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

August 14, 2006

Paul D. Hinnenkamp
Vice President - Operations
Entergy Operations, Inc.
River Bend Station
5485 US Highway 61N
St. Francisville, LA 70775

SUBJECT: RIVER BEND STATION - NRC INTEGRATED INSPECTION REPORT 05000458/2006003

Dear Mr. Hinnenkamp:

On June 30, 2006, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your River Bend Station. The enclosed integrated inspection report documents the inspection results, which were discussed on July 5, 2006, with you and other members of your staff.

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

The report documents three NRC-identified findings and two self-revealing findings of very low safety significance (Green). The NRC has also determined that violations are associated with these findings. However, because these violations were of very low safety significance and were entered into your corrective action program, the NRC is treating these violations as noncited violations, consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest the violations or the significance of the violations, 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 76011-4005; 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 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Website 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/

Kriss M. Kennedy, Chief
Project Branch C
Division of Reactor Projects

Docket: 50-458
License: NPF-47

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NRC Inspection Report 05000458/2006003
w/Attachment: Supplemental Information

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RIV:SRI:DRP/C	RI:DRP/C	C:DRS/OB	C:DRS/EB1	C:DRS/PSB
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C:DRS/EB2	SRA:DRS	C:DRP/C		
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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	50-458
License:	NPF-47
Report:	05000458/2006003
Licensee:	Entergy Operations, Inc.
Facility:	River Bend Station
Location:	5485 U.S. Highway 61 St. Francisville, Louisiana
Dates:	April 1 to June 30, 2006
Inspectors:	P. Alter, Senior Resident Inspector, Project Branch C M. Miller, Resident Inspector, Project Branch C G. Werner, Senior Project Engineer, Project Branch D L. Ricketson, P.E., Senior Health Physicist, Plant Support Branch W. Sifre, Senior Reactor Inspector, Engineering Branch 1
Approved By:	Kriss M. Kennedy, Chief Project Branch C Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000458/2006003; 04/01/2006 - 06/30/2006; River Bend Station; Operability Evaluations, Refueling and Other Outage Activities, Surveillance Testing, Occupational Radiation Safety.

The report covered a 3-month period of routine baseline inspections by resident inspectors and announced baseline inspections by regional engineering and radiation protection inspectors. Five Green noncited violations were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 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. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

Green. A self-revealing noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," was reviewed involving the failure of the licensee to identify that the normal supply breaker to the Division III 4.16 kV engineered safety features bus was not properly racked in for a period of 24 days following maintenance. This issue was entered into the licensee's corrective action program as CR-RBS-2006-02402.

The finding was more than minor because it was associated with the mitigating system cornerstone attribute of configuration control and affected the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Utilizing Manual Chapter 0609, "Significance Determination Process," a Phase 3 analysis concluded that the finding was of very low safety significance. The cause of the finding was related to the crosscutting aspect of problem identification and resolution in that the licensee failed to properly evaluate available indications to identify that the breaker was not properly racked in. (Section 1R15).

Green. An NRC identified noncited violation of 10 CFR 50.65 Maintenance Rule Section (a)(4) was identified for the failure of the licensee to provide prescribed compensatory measures for two Orange shutdown risk conditions during Refueling Outage 13. Specifically, the preoutage risk assessment recommended that two work orders be in place for maintenance electricians to provide power to one spent fuel pool cooling pump in the event of problems with the running pump during periods of electrical bus maintenance. The inspectors found that the work packages were not in place before entering shutdown risk condition Orange on April 26, 2006, during the Division II engineering safety features bus testing, and May 3, 2006, during the Division I engineered safety features bus outage. This issue was entered into the licensee's corrective action program as CR-RBS-2006-01937.

The finding was more than minor because the licensee failed to implement a prescribed compensatory measure during the highest risk condition of Refueling Outage 13. The

specific compensatory measures were called for in the preoutage risk assessment and the shutdown operations protection plan. The finding affected the mitigating system cornerstone because of the increased risk of a sustained loss of spent fuel pool cooling during core offloading operations. The finding could not be evaluated using the significance determination process, therefore the finding was reviewed by regional management and determined to be of very low safety significance. Factors that were considered included: (1) electrical maintenance technicians had previously performed the task of providing alternate power to a spent fuel pool cooling pump, (2) the necessary equipment was staged as part of the abnormal operating procedure for loss of decay heat removal, and (3) the relatively long "time to boil" of the spent fuel storage pool at that time during the refueling outage. The cause of the finding was related to the crosscutting aspect of human performance because the licensee's planned maintenance activities and the predetermined increase in outage risk was not effectively managed by prescribed compensatory measures (Section 1R20).

