, and radiation protection inspectors. Six Green noncited violations, two Green findings, and one unresolved item whose risk significance has yet to be determined, were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609, “Significance Determination Process.” Findings for which the Significance Determination Process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 3, dated July 2000.

A. Inspector Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- TBD. The inspectors determined that the licensee failed to maintain the watertight integrity of two doors at the end of an underground tunnel that opens into the excavation area for Unit 2.

  The finding was unresolved pending completion of the significance determination process. The finding was more than minor because it was associated with flood protection measures and degraded the ability to meet the mitigating systems cornerstone objective. This condition adversely impacted the flooding protection for an area of the plant containing system components related to the safe shutdown of the reactor, namely the standby service water system (Section 1R06).

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, for failure to take proper corrective action following a failure of the low pressure core spray pump minimum flow valve that resulted in an identical failure of the residual heat removal Pump A minimum flow valve 9 months later.

  The inspector-identified noncited violation was greater than minor because it was associated with the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems (residual heat removal Train A) that respond to initiating events to prevent undesirable consequences. The inspectors evaluated the finding using IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Reactors,” and determined that the residual heat removal Pump A minimum flow valve failure was of very low safety significance because the other low pressure coolant injection systems were available and the other train of suppression pool cooling was available at the time (Section 4OA2.1).

Cornerstone: Occupational Radiation Safety

- Green. The licensee failed to develop a sufficiently detailed work plan for the decontamination of the reactor vessel bellows, in violation of Technical Specification 5.4.1.a. The work plan failed to provide guidance on maintaining highly contaminated surfaces (the reactor vessel bellows surface) wet, using a hydrolaser with a rotary surface cleaner, or briefing the individual using the hydrolaser. The lack of a detailed work plan contributed to an unexpected increase in airborne radioactivity and unplanned personnel exposures.

  This self-revealing, noncited violation was greater than minor because it was associated with one of the Occupational Radiation Safety Cornerstone attributes (exposure/contamination control) and the finding affected the associated cornerstone objective (to ensure the adequate protection of worker health and safety from exposure to radiation from radioactive material). The inspector processed the violation through the Occupational Radiation Protection Significance Determination Process because the occurrence involved potential doses (resulting from actions or conditions contrary to licensee procedures) which could have been significantly greater as a result of a single, minor, reasonable alteration of the circumstances. However, because the violation was not an as low as is reasonably achievable finding, there was no personnel overexposure, there was no substantial potential for personnel overexposure, and the finding did not compromise the licensee's ability to assess dose, the violation had no more than very low safety significance (Section 2OS1).

- Green. The licensee failed, in three examples, to survey or evaluate radiological hazards when conducting reactor vessel bellows decontamination, in violation of 10 CFR 20.1501(a). First, the licensee failed to evaluate the highest concentration of radioactive contamination on the reactor vessel bellows. Additionally, the licensee failed to evaluate the airborne radioactivity in the immediate vicinity of reactor vessel bellows contamination. Later, the licensee failed to evaluate the airborne radioactivity levels throughout the containment building when continuous air monitors alarmed.

  This self-revealing, noncited violation was greater than minor because it was associated with one of the Occupational Radiation Safety Cornerstone attributes (exposure/contamination control) and the finding affected the associated cornerstone objective (to ensure the adequate protection of worker health and safety from exposure to radiation from radioactive material). The inspector processed the violation through the Occupational Radiation Protection Significance Determination Process because the occurrence involved potential doses (resulting from actions or conditions contrary to licensee procedures) which could have been significantly greater as a result of a single, minor, reasonable alteration of the circumstances. However, because the violation was not an ALARA finding, there was no personnel overexposure, there was no substantial potential for personnel overexposure, and the finding did not compromise the licensee's ability to assess dose, the violation had no more than very low safety significance (Section 2OS1).

- Green. The licensee failed to post the reactor containment building as an airborne radioactivity area, in violation with 10 CFR 20.1902(d). Airborne radioactivity levels exceeded the allowable limits in 10 CFR Part 20, Appendix B, by as much as 3.5 times. The condition existed for at least 5 hours.

  This self-revealing, noncited violation was greater than minor because it was associated with one of the Occupational Radiation Safety Cornerstone attributes (exposure/contamination control) and the finding affected the associated cornerstone objective (to ensure the adequate protection of worker health and safety from exposure to radiation from radioactive material). The inspector processed the violation through the Occupational Radiation Protection Significance Determination Process because the occurrence involved potential doses (resulting from actions or conditions contrary to licensee procedures) which could have been significantly greater as a result of a single, minor, reasonable alteration of the circumstances. However, because the violation was not an ALARA finding, there was no personnel overexposure, there was no substantial potential for personnel overexposure, and the finding did not compromise the licensee's ability to assess dose, the violation had no more than very low safety significance (Section 2OS1).

- Green. Following an occurrence that caused an airborne radioactivity area, the licensee failed to inform workers of radiological conditions that had changed and of precautions to minimize exposure, in violation of 10 CFR 19.12.

  This self-revealing, noncited violation was greater than minor because it was associated with one of the Occupational Radiation Safety Cornerstone attributes (exposure/contamination control) and the finding affected the associated cornerstone objective (to ensure the adequate protection of worker health and safety from exposure to radiation from radioactive material). The inspector processed the violation through the Occupational Radiation Protection Significance Determination Process because the occurrence involved potential doses (resulting from actions or conditions contrary to licensee procedures) which could have been significantly greater as a result of a single, minor, reasonable alteration of the circumstances. However, because the violation was not an ALARA finding, there was no personnel overexposure, there was no substantial potential for personnel overexposure, and the finding did not compromise the licensee's ability to assess dose, the violation had no more than very low safety significance (Section 2OS1).

- Green. The licensee failed to control an area with dose rates of 1000 millirems per hour as a locked high radiation area, in violation of Technical Specification 5.7.2. After a plant scram on September 18, 2002, a worker entered the reactor core isolation cooling area on the 95-foot elevation of the auxiliary building and received an electronic dosimeter dose rate alarm. A crud burst resulting from a transient that occurred approximately 3 hours previously caused the dose levels in the area entered by the worker to increase to 1000 millirems per hour. Historically, the site has experienced crud bursts under similar conditions and the increase in dose rate should have been anticipated and evaluated.

This self-revealing, noncited violation was greater than minor because it was associated with one of the Occupational Radiation Safety Cornerstone attributes (exposure/contamination control) and the finding affected the associated cornerstone objective (to ensure the adequate protection of worker health and safety from exposure to radiation from radioactive material). The inspector processed the violation through the Occupational Radiation Protection Significance Determination Process because the occurrence involved potential doses (resulting from actions or conditions contrary to licensee procedures) which could have been significantly greater as a result of a single, minor, reasonable alteration of the circumstances. However, because the violation was not an ALARA finding, there was no personnel overexposure, there was no substantial potential for personnel overexposure, and the finding did not compromise the licensee's ability to assess dose, the violation had no more than very low safety significance (Section 2OS1).

- Green. A finding was identified because the licensee failed to maintain collective doses as low as reasonably achievable. Specifically, the work activity collective dose associated with RWP 2003-1929 Task 1, "RFO-11 Recirculation Pump Work," exceeded 5 person-rem and exceeded the dose estimation by more than 50 percent.

  The failure to maintain collective doses as low as reasonably achievable is a performance deficiency. This finding was more than minor because it is associated with the Occupational Radiation Safety Cornerstone attribute (ALARA planning/projected dose) and affected the associated cornerstone objective (to ensure adequate protection of worker health and safety from exposure to radiation). This occurrence involved worker inefficiencies and inadequate planning, scheduling, and supervisory oversight, which resulted in unplanned, unintended occupational collective dose for a work activity. When processed through the Occupational Radiation Safety Significance Determination Process, this finding was found to have no more than very low safety significance because the finding was an ALARA Planning issue, the licensee's 3-year rolling average collective dose was greater than 240 person-rem, the actual dose for the work activity was not more than 25 person-rem, and there were no more than four occurrences (Section 2OS2).

- Green. A finding was identified because the licensee failed to maintain collective doses as low as reasonably achievable. Specifically, the work activity collective dose associated with Radiation Work Permit 2003-1935, "Drywell Valve Maintenance, to include Repacks and Support Work," exceeded 5 person-rem and exceeded the dose estimation by more than 50 percent.

  The failure to maintain collective doses as low as reasonably achievable is a performance deficiency. This finding was more than minor because it is associated with the Occupational Radiation Safety Cornerstone attribute (ALARA planning/projected dose) and affected the associated cornerstone objective (to ensure adequate protection of worker health and safety from exposure to radiation). This occurrence involved worker inefficiencies and inadequate planning which resulted in unplanned, unintended occupational collective dose for a work activity. When processed through the Occupational Radiation Safety Significance Determination Process, this finding was found to have no more than very low safety significance because the finding was an

ALARA Planning issue, the licensee's 3-year rolling average collective dose was greater than 240 person-rem, the actual dose for the work activity was not more than 25 person-rem, and there were no more than four occurrences (Section 2OS2).

B. Licensee Identified Findings

Two violations of very low significance which were identified by the licensee have been reviewed by the inspectors. Corrective actions taken or planned by the licensee appear to be reasonable. These violations are listed in Section 4OA7 of this report.

Report Details

Summary of Plant Status: The plant was operated at 100 percent power from the beginning of the inspection period until January 18, 2003, when reactor power began to decrease in coastdown to the upcoming refueling outage (RFO-11). On February 4, 2003, power was reduced to 82 percent to implement feedwater temperature reduction. On February 5, 2003, power was increased to 98 percent and coastdown operations resumed. On February 22, 2003, the reactor was manually scrammed because of a hydraulic leak in the main turbine control system. Following repairs, the reactor was restarted on February 23, 2003. On March 1, 2003, power was increased to 90 percent and coastdown operations resumed. On March 16, 2003, the reactor was shut down to begin RFO-11. On April 10, 2002, the reactor was restarted, and steam plant startup was in progress with power at 40 percent when the inspection period ended.

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

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

During the weeks of February 24 and March 3, 2003, the inspectors reviewed the licensee's plant procedures used to protect mitigating systems from freezing weather conditions. Specifically the inspectors: (1) verified that selected systems and components had remained functional when challenged by freezing weather conditions; (2) verified that temperature had been frequently monitored; (3) verified that operation of plant features during freezing conditions were appropriate; and (4) evaluated implementation of the freezing weather preparation procedures and compensatory measures for affected systems or components before the onset of and during freezing weather conditions. The inspectors reviewed Operations Section Procedure (OSP) OSP-0043, "Freeze Protection and Temperature Maintenance," Revision 4, including the attachments completed for cold weather conditions during November and December 2002 and January 2003.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

.1 Division II Emergency Diesel Generator Walkdown

On January 23, 2003, the inspectors performed a partial system walkdown of the Division II emergency diesel generator while the Division I emergency diesel generator was out of service for planned maintenance. The inspectors reviewed System Operating Procedure SOP-0053, "Standby Diesel Generator and Auxiliaries," Revision 35, to determine the correct system lineup. Then the inspectors walked down critical portions of the system to identify any discrepancies between the existing equipment lineup and the correct lineup.