Green. An NRC identified noncited violation of Technical Specification 5.4.1.a was identified for the failure of the licensee to provide an adequate surveillance test procedure to perform Technical Specification Surveillance Requirement 3.8.1.1. Specifically, STP-000-0102, "Power Distribution Alignment Check," Revision 4, did not verify the required offsite power circuit breaker alignment and indicated power availability for the Division III 4.16 kV engineered safety features bus as required in Modes 1, 2, and 3. This issue was entered into the licensee's corrective action program as CR-RBS-2006-02675 and -02402.

The finding was more than minor because it was associated with the mitigating system cornerstone attribute of configuration control and affected the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Utilizing Manual Chapter 0609, "Significance Determination Process," a Phase 3 analysis concluded that the finding was of very low safety significance. (Section 1R22).

Cornerstone: Occupational Radiation Safety

- Green. The inspector reviewed a self-revealing noncited violation of Technical Specification 5.7.1, resulting from the licensee's failure to control access to a high radiation area. While transferring reverse osmosis system filters in the radwaste building, the licensee allowed two workers to inadvertently enter a high radiation area. This occurred after a guard prematurely left his post in front of the 123 foot elevation elevator door. The highest dose rate recorded by an electronic alarming dosimeter was 164 millirem per hour. The guard returned and evacuated the workers before they accrued additional radiation dose. Planned corrective action was still being evaluated by the licensee at the conclusion of the inspection.

  The finding was more than minor because it was associated with the occupational radiation safety attribute of exposure control and affected the cornerstone objective in that not controlling a high radiation area could increase personal exposure. Using the Occupational Radiation Safety Significance Determination Process, the inspector determined that the finding was of very low safety significance because it did not

involve: (1) an as low as is reasonably achievable finding, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. Additionally, this finding had crosscutting aspects associated with human performance in that the failure of the individual to guard the elevator door directly contributed to the violation. (Section 2OS1)

- Green. The inspector identified a noncited violation of 10 CFR 20.1501(a) because the license failed to survey airborne radioactivity. During the removal of local power range monitors, the licensee started collecting an air sample of the work area, but discarded the sample before analyzing it. Successful passage through the portal monitors at the exit of the controlled access area confirmed that no worker experienced an uptake of radioactive material. Planned corrective action is still being evaluated.

  The finding was more than minor because it was associated with the occupational radiation safety program attribute of exposure control and affected the cornerstone objective in that the lack of knowledge of radiological conditions could increase personnel dose. Using the Occupational Radiation Safety Significance Determination Process, the inspector determined that the finding was of very low safety significance because it did not involve: (1) an as low as is reasonably achievable finding, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. Additionally, this finding had crosscutting aspects associated with human performance in that the failure to maintain the sample for analysis directly contributed to the violation. (Section 2OS1)

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status: The reactor was operated at 100 percent power from April 1-15, 2006, when the reactor scrammed due to a control circuit failure which caused both reactor recirculation pumps to shift to slow speed. The reactor was restarted on April 17 and attained 100 percent power on April 18. On April 23, the reactor was shut down for Refueling Outage (RFO) -13. On May 12, the reactor was restarted and attained 100 percent power on May 18. On June 15, reactor power was reduced to 23 percent because of a problem with the main turbine bypass valves. The reactor was returned to 100 percent power on June 18. The reactor remained at 100 percent power for the remainder of the inspection period, with the exception of regularly scheduled power reductions for control rod pattern adjustments and turbine testing.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness

1R01 Adverse Weather Protection

a. Inspection Scope

Hurricane Season Preparations

During the week of June 12, 2006, the inspectors completed a review of the licensee's readiness for seasonal susceptibilities involving high winds at the beginning of hurricane season. The inspectors reviewed Procedure ENS-EP-302, “Severe Weather Response,” Revision 4. The inspectors: (1) reviewed plant procedures, the Updated Safety Analysis Report (USAR), and Technical Specifications (TS) to verify that operator actions defined in adverse weather procedures maintained the readiness of essential systems; (2) walked down portions of the protected area to verify that hurricane season preparations were sufficient to support operability of essential systems, including the ability to perform safe shutdown functions; (3) evaluated operator staffing levels to verify the licensee could maintain the readiness of essential systems required by plant procedures; and (4) reviewed the corrective action program (CAP) to determine if the licensee identified and corrected problems related to adverse weather conditions.