.2 Low Pressure Core Spray System Walkdown

On March 5, 2003, the inspectors performed a partial system walkdown of low pressure core spray while low pressure coolant injection Train B was out of service for planned maintenance. The inspectors reviewed System Operating Procedure SOP-0032, "Low Pressure Core Spray System," Revision 18A, to determine the correct system lineup. Then the inspectors walked down critical portions of the system to identify any discrepancies between the existing equipment lineup and the correct lineup.

.3 Shutdown Cooling System Walkdown

On March 23, 2003, the inspectors performed a partial system walkdown of the residual heat removal Train B fuel pool cooling assist mode of operation for shutdown cooling while the normal shutdown cooling suction flowpath was out of service. The inspectors reviewed System Operating Procedure SOP-0031, "Residual Heat Removal System," Revision 40, to determine the correct system lineup. Then the inspectors walked down critical portions of the system to identify any discrepancies between the existing equipment lineup and the correct lineup.

.4 Suppression Pool Cleanup and Alternate Decay Heat Removal Walkdown

On April 2, 2003, the inspectors performed a partial system walkdown of the suppression pool cleanup and alternate decay heat removal system during RFO-11 with residual heat removal System B providing decay heat removal. The inspectors reviewed System Operating Procedure SOP-0140, "Suppression Pool Cleanup and Alternate Decay Heat Removal," Revision 12, to determine the correct system lineup. Then the inspectors walked down critical portions of the system to identify any discrepancies between the existing equipment lineup and the correct lineup.

.5 Residual Heat Removal System Walkdown

On April 10, 2003, the inspectors performed a partial system walkdown of residual heat removal Train A when it was secured from its shutdown cooling lineup and placed in low pressure coolant injection standby lineup. The inspectors reviewed System Operating Procedure SOP-0031, "Residual Heat Removal," Revision 40, to determine the correct system lineup. Then the inspectors walked down critical portions of the system to identify any discrepancies between the existing equipment lineup and the correct lineup.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

.1 Fire Protection Area Walkdowns

Throughout the period the inspectors toured six plant areas important to reactor safety to observe conditions related to: (1) licensee control of transient combustibles and ignition sources; (2) the material condition, operational lineup, and operational effectiveness of fire protection systems, equipment and features; and (3) the material condition and operational status of fire barriers used to prevent fire damage or fire propagation.

- Auxiliary building 70 foot elevation, residual heat removal Train A room, Fire Area AB-5, on February 25, 2003

- Auxiliary building 70 foot elevation, emergency core cooling piping penetration room, Fire Area AB-1/Z-1, on March 5, 2003

- Auxiliary building 70 foot elevation, low pressure core spray room, Fire Area AB-6/Z-1, on March 5, 2003

- Auxiliary building east and west stairwell fire barrier penetrations breached for temporary power to fire protection alarm panels (Temporary Alteration 2002-0026) on March 12, 2003

- Auxiliary building 95 foot elevation, shield building access area, Fire Area AB-15/Z-2, on March 28, 2003

- Control building, southwest stairwell, Fire Area C-30, on March 28, 2003

The inspectors reviewed the following documents during the fire protection inspections:

- Pre-Fire Strategy Book
- Updated Safety Analysis Report (USAR) Section 9A.2, “Fire Hazards Analysis”
- River Bend post-fire safe shutdown analysis

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors conducted a periodic external flooding assessment to verify that the licensee’s flooding mitigation plans and equipment were consistent with design requirements and risk analysis assumptions. The inspectors conducted a walkdown of external areas of the plant and underground G-Tunnel from the Unit 2 excavation area to the standby cooling tower. Specifically, the inspectors examined: (1) sealing surfaces of watertight doors; (2) sealing of equipment below design flood level; (3) sealing of penetrations in floors and walls; (4) operable sump pumps and level alarm circuits; and (5) interconnections with common drain systems. The inspectors reviewed the following documents during the inspection:

- River Bend individual plant examination of external events
- USAR Section 2.4.3, “Probable Maximum Flood on Rivers and Streams”
- Stone & Webster Engineering Calculation 12210 # 8.3.1.32, “Design Basis Flood,” Revision 1, dated June 14, 1984
- Engineering Calculation G13.18.8*004, “Impact of the Construction of the Independent Spent Fuel Storage Installation in the Unit 2 Excavation Area on the Design Basis Flood Levels for RBS Structures,” Revision 0, dated October 5, 2000
- Engineering Calculation G13.18.1.4*10, “Design of Doors TU66-01 & TU67-H1 and Evaluation of Stresses in G-Tunnel End Wall,” dated July 17, 1991
- Engineering Request (ER) ER-98-0284, “Drawing and Control Data Base upgrade for Catagory 1 Building Hazard Barrier Breach program,” dated December 22, 1999
- Environmental Services Procedure ESP-8-048, “West Creek Inspection,” Revision 5A, performed December 16, 2002
- Abnormal Operating Procedure, AOP-0029, “Severe Weather Operation,” Revision 14B, performed on February 21, 2003

b. <u>Findings</u>

<u>Introduction</u>. The inspectors identified a finding in that the licensee failed to maintain the watertight integrity of two doors at the end of an underground tunnel that opens into the excavation area for Unit 2. This was an unresolved item pending completion of the significance determination process.

<u>Description</u>. On February 21, 2003, during a rainstorm, the inspectors performed a visual inspection of the doors at the end of the G-Tunnel that open into the Unit 2 excavation area. The doors did not appear to be watertight and one was leaking enough water to overflow its opening into the G-Tunnel. The doors at the end of the G-Tunnel are described as watertight to a flooding level of 80 feet above mean sea level (MSL) in the Unit 2 excavation area by engineering design Calculation G13.18.1.4*10, “Design of Doors TU066-01 and TU067-H1 and Evaluation of Stresses in G-Tunnel End Wall,” dated July 17, 1991. Engineering Calculation G13.18.8.0*004, “Impact of the construction of the Independent Spent Fuel Storage Installation in the

Unit 2 Excavation Area on the Design Basis Flood Levels for RBS Structures," dated October 5, 2000, states, in part, that "the maximum water ponding level in the Unit 2 excavation due to the Probable Maximum Flood event is 79.94 [feet MSL]. This is less than the design basis maximum water ponding level of 80 feet [MSL]." During the rainstorm on February 21, 2003, the doors were leaking water across their top seals and Door TU066-01 was leaking water over its opening into the G-Tunnel.

On March 13, 2003 the inspectors inspected the exterior of the doors from the Unit 2 excavation area. The bottoms of the doors were buried under several inches of sand that had pooled in the area during previous rainstorms. Subsequent inspection of the interior side of the doors in the G-Tunnel revealed sand between the doors and their openings. After interviews with maintenance personnel, the inspectors determined that there was no routine maintenance task to inspect or replace the rubber gaskets that form the seals on these doors. In addition, plant maintenance records showed that the sump pump installed to remove rainwater from the Unit 2 excavation area had been out of service since October 30, 2002. A minor maintenance request had been written to investigate the cause. On March 14, 2003, the motor electric power cable was replaced and the sump pump was returned to service. This finding was documented in the licensee's corrective action program as Condition Reports (CRs) CR-RBS-2003-00628 and -00690.

Analysis. The finding was more than minor because it was associated with flood protection measures and degraded the ability to meet the mitigating systems cornerstone objective because it adversely impacted the flooding potential of the G-Tunnel and could challenge the availability of the standby service water system. The Divisions I and II standby cooling tower inlet isolation Valves SWP-MOV055A and -B motor operators and the Division III standby cooling tower inlet isolation Valve SWP-AOV599 control circuits are all situated in the G-Tunnel below elevation 75 feet MSL. The inspectors reviewed this finding using MC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." The result of the phase one screening process was referral of the issue to the regional senior reactor analyst for further review and risk determination.

Enforcement Pending determination of its risk significance, the finding was identified as Unresolved Item (URI) 50-458/2003-03-01, failure to maintain watertight integrity of severe weather doors compromised the availability of a safe shutdown system.

1R08 Inservice Inspection Activities (71111.08)

a. Inspection Scope

.1 Performance of Nondestructive Examination (NDE) Activities Other than Steam Generator Tube Inspections

The inspectors observed the ultrasonic examinations and reviewed the completed documentation for the Feedwater Pipe-to-Valve Weld 1FWS*039A-FW001 and Reactor Vessel Third Shell Course ID Vertical Weld D-13-0001-BM.

The inspectors also reviewed the completed documentation of the following examinations of the feedwater, main steam, and residual heat removal systems.

System	Component/Weld Identification	Examination Method
Feedwater	1FWS-038-FW001, -SW012, -SW015, -SW016	Ultrasonic and Magnetic Particle
Feedwater	1FWS-039A-FW001	Magnetic Particle
Feedwater	1FWS-062-FW004	Ultrasonic and Magnetic Particle
Main Steam	Main Steam Line Welds 1MSS-700A3-FWB07 and -FWB08	Ultrasonic and Magnetic Particle
Residual Heat Removal	RHS-084-A-XI-FW001	Radiography

During the review of each examination, the inspectors verified that the correct NDE procedure was used, examinations and conditions were as specified in the procedure, and test instrumentation or equipment was properly calibrated and within the allowable calibration period. The inspectors also reviewed the documentation and radiographic film to determine if the indications revealed by the examinations were compared against the previous outage examination reports and the American Society of Mechanical Engineers (ASME) Code specified acceptance standards. This review was also performed to determine if the indications were appropriately dispositioned. The NDE certifications of those personnel observed performing examinations or identified during review of completed examination packages were reviewed by the inspectors.

The inspectors reviewed the repair and replacement activities associated with replacement of main steam Line A shutoff valve leakage monitoring connection isolation Valve MSS-V134, controlled under maintenance action item (MAI) 351687, including observation of welding valve-to-pipe Weld XI-FW001. The inspectors reviewed the welding procedure specification and its associated procedure qualification records to assure that Section IX requirements of the ASME Code were met. Additionally, the inspectors observed the control and issuance of welding materials, including storage of coated, low-hydrogen electrodes in temperature controlled storage ovens.

b. Findings

No findings of significance were identified.

.2 Identification and Resolution of Problems

The inspectors reviewed inservice inspection-related condition reports issued during the past 2 years and verified that the licensee identified, evaluated, corrected, and trended problems. In this effort, the inspectors evaluated the effectiveness of the licensee's corrective action process, including the adequacy of the technical resolutions.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

1. On January 28, 2003, the inspectors observed simulator training of an operating crew, as part of the operator requalification training program, to assess licensed operator performance and the training evaluator's critique. Emphasis was placed on observing weekly evaluation exercises of high risk licensed operator actions, operator activities associated with the emergency plan, and lessons learned from industry and plant experiences. In addition, the inspectors compared simulator control panel configurations with the actual control room panels for consistency, including recent modifications implemented in the plant. Simulator training lesson Plan RBS-1-SIM-SMS-0525.01, "Loss of Offsite Source/Drywell Steam Rupture," Revision 1, was reviewed as part of the inspection.

2. On February 3, 2003, the inspectors observed training of an operating crew conducted as just-in-time training in preparation for an infrequently performed evolution, final feedwater temperature reduction. This evolution involved isolating the steam supplies to the high pressure feedwater heaters and the moisture separators. The inspectors also observed the crew briefing conducted by licensee management, reactor engineering, procedure writers, and simulator instructors, a review of lessons learned from industry and plant experiences, and crew performance of the evolution. In addition, the inspectors compared the control room simulator response with the plant response. The simulator training lesson Plan RSTG-40301A.00, "Just in Time Training for ER-RB-2000-0797 Final Feedwater Temperature Reduction," Revision 0, was reviewed as part of the inspection.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

The inspectors reviewed two structure, system, or component (SSC) performance problems to assess the effectiveness of the licensee's maintenance efforts for SSCs scoped under the licensee's maintenance rule program. The inspectors verified the licensee's implementation of the maintenance rule (10 CFR 50.65) for the performance problems reviewed by answering the following questions: (1) was the SSC scoped for monitoring in accordance with 10 CFR 50.65; (2) was the SSC assigned the proper safety significance; (3) were the problems characterized properly; (4) as a result of the problems, was the SSC assigned the proper classification under 10 CFR 50.65; and (5) were the appropriate performance criteria established for the SSC or, when necessary, were appropriate goals set and corrective actions taken to restore the SSC status under the maintenance rule. The following performance problems were evaluated:

- CR-RBS-2002-2088, residual heat removal Pump A minimum flow Valve E12-MOVF064A failed to close when the pump started for heat exchanger flush, reviewed during the week of February 3, 2003

- CR-RBS-2003-0059, 0092, and 0305, all related to broken interlock cables on containment airlock doors, reviewed during the week of March 2, 2003

The following documents were reviewed as part of this assessment:

- NUMARC 93-01, Nuclear Energy Institute Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, Revision 2

- River Bend maintenance rule function list

- River Bend maintenance rule performance criteria list

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

a. Inspection Scope

The inspectors reviewed maintenance activities to verify the performance of assessments of plant risk related to five planned and two emergent maintenance work activities. The inspectors verified: (1) the adequacy of the risk assessments and the accuracy and completeness of the information considered; (2) management of the resultant risk and implementation of work controls and risk management actions; and (3) effective control of emergent work, including prompt reassessment of resultant plant risk.

1. Risk Assessment and Management of Risk

On a routine basis, the inspectors verified performance of risk assessments, in accordance with Administrative Procedure ADM-0096, “Risk Management Program Implementation and On-Line Maintenance Risk Assessment,” Revision 01, for planned maintenance activities and emergent work involving SSCs within the scope of the maintenance rule. Additionally, the inspectors evaluated the risk assessments performed during RFO-11 in accordance with Operations Section Procedure OSP-0037, “Shutdown Outage Protection Plan,” Revision 13. Specific work activities evaluated included planned and emergent work for the weeks of January 20, 2003 (Division I safety system equipment outage), March 3, 2003 (Division II safety system equipment outage), and March 16, 23, and 30, 2003 (RFO-11 risk assessments).

2. Emergent Work Control

During emergent work, the inspectors verified that the licensee took actions to minimize the probability of initiating events, maintained the functional capability of mitigating systems, and maintained barrier integrity. The inspectors also reviewed the emergent work activities to ensure the plant was not placed in an unacceptable configuration. Specific emergent work activities evaluated included:

- Diesel-driven instrument air compressor Dryer IAS-DRY4, out of service on January 28, 2003
- Unplanned single loop operations, February 24-25, 2003

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine Plant Evolutions and Events (71111.14)

a. Inspection Scope

1. Implementation of Final Feedwater Temperature Reduction

The inspectors observed reactor engineering and operations personnel performance during implementation of final feedwater temperature reduction on February 4, 2003. During the inspection, the inspectors reviewed the plan for taking the first point feedwater heaters out of service and removing reheat steam from the moisture separator reheaters, just-in-time training given prior to the evolution, and the prejob briefings given to the control room operating crew and the in-plant operators. The inspectors also reviewed the following documents and procedures used by the operators during the evolution:

- Engineering Request ER-RB-2000-0787, “Final Feedwater Temperature Reduction for Cycle Extension,” dated February 3, 2003

- Simulator Instructor Guide RSTG-40301A.00, “Just In Time Training for ER-RB-2000-0787, Final Feedwater Temperature Reduction,” dated February 3, 2003

- Abnormal Operating Procedure AOP-0007, “Loss of Feedwater Heating (FFWTR Operation),” Revision 21

- System Operating Procedure SOP-0010, “MSR & FW Heaters Extraction Steam and Drains,” Revision 24

2. <u>Manual Reactor Scram in Response to Turbine Hydraulic Oil Leak</u>

The inspectors reviewed operations personnel performance following a manual reactor scram in response to a main turbine control system hydraulic oil leak on February 22, 2003. The inspectors evaluated the initiating causes and operator and equipment response to the reactor scram as documented in CRs-RBS-2003-0639, -0640, -0644, and -0651 and General Operating Procedure GOP-0003, “Scram Recovery,” Revision 15, presented to the onsite safety review committee on February 22, 2003. In addition, the inspectors reviewed operator logs and plant computer data to determine what had occurred and that operators responded in accordance with plant procedures and training. The inspectors also reviewed Abnormal Operating Procedure AOP-0001, “Reactor Scram,” Revision 19, used by the operators, during the event.

3. <u>Trip of Reactor Recirculation Pump B During Upshift to Fast Speed</u>

The inspectors reviewed and observed operations and reactor engineering personnel performance following an unexpected trip of the reactor recirculation Pump B trip while being upshifted to fast speed on February 24, 2003. The inspectors evaluated the determination of the initiating causes of the failure of the ground fault trip of the reactor recirculation pump motor Breaker 4B trip, as documented in CR-RBS-2003-0670. In addition, the inspectors reviewed operator logs and plant computer data to determine what occurred and that operators responded in accordance with plant procedures and training. The inspectors reviewed the following procedures used by the operators during the event:

- GOP-0004, “Single Loop Operations,” Revision 16A

- SOP-0003, “Reactor Recirculation System,” Revision 25

- AOP-0024, “Thermal Hydraulic Stability Control,” Revision 20A

4. <u>Reactor Fuel Element Leak</u>

The inspectors reviewed and observed operations and reactor engineering personnel performance following an unexpected rise in off-gas pretreatment radiation levels on December 2, 2002. Subsequent radiochemistry analysis of off-gas pretreatment and reactor coolant samples indicated that there was an apparent reactor fuel element leak. The inspectors observed portions of and reviewed the results of two fuel power suppression tests performed during December 2002 and documented their observations in NRC Inspection Report 50-458/02-04, Section 1R14.

During January and February 2003, the inspectors reviewed reactor engineering core performance monitoring and observed fuel integrity monitoring team actions to track fuel performance for the rest of the operating cycle. During RFO-11 from March 16 to April 9, 2003, the inspectors observed fuel sipping operations and reviewed the results and the licensee's corrective actions for the five leaking fuel bundles. In addition, the inspectors observed reactor fuel engineering planning sessions to understand and evaluate the licensee's resolution of the problem and reconstitution of the reactor core for the upcoming operating cycle.

The inspectors reviewed the following procedures and documents, during their inspection:

- CR-RBS-20020-1911, Possible failed reactor fuel condition
- Administrative Procedure ADM-0084, "Fuel Integrity Monitoring Program and Failed Fuel Action Plan," Revision 2
- Reactor engineering Procedure REP-0057, "Power suppression Testing," Revision 4
- Onsite Safety Review Committee Presentation, "RFO-11 Fuel Inspection Results Summary," presented April 7, 2003
- Onsite Safety Review Committee Presentation, "RBS Revised Cycle 12 Reload Review," presented April 7, 2003
- River Bend Station Cycle 12 Core Operating Limits Report, dated April 7, 2003

b. <u>Findings</u>

No findings of significance were identified.

1R15 <u>Operability Evaluations (71111.15)</u>

a. <u>Inspection Scope</u>

The inspectors reviewed six operability evaluations performed by the licensee for risk significant systems to determine that the operability was justified, such that availability was assured, and no unrecognized increase in risk has occurred. Specific areas evaluated included: (1) the technical adequacy of the evaluation; (2) whether other existing degraded conditions were considered; and (3) if operability was based on compensatory measures, were these measures in place and would they work. The inspectors also reviewed Nuclear Procedure RBNP-078, "Operability Determinations," Revision 6, and Operations Section Procedure OSP-0040, "LCO Tracking and Safety Function Determination Program," Revision 7.

- CR-RBS-2002-2088, Residual heat removal Pump A minimum flow Valve E12-MOVF064A failed to close when pump started for heat exchanger flush, reviewed on December 30, 2003

- CR-RBS-2003-0180, Control room fresh air system operable with remote intake radiation Monitor RMS-RE14B inoperable, reviewed on January 22, 2003

- ER-RB-2001-0381, Engineering evaluation for support of operability determination of reactor core isolation cooling system without Division I standby service water supply to auxiliary building unit Cooler HVR-UC6, reviewed on February 6, 2003

- CR-RBS-2003-0496, Required rework on reactor core isolation cooling suction line check Valve E51-VF011, reviewed on February 14, 2003

- CR-RBS-2003-0558, Division I hydrogen igniters reading fails to meet acceptance criteria during postmaintenance testing, reviewed on March 5, 2003

- Operations Section Procedure OSP-0041, "Alternate Decay Heat Removal," Revision 6, change notice for suppression pool cooling heat exchanger capabilities for alternate decay heat removal, dated March 18, 2003, reviewed March 20, 2003

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

1. Biennial Inspection of Permanent Plant Modifications

The inspectors reviewed 10 permanent plant modification packages and associated documentation, such as review screens and safety evaluations, to verify that they were performed in accordance with regulatory requirements and plant procedures. The inspectors also reviewed procedures governing plant modifications to evaluate the effectiveness of the programs for implementing modifications to risk-significant SSCs such that these changes did not adversely affect the design and licensing basis of the facility. Permanent plant modifications and procedures reviewed are listed in the attachment to this report.