The inspectors completed one inspection sample.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

Partial System Walkdowns

a. Inspection Scope

The inspectors: (1) walked down portions of the three risk important systems listed below and reviewed system operating procedures (SOPs), piping and instrument diagrams, and other documents to verify that critical portions of the selected systems were correctly aligned; and (2) compared deficiencies identified during the walkdown to the licensee's USAR and CAP to verify problems were being identified and corrected.

- Alternate decay heat removal system, which was the backup to the inservice shutdown cooling system during refueling operations, on May 2, 2006

- Reactor core isolation cooling system, while the high pressure core spray diesel was out of service for maintenance, on June 12, 2006

- Division I emergency diesel generator (EDG), while Division II EDG was out of service for planned maintenance, on June 21, 2006

Documents reviewed by the inspectors included:

- SOP-0140, "Suppression Pool Cleanup and Alternate Decay Heat Removal," Revision 16

- SOP-0035, "Reactor Core Isolation Cooling System," Revision 8A

- SOP-0053, "Standby Diesel Generator and Auxiliaries," Revision 44A

The inspectors completed three inspection samples.

h. Findings

No findings of significance were identified.

1R05 Fire Protection

b. Inspection Scope

The inspectors walked down the six plant areas listed below to assess the material condition of active and passive fire protection features and their operational lineup and readiness. The inspectors: (1) verified that transient combustibles were controlled in accordance with plant procedures; (2) observed the condition of fire detection devices to verify they remained functional; (3) observed fire suppression systems to verify they remained functional and that access to manual actuators was unobstructed; (4) verified that fire extinguishers and hose stations were provided at their designated locations and

that they were in a satisfactory condition; (5) verified that passive fire protection features (electrical raceway barriers, fire doors, fire dampers, steel fire proofing, penetration seals, and oil collection systems) were in a satisfactory material condition; (6) verified that adequate compensatory measures were established for degraded or inoperable fire protection features and that the compensatory measures were commensurate with the significance of the deficiency; and (7) reviewed the CAP to determine if the licensee identified and corrected fire protection problems.

- Auxiliary building piping Tunnel D, Fire Area AB-7, on May 9, 2006

- Low pressure core spray pump room, Fire Area AB-6/Z-1, on May 9, 2006

- High pressure core spray pump room, Fire Area AB-2/Z-1, on May 9, 2006

- Control building standby switchgear Room 1A, Fire Area C-15, on June 22, 2006

- Control building safety related cable tray area and stairway Number 3, Fire Area C-16 and C-29, on June 22, 2006

- Division I EDG control and diesel engine rooms, Fire Area DG-6/Z-1, on June 22, 2006

Documents reviewed by the inspectors included:

- Pre-Fire Plan/Strategy Book
- USAR Section 9A.2, “Fire Hazards Analysis,” Revision 10
- River Bend Station postfire safe shutdown analysis
- RBNP-038, “Site Fire Protection Program,” Revision 6B

The inspectors completed six inspection samples.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities

a. Inspection Scope

The inspector witnessed the performance of 12 volumetric (ultrasonic) and four surface (liquid penetrant) examinations. The sample of nondestructive examination (NDE) activities is listed in the attachment.

For each of the NDE activities reviewed, the inspector verified that the examinations were performed in accordance with American Society of Mechanical Engineers (ASME) Code requirements.

During the review of each examination, the inspector verified that appropriate NDE procedures were used, that examinations and conditions were as specified in the procedure, and that test instrumentation or equipment was properly calibrated and within the allowable calibration period. The inspector also reviewed documentation to verify that indications revealed by the examinations were dispositioned in accordance with the ASME Code specified acceptance standards.

The inspector verified the certifications of the NDE personnel observed performing examinations or identified during review of completed examination packages.

The inspection procedure requires review of one or two examinations from the previous outage with recordable indications that were accepted for continued service to ensure that the disposition was done in accordance with the ASME Code. There were no recordable indications that required evaluation during the last outage.

If the licensee completed welding on the pressure boundary for Class 1 or 2 systems since the beginning of the previous outage, the procedure requires verification that acceptance and preservice examinations were done in accordance with the ASME Code for one to three welds. There were no welds available for review.

The procedure also requires verification that one or two ASME Code Section XI repairs or replacements meet code requirements. There were no code repairs or replacements available at the time of this inspection.

The inspectors completed 16 inspection samples.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program

a. Inspection Scope

On June 13, 2006, the inspectors observed testing and training of senior reactor operators and reactor operators to verify the adequacy of training, to assess operator performance, and to assess the evaluators' critique. The training evaluation scenario observed was RSMS-OPS-422, "Loss of Circ Water Pump, Failure of Steam Flow Transmitter and Instrument Air System Leak," Revision 4.