The inspectors interviewed the cognizant engineers for selected modifications as to their understanding of the modification packages.

The inspectors evaluated the effectiveness of the licensee's corrective action process to identify and correct problems associated with the performance of permanent plant modifications. In this effort, the inspectors reviewed the corrective action documents listed in the attachment to this report.

2. Low Pressure Core Spray Minimum Flow Valve Timer Modification

As part of the inspection into the failure of residual heat removal Pump A minimum flow Valve E12-MOVF064A to close when the pump was started for heat exchanger flush on December 28, 2002, the inspectors reviewed plant Modification ER-99-0450, "Install Time Delay Drop Out Relay to Prevent Sudden Motor Reversal and Subsequent Breaker Trip of E12-MOVF064A(B) and E21-MOVF011," dated October 21, 1999. See Section 4OA2.2.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed the postmaintenance testing requirements specified for five MAIs listed below to ensure that testing activities were adequate to verify system operability and functional capability:

- MAI 363701, Troubleshoot station blackout diesel generator low coolant temperature indication, reviewed January 8, 2003
- MAI 366169, Post system outage test of Division I emergency diesel generator, reviewed during the week of January 27, 2003
- MAI 363701, Change high voltage taps of Division I hydrogen igniters power Transformer HCS-XD01A, reviewed February 20, 2003
- MAI 369992, Auxiliary building unit Cooler HVR-UC11B, Breaker EJS-SWG2B-A, tripped as required but did not reclose as required during Division II emergency core cooling systems testing, reviewed March 24, 2003
- MAI 369993, Control room ventilation isolation Dampers HVC-AOD51B, HVC-AOD52B, and HVC-MOV1B did not close as required during Division II emergency core cooling systems testing, reviewed March 24, 2003

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors observed licensee outage planning and execution of RFO-11. The inspectors' review included scheduling, training, outage configuration management, decay heat removal operation and management, reactivity controls, inventory controls, tag-out and clearance activities, foreign material exclusion management, and fuel movement and storage. Specific activities monitored included:

- Outage risk assessment team report to on-site safety review committee for RFO-11

- Portions of reactor shutdown, including downshift of reactor recirculation pumps to slow speed, reactor scram, cooldown, and transition to shutdown cooling

- Fuel movement and in-vessel fuel sipping operations

- Jet pump in-vessel visual inspection report to on-site safety review committee

- Daily assessment of outage risk

- Evaluation of flow accelerated corrosion inspection of reactor feedwater injection piping

- Reactor startup and approach to critical

- Postoutage main turbine overspeed test

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors assessed, by witnessing and reviewing test data, 10 risk-significant system and component surveillance tests against Technical Specification, USAR, and procedure requirements. The inspectors ensured that surveillance tests demonstrated that the systems were capable of performing their intended safety functions and provided operational readiness. The inspectors specifically evaluated surveillance tests for preconditioning, clear acceptance criteria, range, accuracy and current calibration of test equipment and reviewed whether equipment was properly restored at the completion of the testing. The inspectors reviewed and or observed the following surveillance tests and surveillance test procedures (STP):

- STP-402-0202, “Main Control Room Air Conditioning Train B Operability Test,” Revision 5, performed on January 9, 2003
- STP-309-0201, “Division I Diesel Generator Operability Test,” Revision 23, performed on January 25, 2003
- STP-209-6310, “RCIC Quarterly Pump and Valve Operability Test,” Revision 22 performed on February 12, 2003
- STP-254-1401, “Division I Hydrogen Igniter Train Current and Voltage Check,” Revision 3, performed on February 19, 2003
- STP-052-3701, “Control Rod Scram Testing,” Revision 20, performed after manual reactor scam on February 22, 2003
- STP-109-6802, “MSIV Cold Shutdown Full Stroke Operability Test,” Revision 2, performed on March 27, 2003
- STP-051-4262, “RPS-Main Steam Isolation Valve-Closure Chanel Calibration and LSFT,” Revision 13A, performed on March 29, 2003
- STP-309-0602, “Division II ECCS Test,” Revision 18, performed on March 17, 2003
- STP-309-0201, “Division I Diesel Generator Operability Test,” Revision 24, performed on April 4, 2003
- STP-050-3601, “Shutdown Margin Demonstration,” Revision 24, performed on April 10, 2003

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

During the week of March 24, 2003, the inspectors reviewed Temporary Alteration 2002-0026-02 made to provide alternate power to plant fire protection alarm panels while their normal power supplies were out of service for planned maintenance. Specifically, the inspectors: (1) reviewed the temporary modification and its associated 10 CFR 50.59 screening against the system’s design basis documentation, including the USAR and Technical Specifications; (2) verified that the installation of the temporary modification was consistent with the modification documents; and (3) reviewed the postinstallation test results to confirm that the actual impact of the temporary modification on the affected system had been adequately verified.

b. Findings

No findings of significance were identified.

Emergency Preparedness [EP]

1EP2 Alert Notification System Testing (71114.02)

a. Inspection Scope

The licensee's siren testing program was compared with the guidance of NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1, and Federal Emergency Management Agency document REP-10, "Guide for the Evaluation of Alert and Notification Systems for Nuclear Power Plant." The inspector reviewed siren failure trend data and testing records for Calendar Year 2002. The inspector also reviewed Emergency Preparedness Procedure EPP 2-701, "ANS Siren Testing and Maintenance," Revision 17, and Self-Assessment LO-RLO-2002-00133, November 20, 2002. The inspector reviewed the approved Federal Emergency Management Agency Alert Notification System description and compared the approved configuration and testing programs to the current design. The inspector also reviewed condition reports related to siren failures and related MAI requests, as well as observed the conduct of the January 29, 2003, weekly silent siren system test.

b. Findings

No findings of significance were identified.

1EP3 Emergency Response Organization Augmentation Testing (71114.03)

a. Inspection Scope

The inspector discussed with the licensee changes made in the installed systems and testing programs for automatic phone dialing and paging systems during Calendar Year 2002 to evaluate the licensee's continued ability to staff emergency response facilities in accordance with the licensee emergency plan and the requirements of 10 CFR Part 50, Appendix E. The inspector reviewed the results of quarterly pager tests conducted during Calendar Year 2002 and the following data and procedures:

- EIP 2-006, "Notifications," Revision 30
- EIP 2-502, "Emergency Communications Equipment Testing," Revision 20
- PL-140, "Emergency Response Organization Respiratory Protection Guidelines," Revision 2
- P- PL-026, "Respiratory Protection Policy," Revision 3

- TQ-110, “Emergency Preparedness Training Program,” Revision 1
- TQ-201, “Systematic Approach to Training Process,” Revision 1
- EPP 2-202, “Emergency Response Organization,” Revision 11

b. Findings

No findings of significance were identified.

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04)

a. Inspection Scope

The inspector reviewed Revision 12 to EIP-2-001, “Classification of Emergencies,” and the associated 10 CFR 50.54(q) analysis supporting the revision to determine if the revision decreased the effectiveness of the emergency plan.

b. Findings

No findings of significance were identified.

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies (71114.05)

a. Inspection Scope

The inspector reviewed the following documents related to the licensee’s corrective action program to determine the licensee’s ability to identify and correct problems in accordance with the requirements of 10 CFR 50.47(b)(14) and 10 CFR Part 50, Appendix E:

- LI-102, “Corrective Action Process,” Revision 2
- EIP 2-020, “Emergency Operations Facility,” Revision 25
- 2003 River Bend Station Assessment and benchmarking schedule
- ERO Team Bravo Dress Rehearsal Drill Self Assessment, May 22, 2002
- QA-7-2002-RBS-1, “River Bend Station Quality Assurance Audit of the Emergency Plan,” April 8 through May 6, 2002
- LO-RLO-2002-00018, “Drill Trend Assessment”
- LO-RLO-2002-00080, “ERO Team Charlie Drill”
- LO-RLO-2002-00034, “ERO Team Alpha Drill March 6, 2002”

- LO-RLO-2002-00059, “ERO Team Bravo Drill June 18, 2002”
- LO-RLO-2002-00123, “ERO Team Delta Drill”
- LO-RLO-2002-00133, “Assessment of Industry Emergency Planning Issues”
- Summary of emergency preparedness related condition reports for Calendar Year 2002

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors observed the emergency response organization team training drill conducted on February 18, 2002, to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation development activities. The inspectors also evaluated the licensee assessment of classification, notification, and protective action recommendation development during the drill in accordance with plant procedures and NRC guidelines. The following procedures and documents were reviewed during the inspection:

- EIP-2-001, “Classification of Emergencies,” Revision 11
- EIP-2-006, “Notifications,” Revision 27
- EIP-2-007, “Protective Action Guidelines Recommendations,” Revision 18
- EP-M-03-011, “February 18, 2003 Drill Evaluation Report, ERO Team A,” dated March 5, 2003

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY
Cornerstone: Occupational Radiation Safety [OS]

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope

To review and assess the licensee's performance in implementing physical and administrative controls for airborne radioactivity areas, radiation areas, and high radiation areas, the inspector interviewed supervisors, radiation workers, and radiation protection personnel involved in high dose rate and high exposure jobs in RFO-11. The inspector conducted plant walkdowns within the controlled access area and conducted independent radiation surveys of selected work areas. The following items were reviewed and compared with regulatory requirements:

- Area postings, radiation work permits (RWPs), radiological surveys, and other controls for airborne radioactivity areas, radiation areas, and high radiation areas
- Locked high radiation area key control
- Internal dose assessment for exposures exceeding 50 mrem committed effective dose equivalent
- Setting, use, and response of electronic personal dosimeter alarms
- Prejob briefings for work in locked high radiation areas
- Conduct of work by radiation protection technicians and radiation workers in areas with the potential for high radiation dose and the associated RWPs, radiological surveys, and controls for the work
- Dosimetry placement when work involved a significant dose gradient
- Controls involved with the storage of highly radioactive items in the spent fuel pool
- Audits and self-assessments (Radiation Protection Self-Assessment, July 28 through August 2, 2002) involving high radiation area controls and staff performance
- Selected corrective action documents involving high radiation area incidents, radiation protection technician and radiation worker errors, and repetitive or significant individual deficiencies (CR-RBS-2002-01380, CR-RBS-2002-01896, CR-RBS-2003-229, CR-RBS-2003-00653, and CR-RBS-2003-00874)

The inspector also reviewed the circumstances associated with a containment contamination event that occurred on March 17, 2003. The event occurred during reactor disassembly in preparation for RFO-11 and revealed itself when workers alarmed the personnel contamination monitors as they attempted to exit the controlled access area. The licensee conducted whole-body counts on 55 workers to determine the dose consequences. Preliminary dose estimates indicated that the highest individual committed effective dose equivalent was less than 73 millirems. As a result of the event, the licensee established a significant event review team (SERT) to determine the chronology of events and the root cause. Although, the SERT had not finalized its report, the inspector interviewed members to determine the facts associated with the event.

b. Findings

1. Introduction. The licensee failed to develop a sufficiently detailed work plan for the decontamination of the reactor vessel bellows.

Description. Following the removal of the drywell head and the insulation frame on March 17, 2003, the licensee began the decontamination of the reactor vessel bellows, a concentric groove surrounding the reactor pressure vessel flange. Licensee personnel attempted to keep the bellows wet to prevent radioactivity from becoming airborne. However, the bellows temperature was elevated and caused the water to evaporate. The licensee’s decontamination plan did not address a minimum time necessary for the bellows to cool naturally. A hydrolaser was used to clean the bellows. An attachment called a rotary surface cleaner was fitted to the nozzle of the hydrolaser to shield the spray from the hydrolaser and prevent the spread of airborne contamination. However, because of the cross-sectional shape of this particular bellows, the attachment did not form a flush fit with the bottom of bellows. This technique had not been used previously by this licensee, and the licensee’s decontamination plan did not consider that the bellows cross-section would not accommodate the rotary surface cleaner. The decontamination worker using the hydrolaser was not briefed on the job objectives or on precautions to prevent the spread of contamination. The SERT concluded that using the hydrolaser in this manner caused radioactive contamination to become airborne.

Analysis. This self-revealing finding was greater than minor because it was associated with one of the Occupational Radiation Safety Cornerstone attributes (exposure/contamination control) and the finding affected the associated cornerstone objective (to ensure the adequate protection of worker health and safety from exposure to radiation from radioactive material). The inspector processed the violation through the Occupational Radiation Protection Significance Determination Process because the occurrence involved potential doses (resulting from actions or conditions contrary to licensee procedures) which could have been significantly greater as a result of a single, minor, reasonable alteration of the circumstances. However, because the violation was not an as low as reasonably achievable (ALARA) finding, there was no personnel overexposure, there was no substantial potential for personnel overexposure, and the finding did not compromise the licensee’s ability to assess dose, the violation had no more than very low safety significance.

Enforcement. Technical Specification 5.4.1.a requires that procedures be established, implemented, and maintained covering the applicable procedures in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978. Regulatory Guide 1.33, Appendix A, Section 7.e.9, lists procedures for implementation of an ALARA program. Administrative Procedure ADM-0039, “ALARA Program,” Revision 09, Section 4.16, stated that Planning and Scheduling/Outage Groups were responsible for providing detailed work plans to allow for ALARA Planning to designate adequate radiological controls. However, the outage group responsible for reactor cavity and upper pool decontamination did not provide a work plan that was detailed enough to provide guidance on maintaining the bellows surface wet, using a hydrolaser with a rotary surface cleaner, or briefing the individual using the hydrolaser. Because the failure to provide detailed work plans for decontamination of the reactor vessel bellows was of very low safety significance and was documented in the licensee's corrective action program as CR-RBS-2003-1016, Corrective Action Item 10, this violation is being treated as a noncited violation (NCV), consistent with Section VI.A of the NRC Enforcement Policy (NCV 50-458/2003-03-02).

2. Introduction. In three examples, the licensee failed to survey or evaluate radiological hazards associated with reactor vessel bellows decontamination.

Description. In the first example, a radiation protection technician performed a contamination survey of the reactor vessel bellows before the decontamination activities of March 17, 2003. However, the contamination survey was conducted on the upper wall of the bellows instead of the bottom. The licensee acknowledged that the bottom of the bellows typically was the location with the highest contamination levels. In the second example, continuous air monitors were present on the refueling floor but not in the vicinity of the bellows contamination work. A representative grab air sample was not taken to document the airborne radioactivity present during decontamination of the bellows. In the third example, even after the continuous air monitor began to alarm, radiation protection personnel failed to analyze representative grab air samples before allowing personnel to continue to work in the containment building. Air samples were analyzed 5-6 hours after the initiation of the event and confirmed that airborne radioactivity levels on the refueling floor equaled 3.52 times the derived air concentration limits for Cobalt-60 and Manganese-54 particulate. The licensee's analysis confirmed that no alpha emitting radionuclides were present.

Analysis. This self-revealing finding was greater than minor because it was associated with one of the Occupational Radiation Safety Cornerstone attributes (exposure/contamination control) and the finding affected the associated cornerstone objective (to ensure the adequate protection of worker health and safety from exposure to radiation from radioactive material). The inspector processed the violation through the Occupational Radiation Protection Significance Determination Process because the occurrence involved potential doses (resulting from actions or conditions contrary to licensee procedures) which could have been significantly greater as a result of a single, minor, reasonable alteration of the circumstances. However, because the violation was not an ALARA finding, there was no personnel overexposure, there was no substantial potential for personnel overexposure, and the finding did not compromise the licensee's ability to assess dose, the violation had no more than very low safety significance.

Enforcement. 10 CFR 20.1501(a) states that each licensee shall make, or cause to be made, surveys that: (1) may be necessary for the licensee to comply with the regulations in this part; and (2) are reasonable under the circumstances to evaluate the magnitude and extent of radiation levels, concentrations or quantities of radioactive material, and the potential radiological hazards. 10 CFR 20.1003 defines survey as an evaluation of the radiological conditions and potential hazards incident to the production, use, transfer, release, disposal, or presence of radioactive material or other sources of radiation. When appropriate, such an evaluation includes a physical survey of the location of radioactive material and measurements or calculations of levels of radiation or concentrations or quantities of radioactive material present. Under the circumstances, a survey would have been necessary to verify compliance with 10 CFR 20.1201, “Occupational Dose Limits for Adults,” or 10 CFR 20.1902(d), “Posting of Airborne Radioactivity Areas.” However, the licensee failed to conduct adequate surveys on three occasions. The licensee failed to evaluate the highest concentration of radioactive contamination on the reactor vessel bellows. Additionally, the licensee failed to evaluate the airborne radioactivity in the immediate vicinity of the bellows contamination and, later, the airborne radioactivity levels throughout the containment building. Because the failure to survey or evaluate radiological hazards associated with reactor vessel bellows decontamination was of very low safety significance and was documented in the licensee’s corrective action program as CR-RBS-2003-1178, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 50-458/2003-03-03).

3. Introduction. The licensee failed to post the reactor containment building as an airborne radioactivity area.

Description. On March 17, 2003, following the decontamination of the reactor vessel bellows, air samples taken at 5:25 p.m. confirmed that airborne radioactivity levels on the refueling floor equaled 3.52 times the derived air concentration limits for Cobalt-60 and Manganese-54 particulate. However, radiation protection personnel responsible for oversight of work in the containment building acknowledged that the refueling floor was not posted as an airborne radioactivity area until approximately 10:30 p.m. This was after the contamination event was revealed when contaminated workers attempted to leave the controlled access area during the shift change.

Analysis. This self-revealing finding was greater than minor because it was associated with one of the Occupational Radiation Safety Cornerstone attributes (exposure/contamination control) and the finding affected the associated cornerstone objective (to ensure the adequate protection of worker health and safety from exposure to radiation from radioactive material). The inspector processed the violation through the Occupational Radiation Protection Significance Determination Process because the occurrence involved potential doses (resulting from actions or conditions contrary to licensee procedures) which could have been significantly greater as a result of a single, minor, reasonable alteration of the circumstances. However, because the violation was not an ALARA finding, there was no personnel overexposure, there was no substantial potential for personnel overexposure, and the finding did not compromise the licensee’s ability to assess dose, the violation had no more than very low safety significance.

Enforcement. 10 CFR 1902(d) requires that the licensee post each airborne radioactivity area with a conspicuous sign or signs bearing the radiation symbol and the words "CAUTION, AIRBORNE RADIOACTIVITY AREA" or "DANGER, AIRBORNE RADIOACTIVITY AREA." 10 CFR 20.1003 defines an airborne radioactivity area as a room, enclosure, or area in which airborne radioactive materials, composed wholly or partly of licensed material, exist in concentrations in excess of the derived air concentrations specified in 10 CFR Part 20, Appendix B. However, the licensee failed to post the refueling floor as an airborne radioactivity area from 5:25 p.m. to 10:30 p.m., when the airborne radioactivity levels were as high as 3.52 times the derived airborne concentration limits for the radionuclides present. Because the failure to post the refueling floor was of very low safety significance and was documented in the licensee's corrective action program as CR-RBS-2003-1205, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 50-458/2003-03-04).

4. Introduction. The licensee failed to inform workers of changes in radiological conditions and in precautions to minimize exposure to radiation.

   Description. On March 17, 2003, after the continuous air monitors alarmed and airborne radioactivity concentrations in the containment building exceeded the 10 CFR Part 20, Appendix B, limits, work continued unchecked. Radiation protection personnel did not exercise their authority to stop work and evacuate personnel. Alternately, radiation protection personnel did not inform workers that radiation conditions had changed and that additional precautions were necessary to minimize exposure to radiation.

   Analysis. This self-revealing finding was greater than minor because it was associated with one of the Occupational Radiation Safety Cornerstone attributes (exposure/contamination control) and the finding affected the associated cornerstone objective (to ensure the adequate protection of worker health and safety from exposure to radiation from radioactive material). The inspector processed the violation through the Occupational Radiation Protection Significance Determination Process because the occurrence involved potential doses (resulting from actions or conditions contrary to licensee procedures) which could have been significantly greater as a result of a single, minor, reasonable alteration of the circumstances. However, because the violation was not an ALARA finding, there was no personnel overexposure, there was no substantial potential for personnel overexposure, and the finding did not compromise the licensee's ability to assess dose, the violation had no more than very low safety significance.

   Enforcement. 10 CFR 19.12 requires that all individuals who in the course of employment are likely to receive in a year an occupational dose in excess of 100 millirems shall be kept informed of the transfer or use of radioactive material and in precautions to minimize exposure. However, when radiological hazards increased with the rise of airborne radioactivity levels, the licensee failed to inform the workers in the containment building of the change in conditions and the precautions necessary to minimize exposure to radiation. Because the failure to instruct workers was of very low safety significance and was documented in the licensee's corrective action program as CR-RBS-2003-1016, Corrective Action 14, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 50-458/2003-03-05).

5. Introduction. The licensee failed to control an area with radiation levels greater than 1000 millirems per hour at 30 centimeters as a locked high radiation area.

   Description. After a plant scram on September 18, 2002, a worker entered the reactor core isolation cooling area on the 95-foot elevation of the auxiliary building and received an electronic dosimeter dose rate alarm. A crud burst resulting from a transient that occurred approximately 3 hours previously caused the dose levels in the area entered by the worker to increase to 1000 millirems per hour. The area was controlled as a high radiation area; however, it was not locked or continuously guarded to prevent unauthorized entry. Historically, the plant has experienced crud bursts and elevated dose rates under similar conditions. Therefore, the increase in dose rate should have been anticipated and evaluated.