The inspectors completed one inspection sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the condition reports (CR) listed below which documented equipment problems to: (1) verify the appropriate handling of structure, system, and component (SSC) performance or condition problems; (2) verify the appropriate handling of degraded SSC functional performance; (3) evaluate the role of work practices and common cause problems; and (4) evaluate the handling of SSC issues reviewed under the requirements of the maintenance rule; 10 CFR Part 50, Appendix B; and TS.

- CR-RBS-2006-1898, main steam stop Valve B21-MOVF098C leakage, reviewed on June 2, 2006, and CR-RBS-2004-4338, main steam stop Valve B21-MOVF098C high leakage during RFO-11 and -12, reviewed on June 26, 2006.

- CR-RBS-2006-2302, primary containment integrity maintenance rule repetitive functional failure, reviewed on June 26, 2006.

Documents reviewed by the inspectors included:

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

- Maintenance rule function list

- Maintenance rule performance criteria list

- Main steam stop valve maintenance rule performance evaluations

The inspectors completed two inspection samples.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

.1 Risk Assessment and Management of Risk

The inspectors reviewed the planned work weeks listed below to verify: (1) that the licensee performed risk assessments when required by 10 CFR 50.65 (a)(4) and administrative Procedure ADM-096, “Risk Management Program Implementation and On-Line Maintenance Risk Assessment,” Revision 4B, prior to changes in plant configuration for maintenance activities and plant operations; (2) the accuracy, adequacy, and completeness of the information considered in the risk assessment;

(3) that the licensee recognized, and entered as applicable, the appropriate licensee established risk category according to the risk assessment results and Procedure ADM-096; and (4) that the licensee identified and corrected problems related to maintenance risk assessments. Specific work activities evaluated included planned and emergent work for the weeks of:

- June 5, 2006, Division I work week and preferred station service Transformer RTX-ESR1F cooling oil dehydration
- June 19, 2006, planned Division II EDG outage week
- June 26, 2006, nondivisional work week and potential labor work stoppage

.2 Emergent Work Control

For the two emergent work activities listed below, the inspectors: (1) verified that the licensee performed actions to minimize the probability of initiating events and maintained the functional capability of mitigating systems and barrier integrity systems; (2) verified that emergent work related activities such as troubleshooting, work planning/scheduling, establishing plant conditions, aligning equipment, tagging, temporary modifications, and equipment restoration did not place the plant in an unacceptable configuration; and (3) reviewed the CAP to determine if the licensee identified and corrected risk assessment and emergent work control problems.

- Preferred station service Transformer RTX-ESR1F sudden pressure relay failure on May 30, 2006
- Main turbine bypass valves inoperable due to hydraulic oil leak on June 2, 2006

The inspectors completed five inspection samples.

c. Findings

No findings of significance were identified.

1R14 Operator Performance During Nonroutine Evolutions and Events

a. Inspection Scope

1. April 4, 2006, Automatic Initiation of Standby Service Water

The inspectors: (1) reviewed operator logs, plant computer data, and strip charts for the April 4, 2006, unexpected initiation of Division II standby service water that occurred while swapping the running normal service water pumps to evaluate operator performance in coping with the event; (2) verified that operator actions were in accordance with the response required by plant procedures and training; and (3) verified that the licensee identified and implemented appropriate corrective actions associated with personnel performance problems that occurred during the transient. In addition, the

inspectors reviewed CR-RBS-2006-01257, which documented the procedural problems that led to the event and reviewed the following procedures used by the operators:

- AOP-53, “Initiation of Standby Service Water With Normal Service Water Running,” Revision 8

- SOP-42, “Standby Service Water System,” Revision 25

- SOP-66, “Control Building HVAC Chilled Water System,” Revision 33B

2. April 15, 2006, Reactor Scram

The inspectors: (1) reviewed operator logs, plant computer data, and strip charts for the April 15, 2006, unexpected reactor recirculation pump downshift and subsequent reactor scram to evaluate operator performance in coping with the event; (2) verified that operator actions were in accordance with the response required by plant procedures and training; and (3) verified that the licensee identified and implemented appropriate corrective actions associated with personnel performance problems that occurred during the transient. In addition the inspectors reviewed the postscram report documented in Procedure GOP-003, “Scram Recovery,” Revision 16A, and observed the onsite safety review committee review of the postscram report.