Analysis. This self-revealing finding was greater than minor because it was associated with one of the Occupational Radiation Safety Cornerstone attributes (exposure/contamination control) and the finding affected the associated cornerstone objective (to ensure the adequate protection of worker health and safety from exposure to radiation from radioactive material). The inspector processed the violation through the Occupational Radiation Protection Significance Determination Process because the occurrence involved potential doses (resulting from actions or conditions contrary to licensee procedures) which could have been significantly greater as a result of a single, minor, reasonable alteration of the circumstances. However, because the violation was not an ALARA finding, there was no personnel overexposure, there was no substantial potential for personnel overexposure, and the finding did not compromise the licensee's ability to assess dose, the violation had no more than very low safety significance.

Enforcement. Technical Specification 5.7.2 requires that areas with radiation levels equal to or greater than 1000 millirems per hour be provided with locked or continuously guarded doors to prevent unauthorized entry. However, following a plant scram on September 18, 2002, the reactor core isolation cooling area on the 95-foot elevation of the auxiliary building had radiation dose rates greater than or equal to 1000 millirems per hour and was not provided with locked or continuously guarded doors to prevent unauthorized entry. Because the failure to control the reactor core isolation cooling area as a locked high radiation area was of very low safety significance and was documented in the licensee's corrective action program as CR-RBS-2002-1380, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 50-458/2003-03-06).

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

To assess the licensee's performance in implementing physical and administrative controls for airborne radioactivity areas, radiation areas, and high radiation areas, the inspector interviewed radiation protection personnel and radiation workers involved in high dose rate, high collective exposures, and airborne area work activities for radiation worker practices and work activity results. The inspector observed high dose work activity involving Drywell valve work associated with RWPs 2003-1926, "MOV Refurbishment/PM's/Signature Test/Inspections," and 2003-1935, "Drywell Valve Maintenance," to determine if personnel work practices were ALARA and within regulatory and procedural compliance.

The inspector interviewed radiation protection staff and other radiation workers to determine the level of planning, communication, integration, and supervision of ALARA practices into work activities and/or packages. The inspector reviewed information provided to the radiation protection group for accuracy.

The following items were reviewed and compared with regulatory requirements to assess the licensee's program to maintain occupational exposures ALARA:

- Plant collective exposure history for the past 3 years, current exposure trends, source term measurements, and 3-year rolling average dose information
- ALARA program procedures
- Processes, methodology, and bases used to estimate, justify, adjust, track, and evaluate exposures
- Five ALARA prejob and in-progress job reviews and associated RWP packages from RFO-11 activities which resulted in the highest personnel collective exposures (RWP 2003-1800, "RF11 Refueling Activities"; RWP 2003-1917, "Repair undervessel carousel and replace/rebuild/leak test 15 CRDM's, including all support work"; RWP 2003-1936, "RF11 Temporary Shielding in the Drywell"; RWP 2003-1950, "Scaffolding in the Drywell in RF11"; and RWP 2003-1951, "Insulation removal/replacement in the Drywell in RF11")
- The use and result of administrative and engineering controls to achieve dose reductions
- Plant source term evaluation and control strategy/program
- Hot spot tracking and reduction program
- Temporary shielding program
- 2001 Annual ALARA Program Report; Benchmark Report "Source Term Reduction" (LO-RLO-2003-00038); and quality surveillance reports (QS-2002-RBS-015 and QS-2002-RBS-018) reviewing ALARA performance
- ALARA Committee meeting minutes and presentations
- Selected corrective action documentation involving exposure tracking, higher than planned exposure levels, radiation worker and radiation protection practices, and repetitive deficiencies since the last inspection in this area (CR-RBS-2002-0382, -0793, -1190, -1382; and -2003-1213, -1256, -1296)

b. Findings

.1 Introduction. The inspector identified collective doses for selected work activities performed during RFO-11 that were not maintained ALARA. The inspector determined that the work activity associated with RWP 2003-1929, "RFO-11 Recirculation Pump Work," Task 1, exceeded 5 person-rem and exceeded the dose estimation by more than 50 percent.

Description. During a review of RWP packages and the current accumulated dose for in-progress outage work, the inspector identified performance deficiencies in the work activity dose estimate adjustments. The licensee adjusted dose estimates to account for worker inefficiencies and inadequate planning, scheduling, and supervisory oversight.

For example, RWP 2003-1929, Revision 1, Task 1, had an increase in the work activity dose estimate from 10.84 rem to 15.37 rem. The RWP in-progress reviews and the work activity lessons learned document provided the following examples of inadequate planning and scheduling:

- “Best” performance estimates used were not feasible or realistic.
- There was poor accuracy in estimating the duration of snubber and strut work, instrument removal, and hoist installation.
- There was a lack of preplanning consideration to use a pump to assist in system draindown and to remove the instrument terminal box from the motor before attempting to move the motor through the drywell door.
- Workers removed a motor coupling in a higher dose area for 6 hours before moving the part to a lower dose area to continue work.
- The job was started prior to shielding being installed.

Worker inefficiencies and inadequate supervisory oversight were also documented as performance deficiencies. Worker inefficiencies added an additional 6 person-hours to a work activity when inexperienced workers performed the addition of oil and testing of the oil switches for the pump. In addition, inadequate supervisory oversight was documented because craft supervisors and radiation protection personnel allowed craftsmen to begin work before shielding at the jobsite was completed. The lessons learned document stated “dose was wasted prior to the shielding installation when craftsmen entered the drywell to familiarize themselves with the jobsite and the tasks to be performed.”

The RWP in-progress reviews and lessons learned document described workers encountering various mechanical problems and difficulties. While some of these job difficulties and problems may be justifiable reasons for a work activity dose increase, the bases for these adjustments were not clearly quantified and separated from the other dose adjustments that identified as-found performance deficiencies.

In regard to RWP 2003-1929, Revision 2, Task 1, the work activity dose estimate increased from 15.37 rem to 20.34 rem. The reasons documented within the Revision 2 RWP in-progress review were a restatement of the reasons mentioned in the Revision 1 in-progress reviews and the lessons learned document. The inspector determined that there was no documentation of additional emergent work or work scope changes to justify the dose estimate increase.

Consequently, the actual collective dose for the RWP 2003-1929, Task 1, work activity on March 28, 2003, exceeded 5 rem (16.54 rem) and exceeded the original dose estimate (10.84 rem) by more than 50 percent.

Analysis. The failure to maintain collective doses ALARA is a performance deficiency. This finding was more than minor because it is associated with the Occupational Radiation Safety Cornerstone attribute (ALARA planning/projected dose) and affected the associated cornerstone objective (to ensure adequate protection of worker health and safety from exposure to radiation). This occurrence involved worker inefficiencies, inadequate planning, scheduling, and supervisory oversight which resulted in unplanned, unintended occupational collective dose for a work activity. When processed through the Occupational Radiation Safety Significance Determination Process, this finding was found to have no more than very low safety significance because the finding was an ALARA planning issue, the licensee's 3-year rolling average collective dose was greater than 240 person-rem, the actual dose for the work activity was not more than 25 person-rem, and there were no more than four occurrences. The finding was documented in the licensee's corrective action program as CR-RBS-2003-1213 (FIN 50-458/2003-03-07).

.2 Introduction. The inspector identified collective doses for selected work activities performed during RFO-11 that were not maintained ALARA. The inspector determined that the work activity associated with RWP 2003-1935, "Drywell Valve Maintenance, to include Repacks and Support Work," exceeded 5 person-rem and exceeded the dose estimation by more than 50 percent.

Description. During a review of RWP packages and the current accumulated dose for in-progress outage work, the inspector identified performance deficiencies in the work activity dose estimate adjustments. The licensee adjusted dose estimates to account for inadequate planning and worker inefficiencies.

For example, from a review of the RWP 2003-1935, Revision 0, in-progress review, the inspector concluded the review did not provide an adequate basis to increase the work activity dose estimate from 4.16 rem to 9.43 rem. The in-progress review documented inadequate work planning, such as planners providing "best case" performance man-hours which were determined by the licensee to not be feasible or realistic and a water-pik not being used in some cases for packing extraction as required in the work activity plan. Of the 53 valves listed in the RWP, more than half of the valve work activities (37) received an increase in person-hours for unspecified reasons. This increase accounts for approximately 3.2 person-rem. In addition, a scope increase to include washers and stroking of the valves for in-service testing was not included in the initial work activity planning.

Worker inefficiency was an additional documented performance deficiency. The in-progress review noted that new workers were taking longer to locate assigned valves. Therefore, the licensee reorganized work crews to use more experienced personnel in support of RFO-11 valve work.

The RWP in-progress review documented increased dose rates and expanded work scope, as well as workers encountering various mechanical problems and difficulties. However, while the expanded work scope and increased dose rates were justifiable reasons for a work activity dose increase, the inspector concluded that the specific contributions for these adjustments were not clearly quantified and separated from the other performance deficiencies mentioned within the in-progress review.

Consequently, the actual collective dose for the RWP 2003-1935 work activity on March 28, 2003, exceeded 5 rem (6.95 rem) and exceeded the original dose estimate (4.16 rem) by more than 50 percent.

Analysis. The failure to maintain collective doses ALARA is a performance deficiency. This finding was more than minor because it is associated with the Occupational Radiation Safety Cornerstone attribute (ALARA planning/projected dose) and affected the associated cornerstone objective (to ensure adequate protection of worker health and safety from exposure to radiation). This occurrence involved worker inefficiencies and inadequate planning which resulted in unplanned, unintended occupational collective dose for a work activity. When processed through the Occupational Radiation Safety Significance Determination Process, this finding was found to have no more than very low safety significance because the finding was an ALARA planning issue, the licensee's 3-year rolling average collective dose was greater than 240 person-rem, the actual dose for the work activity was not more than 25 person-rem, and there were no more than four occurrences. The licensee has entered this finding into their corrective action program as CR-2003-1213 (FIN 50-458/2003-03-08).

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

1. Initiating Event Cornerstone

The inspectors verified the accuracy and completeness of the data used to calculate and report performance indicator data for the second quarter of 2002 through the first quarter of 2003. The inspectors used Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 2, as guidance and interviewed licensee personnel responsible for compiling the information. The following performance indicators were reviewed:

- Unplanned scrams per 7,000 critical hours
- Scrams with a loss of normal heat removal per 12 quarters
- Unplanned power changes per 7,000 critical hours

2. Emergency Preparedness Cornerstone

The inspectors verified the accuracy and completeness of the data used to calculate and report performance indicator data for the second and third quarters of Calendar Year 2002. The inspectors used NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 2, as guidance and interviewed licensee personnel responsible for compiling the information. The following performance indicators were reviewed:

- Emergency Plan Drill/Exercise Performance
- Emergency Response Organization Drill Participation
- Alert and Notification System Reliability

The inspector reviewed the following documents related to the emergency plan drill and exercise performance, emergency response organization drill participation, and alert and notification system reliability performance indicators in order to verify the licensee's reported data:

- Procedure LI-107, "NRC Performance Indicator Process," Revision 1

- Procedure EIP 2-007, "Protection Action Recommendation Guidelines," Revision 18

- EPP 2-703, "Performance Indicators," Revision 2

- Drill schedules for Calendar Year 2002

- Drill scenarios, notification forms, and participant logs for drills conducted during the second through third quarters of Calendar Year 2002

- Drill evaluation records for a sampling of drills conducted during the second through third quarters of Calendar Year 2002

- Performance indicator summary sheets and reports

- Emergency response organization rosters for the second through third quarters of Calendar Year 2002

- List of key emergency response organization positions

- Drill participation and qualification training records for a sample of eight key responders for drills conducted during the second through third quarters of Calendar Year 2002

- Siren testing records for a sample of tests conducted for the second and third quarters of Calendar Year 2002

- Performance indicator summary sheets and reports

3. Occupational Radiation Safety Cornerstone

The inspectors verified the accuracy and completeness of the data used to calculate and report the occupational exposure control effectiveness performance indicator data for the second quarter of Calendar Year 2002 through the first quarter of 2003. The inspectors used NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 2, as guidance and interviewed licensee personnel responsible for compiling the information. Controlled access area entries with exposures greater than 100 millirems within the past 12 months were reviewed, and selected examples were examined to determine whether they were within the dose projections of the governing RWPs. Whole-body counts or dose estimates were reviewed if radiation workers received committed effective dose equivalents of more than 100 millirems. Where applicable, the inspector reviewed the summation of unintended deep dose equivalent and committed effective dose equivalent to verify that the total effective dose equivalent did not surpass the performance indicator threshold without being reported.

4. Public Radiation Safety Cornerstone

The inspectors verified the accuracy and completeness of the data used to calculate and report the radiological effluent Technical Specification/offsite dose calculation manual radiological effluent occurrences performance indicator data for the second quarter of Calendar Year 2002 through the first quarter of 2003. The inspectors used NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 2, as guidance and reviewed radiological effluent release program corrective action records, licensee event reports (LERs), and annual effluent release reports and interviewed licensee personnel responsible for compiling the information.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution (71152)

a. Inspection Scope

The inspectors examined two licensee corrective action program issues to provide an indication of the overall problem identification and resolution performance. The licensee’s problem identification process was reviewed against the criteria in 10 CFR Part 50 Appendix B, Criterion XVI.

.1 Primary Containment Airlocks

During the week of February 24, 2003, the inspectors interviewed members of the engineering staff and examined engineering Review ER-RB-2003-0053, “JRB-DRA1 Airlock Interlock Cable,” to assess the licensee’s identification of root and contributing causes.

.2 Low Pressure Emergency Core Cooling System Minimum Flow Valves

During the week of December 30, 2003, the inspectors interviewed members of the engineering and operations departments and reviewed CR-RBS-2002-0437, “Low Pressure Core Spray Pump Minimum Flow Valve E21-MOVF011 Failed to Close During Surveillance Testing,” to assess the licensee’s identification of the problem’s root cause and that appropriate corrective actions were taken to prevent recurrence.

b. Findings

Introduction. The inspectors identified a green NCV for the failure to take proper corrective action following a failure of the low pressure core spray pump minimum flow valve that resulted in an identical failure of the residual heat removal Pump A minimum flow valve 9 months later.

Description. On December 28, 2001, residual heat removal Pump A minimum flow Valve E12-MOVF064A failed to close as required when the pump was started in suppression pool cooling for a periodic heat exchanger flush. The licensee determined the cause to be a failure of the anti-sudden reversal relay, which when de-energized disabled the automatic closure of the valve until the valve had been closed at least one second. The relay failure also disabled the ability to manually close the valve from the main control room. The inspectors determined that the failure to properly evaluate and take corrective actions for an identical failure of Low Pressure Core Spray Pump Minimum Flow Valve E21-MOVF011 on March 19, 2002, led to the failure of Valve E12-MOVF064A.

During the March 19, 2002, performance of STP-205-6301, “LPCS Quarterly Pump and Valve Operability Test,” low pressure core spray pump minimum flow Valve E21-MOVF011 did not close as required. The licensee initiated CR-RBS-2002-00437 and troubleshooting showed that the anti-sudden reversal relay failed. An apparent cause investigation was conducted and determined the failure to be “a random failure” because no other failures of similar relays were identified. On April 11, 2003, the CR review group upgraded the significance of the CR to perform a root cause determination, because the failure of Valve E21-MOVF011 was determined to be a maintenance rule functional failure.

Root Cause Analysis Report, “Low Pressure Core Spray System Minimum Flow Valve E21-MOVF011 Did Not Stroke Closed During STP-205-6301," dated July 29, 2003, concluded that the root cause of the failure of Valve E21-MOVF011 to stroke was a random failure of the anti-sudden reversal relay. The root cause analysis included a historical search of industry and plant records. No problems were identified. The root cause analysis did acknowledge that residual heat removal Pumps A and B minimum flow Valves E12-MOVF064A and -B had the same relays in their control circuit, but it stated that “the time delay relays installed in E12-MOVF064A and MOVF64B circuits have performed satisfactory.” There were no action items in the root cause analysis or CR to investigate the potential for the failure of the same relays that were installed in Valves E12-MOVF064A and -B.

The inspectors determined that the root cause analysis failed to reference NRC Information Notice (IN) 92-27, “Thermally Induced Aging and Failure of ITE/Gould A. C. Relays Used in Safety Related Applications,” dated April 3, 1992. Supplement 1 to IN 92-27, dated March 21, 1997, identified that, although IN 92-27 focused on the importance of thermal effects resulting from ganged mounting of these relays, problems could exist with different mounting configurations and with other similar ITE/Gould relays. Additionally, the root cause analysis failed to identify Procurement Document MAR-01320, dated March 5, 1998. Document MAR-01320 documented the return of 20 identical Gould solid state timer relays to stock and placed a restriction on their use until procurement engineering evaluated further use of the relays in safety-related applications due to numerous failures in the other plant applications.

The inspectors also determined that Plant Modification ER 99-0450, dated October 21, 1999, installed these anti-sudden reversal relays in the valve control circuits for Valves E21-MOV-F011 and E12-MOV-F064A and -B, but did not reference IN 92-27, Supplement 1. Additionally, ER 99-0450 did not contain a procurement engineering review of this application of these relays, as required by Document MAR-01320.

Analysis. The inspectors determined that the failure to evaluate and take proper corrective actions for the failure of the low pressure core spray pump minimum flow valve failure led to the subsequent failure of the residual heat removal Pump A minimum flow valve on December 28, 2002. The pump minimum flow valve was maintained open, when residual heat removal Train A was in its normal standby lineup. With the minimum flow valve open, residual heat removal Train A was not able to meet its design flow rate for either the low pressure coolant injection or suppression pool cooling mode of system operation. The finding was more than minor because it is associated with the mitigating systems cornerstone objective to ensure the availability, reliability and capability of systems (residual heat removal Train A) that respond to initiating events to prevent undesirable consequences. The inspectors evaluated the finding using IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Reactors,” and determined that the residual heat removal Pump A minimum flow valve failure was of very low safety significance (Green) because the other low pressure coolant injection systems were available and the other train of suppression pool cooling was available at the time. This problem identification and resolution issue was documented in the licensee’s corrective action program as CR-RBS-2002-2088.

Enforcement. The inspectors determined that the failure to evaluate and take proper corrective actions for the low pressure core spray pump minimum flow valve failure was a violation of 10 CFR Part 2, Appendix B, Criterion XVI, which states, in part, that “in the case of significant conditions adverse to quality, the measures taken shall ensure that the cause of the condition is determined and corrective actions are taken to preclude repetition.” Because this problem identification and resolution issue was of very low safety significance and has been entered in the licensee’s corrective action program, this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy, NUREG-1600 (NCV 50-458/2003-03-09).

4OA3 Event Followup (71153)

.1 (Closed) LER 2002-001-00 September 18, 2002, Automatic Reactor Scram due to Main Turbine Electrohydraulic Control Malfunction

On September 18, 2002, an automatic reactor scram occurred due to a failure of a power supply in the main turbine electrohydraulic control system. During the scram, full flow condensate filter bypass Valve CNM-FCV200 closed unexpectedly, resulting in a loss of the condensate and feedwater systems. The licensee documented the event in CR-RBS-2002-1371 and -1372. The NRC conducted a special inspection of the event, the results of that inspection were documented in NRC Inspection Report 50-458/02-07. The inspectors reviewed the LER with respect to the accuracy of the report, appropriateness of corrective actions, violations of requirements, and generic issues and no findings of significance were identified.

.2 Reactor Recirculation Pump Motor Breaker 4B Ground Fault Trip

a. Inspection Scope

On February 24, 2003, reactor recirculation Pump B tripped while being upshifted to fast speed during return to full power operations following a manual reactor scram. The inspectors observed operations and reactor engineering personnel response to the event and documented those observations in Section 1R14.3 of this report.

Following the event, the inspectors evaluated the licensee's response to the failure of the ground fault detection circuit for reactor recirculation pump motor Breaker 4B. During that evaluation, the inspectors reviewed the following documents:

- CR-RBS-2003-0682, Reactor recirculation pump motor Breaker 4B trip while upshifting pump to fast speed
- ABB Power Distribution Inc. Current Transformer 10 CFR 21 Report, dated April 9, 1991
- Operating Event Report 9586, "Grand Gulf Nuclear Station Circuit Breaker Current Transformer Liquefaction," dated January 19, 1999
- Routine Maintenance Task 11497, "Reactor Recirculation Pump Motor Pump B Switchgear 4B Current Transformer Inspection," dated September 28, 2001
- Elementary Diagram ESK-5RCS08, "4.16 KV Switchgear Control Circuit Reactor Recirculation Pump Breaker 1ENS*ACB38 (4B)," Revision 9

2. Findings

No findings of significance were identified.

4OA6 Management Meetings

Exit Meetings

The inspectors presented the inspection results to Mr. P. Hinnenkamp, Vice President-Operations, and other members of licensee management at the conclusion of the emergency preparedness inspection on January 30, 2003. Licensee management acknowledged the inspection findings.

The inspectors presented the inspection results to Mr. W. Brian, Director, Engineering, and other members of licensee management at the conclusion of the biennial permanent plant modifications inspection on February 14, 2003. Licensee management acknowledged the inspection findings.