The inspectors completed two inspection samples.

e. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

For the operability evaluations associated with the documents listed below, the inspectors: (1) reviewed plants status documents such as operator shift logs, emergent work documentation, deferred modifications, and standing orders, to determine if an operability evaluation was warranted for degraded components; (2) referred to the USAR and design basis documents to review the technical adequacy of licensee operability evaluations; (3) evaluated compensatory measures associated with operability evaluations; (4) determined degraded component impact on any TS; (5) used the significance determination process to evaluate the risk significance of degraded or inoperable equipment; and (6) verified that the licensee identified and implemented appropriate corrective actions associated with degraded components.

- CR-RBS-2006-01207 and -01215, Primary containment purge exhaust line fails to meet leak rate acceptance criteria, reviewed during the week of April 3, 2006

- CR-RBS-2005-02805, Inserted control Rod 24-29 control blade lifetime calculation revised for extended operating cycle, reviewed during the week of April 17, 2006

- Work Request (WR) 76625, NNS-ACB23 “control power” light out, suspect bad socket, reviewed during the week of May 29, 2006

- TS-LCO-06-0711, Division II EDG Generator Output Breaker charging springs did not charge during tagout restoration, reviewed on June 23, 2006

- CR-RBS-2006-01257, Division II standby service water start on low service water pressure, reviewed on June 28, 2006

- CR-RBS-2006-02632, turbine bypass valves hydraulic oil leak, reviewed on June 28, 2006

Other documents reviewed by the inspectors are listed in the attachment.

The inspectors completed six inspection samples.

b. Findings

Introduction: The inspectors reviewed a self-revealing noncited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," involving the failure of the licensee to identify that the normal supply breaker to the Division III 4.16 kV engineered safety features (ESF) bus was not properly racked in following maintenance.

Description: Following the completion of planned maintenance on Switchgear NNS-SWG1A on April 29, 2006, operators were assigned to clear equipment tags and restore the system alignment. As part of this task, operators racked in Breaker NNS-ACB23, the normal supply breaker to 4.16 kV Switchgear NNS-SWG1C. No actions, such as cycling the breaker, were required to verify that the breaker was properly racked in.

On May 9, 2006, after noting that the control power light associated with Breaker NNS-ACB23 was not lit, operators wrote WR 76625 to repair the light. The WR stated that the white control power light on Control Room Panel H13-P808 was out with the breaker racked in and the control power fuses installed. The WR also indicated that the suspected cause was a bad socket and that position Switch 52H had failed in the past to make up during closure. A work control center senior reactor operator determined that an operability evaluation was not required for the condition described in WR 76625. The WR was classified “4D,” which indicated that it should be scheduled as resources allowed within the normal 16-week work planning schedule. The inspectors noted the licensee did not write a CR. The white control power light provides indication that the breaker is functional, specifically, that: (1) there is no electrical fault on the line or load side of the breaker, (2) the breaker “Lockout” button is not depressed on Panel 808, and (3) the breaker is fully racked into the switchgear. On May 9, 2006, there were no electrical faults on Breaker NNS-ACB23 and the “Lockout” was reset on Panel 808.

On May 22, 2006, while aligning Switchgear NNS-SWG1C and the Division III 4.16 kV ESF bus to the Transformer RSS1 offsite power supply, Breaker NNS-ACB23 failed to close. Operators racked the breaker out and in, but the breaker failed to close on the second attempt. Subsequent troubleshooting identified that the breaker had not been fully racked in as electricians were able to rotate the racking device one additional turn. The white light on Panel 808 came on and the breaker was successfully closed. The operators and electricians determined that Breaker NNS-ACB23 had not been not properly racked in, wrote CR-RBS-2006-02325 and -02337 and initiated WR 77478 to investigate the problem with racking in Breaker NNS-ACB23.

On May 25, 2006, the inspectors questioned the impact that the failure of the breaker to close had on the licensee's compliance with TS. Specifically, TS 3.8.1.a requires two qualified circuits between the offsite transmission network and the onsite Class 1E ac electrical power distribution system when the plant is in Modes 1, 2, and 3. On May 12, the plant was taken from Mode 4 to Mode 2 without two qualified offsite power sources available to the Division III 4.16 kV ESF bus. The licensee wrote CR-RBS-2006-2402 and determined that they did not comply with TS 3.8.1.a when they changed modes on May 12. In addition, the Division III 4.16 kV ESF bus was inoperable for a period of 10 days (May 12-22), which exceeded the allowed outage time of 72 hours specified in TS Condition 3.8.1.A. The licensee also discovered that, on May 14 during the conduct of maintenance on the Division I EDG, with Breaker NNS-ACB23 unable to be closed, they unknowingly entered TS Condition 3.8.1.d. TS Condition 3.8.1.d states that with "One required offsite circuit inoperable AND on required [E]DG inoperable," restore the EDG or the offsite power supply to an operable status in 12 hours or place the plant in Mode 3 within the next 12 hours. The Division I EDG was inoperable for 15 hours and 15 minutes.