The inspectors presented the inspection results to Mr. P. Hinnenkamp, Vice President-Operations, and other members of licensee management at the conclusion of the inservice testing inspection on March 28, 2003. Licensee management acknowledged the inspection findings.

The inspectors presented the routine resident inspection results to Mr. P. Hinnenkamp, Vice President-Operations, and other members of licensee management at the conclusion of the inspection period on April 1, 2003. Licensee management acknowledged the inspection findings.

The inspectors presented the inspection results via teleconference with Mr. P. Hinnenkamp, Vice President-Operations, and other members of licensee management at the conclusion of the occupational radiation safety inspections on April 9, 2003. Licensee management acknowledged the inspection findings.

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

4OA7 Licensee-Identified Violations

The following findings of very low safety significance were identified by the licensee as violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as NCVs.

- Technical Specifications Section 5.4.1.a requires that “procedures shall be established implemented and maintained covering the . . . applicable procedures recommended in Regulatory Guide 1.33 Revision 2, Appendix A, February 1978.” Specifically, Reg Guide 1.33, Appendix A, Section 7, “Procedure for Control of Radioactivity,” Subsection d, “BWR Air Extraction, Offgas Treatment, and Other Gaseous Effluent Systems,” Item (1), refers to “Mechanical Vacuum Pump Operation.” On February 23, 2003, the licensee identified that, contrary to technical requirements manual Table 3.3.6.1-1, Section 1.l, “Main Steam Line Radiation - High-High,” which states, in part, that the nominal setpoint shall be “3.0 x full power backround” and Note (h) which states that the “Setpoints [are] to be verified: 1) Within 30 days after a significant change in hydrogen injection, or 2) During Mode 1 or 2 with the mechanical vacuum pump in operation," mechanical vacuum Pump A was in operation while in Mode 2 with the main steam line radiation monitor setpoints greater than three times the expected full power backround radiation levels. This human performance error was documented in the licensee's corrective action program as CR-RBS-2003-0661. This NCV was only of very low safety significance (Green) because it only potentially affected the barrier integrity cornerstone and the condition existed for less than 16 hours.

- Technical Specification 5.4.1.a requires written procedures in accordance with Regulatory Guide 1.33, Revision 2, February 1978, Appendix A. Procedure RSP-0126, Revision 4A, "Radioactive Source Control," Section 7.4.2, requires qualified source users to maintain positive control of radioactive sources until they are returned to the radioactive source custodian. However, on March 20, 2003, the licensee identified a failure to maintain positive control of a radioactive source. This human performance error was documented in the licensee's corrective action program as CR-RBS-2003-1099. This NCV is only of very low safety significance (Green) because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.

ATTACHMENT

SUPPLEMENTARY INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

B. Allen, Manager, Emergency Preparedness
D. Beauchamp, Superintendent, Quality Control
W. Brian, Director, Engineering
D. Burnett, Superintendent, Chemistry
C. Bush, Assistant Operations Manager
J. Fowler, Manager, Quality Assurance
A. James, Superintendent, Plant Security
T. Gates, Manager, System Engineering
H. Goodman, Manager, Nuclear Engineering
R. Godwin, Manager, Training and Development
J. Heckenberger, Manager, Planning, Scheduling and Outage
P. Hinnenkamp, Vice President, Operations
B. Houston, Superintendent, Radiation Protection, Waterford Steam Electric Station, Unit 3
R. King, Director, Nuclear Safety Assurance
J. Leavines, Manager, Licensing
T. Lynch, Manager, Operations
J. Malara, Manager, Design Engineering
W. Mashburn, Manager, Programs and Components
J. McGhee, Manager, Maintenance
T. Trepanier, General Manager, Plant Operations
W. Trudell, Manager, Corrective Action and Assessment
D. Wells, Superintendent, Radiation Protection

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-458/2003-03-01	URI	Failure to maintain watertight integrity of severe weather doors compromised the availability of a safe shutdown system (Section 1RO6)

Closed

50-458/2002-001-00	LER	September 18, 2002, Automatic Reactor Scram due to Main Turbine Electrohydraulic Control Malfunction (Section 4OA3.1)

Opened and Closed

50-458/2003-03-02	NCV	Failure to develop a sufficiently detailed work plan (Section 2OS1b.1)

50-458/2003-03-03	NCV	Failure to survey (Section 2OS1b.2)
50-458/2003-03-04	NCV	Failure to post an airborne radioactivity area (Section 2OS1b.3)
50-458/2003-03-05	NCV	Failure to instruct workers (Section 2OS1b.4)
50-458/2003-03-06	NCV	Failure to control a locked high radiation area (Section 2OS1b.5)
50-458/2003-03-07	FIN	Failure to maintain collective doses ALARA that were associated with RWP 2003-1929 (Section 2OS2b.1)
50-458/2003-03-08	FIN	Failure to maintain collective doses ALARA that were associated with RWP 2003-1935 (Section 2OS2b.2)
50-458/2003-03-09	NCV	Failure to take proper corrective actions for low pressure core spray pump minimum flow valve failure resulted in the failure of the residual heat removal pump minimum flow valve (Section 4OA2.2)

Discussed

None.

DOCUMENTS REVIEWED

The following documents were selected and reviewed by the inspectors to accomplish the objectives and scope of the inspection and to support any findings:

Condition Reports

CR-RBS-2001-0422	CR-RBS-2002-0589	CR-RBS-2002-1332	CR-RBS-2003-0318
CR-RBS-2001-0615	CR-RBS-2002-0592	CR-RBS-2002-1507	CR-RBS-2003-0504
CR-RBS-2001-0678	CR-RBS-2002-0603	CR-RBS-2002-1699	CR-RBS-2003-1012
CR-RBS-2001-1569	CR-RBS-2002-0606	CR-RBS-2002-1742	CR-RBS-2003-1071
CR-RBS-2002-0183	CR-RBS-2002-0636	CR-RBS-2002-1779	CR-RBS-2003-1136
CR-RBS-2002-0255	CR-RBS-2002-0711	CR-RBS-2002-1856	CR-RBS-2003-1145
CR-RBS-2002-0292	CR-RBS-2002-0762	CR-RBS-2003-0034	CR-RBS-2003-1209
CR-RBS-2002-0397	CR-RBS-2002-0788	CR-RBS-2003-0037	CR-RBS-2003-1341
CR-RBS-2002-0448	CR-RBS-2002-0875	CR-RBS-2003-0039	CR-RBS-2003-1372
CR-RBS-2002-0483	CR-RBS-2002-0876	CR-RBS-2003-0054	

CR-RBS-2002-0539	CR-RBS-2002-0946	CR-RBS-2003-0250
CR-RBS-2002-0574	CR-RBS-2002-1118	CR-RBS-2003-0274

Engineering Reports:

ER96-0712-000-0, “Replace IAS Safety Related Accumulator Tank Relief Valves with Relief Valves with Limited Blowdown,” Revision 0

ER97-01-0197-000-0, “Delete the SLC System Heat Tracing Circuitry and Controls,” Revision 0

ER-RB-2002-0398-001, “Evaluate Temporary Drain Plug on HDL-LV238 as a Permanent Modification,” Revision 0

ER-RB-1999-0730, “Modify Safety Related Limitorque Actuator to Increase the Torque Output Capability as a Result of Limitorque Technical Update 98-01,” Revision 0

ER-RB-2000-0375-000, “IFTS Time Delay Relay,” Revision 0

ER 98-0729, “SRV Logic Rosemount Trip Unit Modifications,” Revision 0

MR-96-0008, “Installation of Thermocouple to Monitor Various Points on Reactor Coolant Pumps and Seals,” Revision 0

MR-96-0066, “Installation of Ground Detector System in Division III 125VDC Battery System," Revision 0

ER-2001-0807, “Repair Options for Intercoolers on Division I and Division II Emergency Diesel Generator Engines,” Revision 0

ER-2002-0027, “Replacement Stator Winding Thermocouples for CWS-PID,” Revision 0

ER-2000-0330, “Install Relief Valves on the Return Lines of the Containment Unit Coolers,” Revision 0

ER-RB-2002-0256-000, “Flow Accelerated Corrosion Tcrit Data for RF11,” Revision 0

Calculations:

G13.3.E-192, “Standby Diesel Loading Calculation,” Revision 4

G13.18.10.2*143, “Determination of Minimum Wall Thickness of Piping Components Affected by Errosion/Corrosion,” Revision 1Q Addendum

Maintenance Action Items

MAI 351687, Replace Main Steam Line “A” Shutoff Valve Leakage Monitoring Connection Isolation Valve MSS-V134

Procedures

SPP-7009, “Storing, Handling, And Issuing Welding, Brazing, And Soldering Material,” Revision 13

NDE 9.04, “Ultrasonic Examination of Ferritic Piping Welds (ASME Section XI),” Revision 2

NDE 9.23, “Ultrasonic Examination of Austenitic Piping Welds (ASME Section XI),” Revision 2

NDE 9.19, “Ultrasonic Instrument Linearity Verification,” Revision 3

NDE 9.31, “Magnetic Particle Examination (MT) for ASME Section XI,” Revision 2

ISWT-PDI-AUT1, “Automated Inside Surface Ultrasonic Examination of Ferritic Vessel Walls Greater Than 4.0 Inches in Thickness,” Revision 0

WPS E-PI-T-A1, “Manual GTAW of P No. 1 Carbon Steels Using ER 70S-2 Filler Metal,” Revision 1

Welding Procedure Qualifications Records

PQR 015, Revision 1
PQR 029, Revision 1
PQR 330, Revision 1
PQR 331, Revision 1

Inservice Inspection Test Reports

Magnetic Particle Examinations

03IR20403
03IR20406
03IR20397
03IR20432

Radiographic Examination

03IR20463

Ultrasonic Examinations

03IR20414
03IR20405
03IR20407
03IR20413

Miscellaneous Reports

NDE Personnel Certification Packages Including Visual Acuity
CMTRs for Welding Material, Type 7018, Heats 40511, 91191, and 625910
CMTR for Welding Material, Type ER 70S-2, Heat 065595

LIST OF ACRONYMS AND INITIALISMS USED

ALARA	as low as reasonably achievable
ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
CR	condition report
CR-RBS	River Bend Station condition report
EPP	emergency preparedness procedure
ER	engineering request
ERO	Emergency Response Organization
IMC	NRC inspection manual chapter
IN	NRC Information Notice
LER	licensee event report
MAI	maintenance action item
MSL	mean sea level
NCV	noncited violation
NDE	nondestructive examination
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
RFO-11	11th refueling outage
RWP	radiation work permit
SERT	significant event review team
SSC	structure, system, or component
STP	surveillance test procedure
URI	unresolved item
USAR	Updated Safety Analysis Report