The inspectors found that the licensee's procedures did not require Breaker NNS-ACB23 to be cycled to verify proper operation after it was racked in on April 29. Procedure OSP-0022, "Operations General Administrative Guidelines," Revision 01, step 4.5.5, required that breakers be functionally tested "following any activity involving safety related equipment which requires the breaker to be racked out." Because Breaker NNS-ACB23 is not classified as a safety-related breaker, it was not required to be functionally tested after it was racked in on April 29.

<u>Analysis</u>: The performance deficiency associated with this finding involved the failure of operators to identify that Breaker NNS-ACB23 was not functional on April 29, 2006. The finding was more than minor because it was associated with the mitigating system cornerstone attribute of configuration control and affected the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The Phase 1 worksheets in Manual Chapter (MC) 0609, "Significance Determination Process," were used to conclude that a Phase 2 analysis was required because both the mitigating systems and the containment barrier cornerstones were affected.

In accordance with NRC Inspection Manual Chapter 0609, Appendix A, Attachment 1, "User Guidance for Determining the Significance of Reactor Inspection Findings for At-Power Situations," the inspectors estimated the risk of the subject finding using the

Risk-Informed Inspection Notebook for River Bend Station, Revision 2. The inspectors assumed that Division III power was available, but degraded, while Breaker NNS-ACB23 was not properly installed for the 10 days that the plant was in Mode 3 or above, from May 12-22, 2006. Therefore, the exposure window used was 3-30 days. No operator recovery was credited because on two occasions, operators had proven incapable of properly positioning the breaker, ultimately requiring maintenance technicians to properly install the breaker. Using Manual Chapter 0609, Appendix A, Attachment 2, Rule 2.1, “Inspection Finding that Degrades Mitigation Capability and Does Not Reduce Remaining Mitigation Capability Credit to a Value Less Than Full Mitigation Credit,” the inspectors determined that all sequences containing the functions that would be affected by a loss of Division III power, including the Division I standby service water loop (HPCS, LPI, CHR, HPCS/LC, and REC/SSW), should be quantified, giving full mitigation capability credit to each of these functions. Because the performance deficiency affected the electric power system, Table 2 of the risk-informed notebook required that all worksheets be evaluated. The resulting dominant sequences are provided in Table 1 below:

Table 1
Phase 2 Worksheet Results

Initiator	Sequence	IEL	Mitigating Functions	Result
TNSW	5	3	SSW - REC/SSW	7*
	4	3	RCIC - HPCS - DEP	9*
LOOP	1	3	CHR - LDEP	8
	2	3	CHR - SPCFAN	8
	4	3	RCIC - HPCS - DEP	9*
	6	3	EAC1&2 - HPCS - REC6 - FPW	9*
	8	3	EAC1&2 - HPCS - SBODG - REC4	9*
	9	3	EAC1&2 - REC1 - HPCS -RCIC	9*
SORV	1	3	CHR-LDEP	8
	2	3	CHR - SPCFAN	9
	4	3	RCIC - HPCS - DEP	9*
LOIA	2	4	CHR - SPCFAN	8
	1	4	CHR-LDEP	9
TPCS	4	2	RCIC - HPCS - DEP	8
ATWS	1	6	CHR	9

* Denotes sequences indicated as LERF contributors in the Phase 2 notebook.

By application of the counting rule, the internal event risk contribution of this finding to the change in core damage frequency (ΔCDF) was determined to be of low to moderate risk significance (WHITE).

A senior reactor analyst performed further evaluation of the risk associated with this issue (Phase 3/Modified Phase 2). Because the assumptions made during the Phase 2 estimation process were overly conservative and did not completely represent the actual exposure time nor the actual affect the performance deficiency had on the availability of power to the Division III diesel generator, the senior reactor analyst modified these

assumptions to more precisely quantify the change in risk. Specifically, the exposure time was 10 days as opposed to the 30 days used in the risk-informed notebook. Additionally, the Phase 2 evaluation included loss of offsite power initiating events that were not affected by the performance deficiency because offsite power to Division III would in all likelihood be lost during a design basis loss of offsite power. The senior reactor analyst performed a modified Phase 2 estimation and determined that the internal event risk contribution of the subject finding to the ΔCDF was of very low risk significance (Green). The best estimate value of this probability ($\Delta CDF_{INTERNAL}$) was calculated by the senior reactor analyst to be 1.2×10^{-7}. The analyst evaluated the contribution of external initiating events to the risk and calculated a bounding risk estimate of 2.9×10^{-7} as the ΔCDF for internal fire events.

Using Manual Chapter 0609, Appendix H, "Containment Integrity Significance Determination Process," the analyst estimated that the potential risk contribution from large early release frequency was 6.6×10^{-8}.

Given the independence of each initiating event, the analyst determined that the best estimate of the total risk related to the subject performance deficiency was the summation of the ΔCDF calculated for both internal and external initiators. Therefore, the best estimate was 4.1×10^{-7}. The change in risk related to large early release frequency was determined to be below 6.6×10^{-8}, corroborating that the finding was of very low risk significance. The performance deficiency resulted in a finding that was of very low risk significance (Green). The cause of the finding was related to the crosscutting aspect of problem identification and resolution in that operators failed to identify that Breaker NNS-ACB23 was not properly racked in.

<u>Enforcement</u>: 10 CFR Part 50, Appendix B, Criterion XVI, Corrective Action, requires, in part, that measures be established to assure that conditions adverse to quality are promptly identified and corrected. Contrary to this, from April 29 to May 22, 2006, the licensee failed to identify that Breaker NNS-ACB23, which supplied one of the two required offsite power supplies to the Division III 4.16 kV ESF bus, was not properly racked in to Switchgear NNS-SWGIC. The root cause involved the licensee's lack of understanding that Breaker NNS-ACB23 was required to be functional to meet TS 3.8.1.a requirements for two offsite power circuits to the Division III 4.16 kV ESF bus. The corrective actions to restore compliance included: (1) changes to operations section procedures to verify the white control power light, when applicable, after a circuit breaker is racked in, (2) expansion of the requirement to functionally test safety-related breakers to the nonsafety-related breakers in the TS required offsite power circuits, and (3) operator lessons learned training on the event and all of its ramifications. Because the finding was of very low safety significance and has been entered into the licensee's CAP as CR-RBS-2006-02402, this violation is being treated as an NCV consistent with Section VI.A of the Enforcement Policy: NCV 05000458/2006003-01, "Failure to identify Division III ESF bus supply breaker not racked in."

1R19 Postmaintenance Testing

a. Inspection Scope

For the five postmaintenance test activities of risk significant systems or components listed below, the inspectors: (1) reviewed the applicable licensing basis and/or design-basis documents to determine the safety functions; (2) evaluated the safety functions that may have been affected by the maintenance activity; and (3) reviewed the test procedure to verify that it adequately tested the safety function that may have been affected. The inspectors either witnessed or reviewed test data to verify that acceptance criteria were met, plant impacts were evaluated, test equipment was calibrated, procedures were followed, jumpers were properly controlled, the test data results were complete and accurate, the test equipment was removed, the system was properly re-aligned, and deficiencies during testing were documented. The inspectors also reviewed the CAP to determine if the licensee identified and corrected problems related to postmaintenance testing.

- Work Order (WO) 50370422, Division II battery cell post seal replacement, reviewed during the week of May 8, 2006

- WO 87721, replace control Rods 40-37, 44-41, and 48-25 and 12-25 individual scram test switches, reviewed May 19, 2006

- WO 69816, low pressure core spray keep fill pump discharge check valve, E21-VF033 replacement, reviewed during the week of June 19, 2006

- WO 85194, signature testing on high pressure core spray room unit cooler service water outlet valve, SWP-MOV74B, reviewed during the week of June 19, 2006

- WO 90342, Division II EDG generator output Breaker ENS-SWG1B-ACB027 charging springs failed to charge during tagout restoration, reviewed on June 23, 2006

The inspectors completed five inspection samples.

g. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities

a. Inspection Scope

The inspectors reviewed the following risk important refueling outage activities to verify defense in depth commensurate with the outage risk control plan and compliance with the TS during RFO-13 from April 23 to May 12, 2006: (1) the risk control plan; (2) tagging/clearance activities; (3) reactor coolant system instrumentation; (4) electrical

power; (5) decay heat removal; (6) spent fuel pool cooling; (7) inventory control; (8) reactivity control; (9) containment closure; (10) reduced inventory conditions; (11) refueling activities; (12) heatup and cooldown activities; (13) restart activities; and (14) licensee identification and implementation of appropriate corrective actions associated with RFO activities. The inspectors' containment inspections included observations of the containment sump for damage and debris, and supports, braces, and snubbers for evidence of excessive stress, water hammer, or aging. Specific outage activities observed and reviewed included:

- Outage risk assessment team (ORAT) report to onsite safety review committee
- Reactor shutdown, cooldown, and vessel disassembly
- Refueling operations, fuel sipping, and off loaded fuel inspections
- Daily/shiftly shutdown operations protection plan assessments
- Shutdown postscram report to onsite safety review committee
- Reactor recirculation pump trip logic modification installation and testing
- Main steam line local leak rate testing
- Transformer RSS1 offsite power line equipment inspection and upgrade
- Division II to Division I protected division swap
- Infrequently performed test or evolution briefings for:
  - Divisional loss of offsite power/loss of coolant accident testing
  - Concurrent control rod mechanism and blade changeout
  - Reactor vessel pressure test and scram time testing
  - Reactor startup, heatup, and power ascension
  - Onsite safety review committee meeting to recommend startup
  - Drywell 900 psi walkdown (after shutdown and during startup)

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one inspection sample.

b. Findings

Introduction: An NRC identified NCV of 10 CFR 50.65, “Maintenance Rule,” Section (a)(4) was identified for the failure of the licensee to provide prescribed compensatory measures for the highest shutdown risk condition during RFO-13. Specifically, the preoutage risk assessment recommended that two WOs be in place for maintenance electricians to provide power to one spent fuel pool cooling pump in the event of problems with the running pump during periods of safety-related electrical bus maintenance. The inspectors found that the WOs were not in place before entering shutdown risk condition Orange on April 26, 2006, during the Division II ESF bus testing, and on May 3, 2006, during the Division I ESF bus outage.

Description: The inspectors observed the onsite safety review committee meeting to discuss and approve the ORAT report for RFO-13. The report noted two Orange shutdown risk conditions for spent fuel pool cooling (SFC). Only one SFC pump would be available after the beginning of core offload: (1) during the Division II ESF bus testing with the SFC-P1B breaker racked out, and (2) during the Division I ESF bus outage when SFC-P1A was without power. As a result of the ORAT review of

Procedure AOP-0051, “Loss of Decay Heat Removal,” Revision 17, they recommended that the planned maintenance optimization group develop WOs for maintenance electricians to provide alternate power from the station blackout diesel generator to the deenergized SFC pump in the event of a failure of the running pump.

In addition, Procedure OSP-0037, “Shutdown Operations Protection Plan,” Revision 16, Section 4.7, “Fuel Pool Cooling,” required that: (1) if work was required on SFC during the outage, then it should be done as early as possible in the outage and not after fuel offload (when heat load is the highest); and (2) if work was required after fuel offload, then a contingency plan shall be in place prior to removing the system from service. The inspectors determined that this requirement applied to deenergizing an SFC pump for electrical bus maintenance.

On May 3, 2006, during the Division I ESF bus outage, the inspectors asked the operations shift manager if the required WO was available to provide alternate power to SFC-P1A in the event that the running SFC-P1B failed. He stated that he assumed that the WO was written and that he would check. The inspectors then requested a copy of the WO and a senior work planner reported that the WO was not available since it was not yet approved for use in the electronic work planning program. Following discussions with operators in the work management center, the licensee immediately took actions to ensure that both WOs were processed and made ready for use.

The inspectors reviewed AOP-0051, Attachment 1, “Spent Fuel Pool Curves,” and determined that the approximate “time to boil” for the spent fuel pool at that time with offload fuel in the pool was approximately 8 hours. Based on that data and the time needed to generate the WOs, the inspectors determined that there was adequate time for the licensee to connect an alternate power supply to the SFC pumps before the spent fuel pool water started to boil if there was a failure of the running pump.

Analysis: The performance deficiency associated with this finding involved the failure to establish prescribed compensatory measures for the highest outage risk condition during RFO-13 as required by the shutdown operations protection plan. The finding was more than minor because the licensee failed to implement prescribed compensatory measures and failed to effectively manage those measures. The finding affected the mitigating system cornerstone because of the increased risk of a sustained loss of SFC during core offloading operations. The finding could not be evaluated using the significance determination process; therefore, the finding was reviewed by regional management and determined to be of very low safety significance. Factors that were considered included: (1) electrical maintenance technicians had previously performed the task of providing alternate power to an SFC pump, (2) the necessary equipment was staged as part of the abnormal operating procedure for loss of decay heat removal, and (3) the relatively long “time to boil” of the spent fuel storage pool at that time during the refueling outage. The cause of the finding was related to the crosscutting aspect of human performance because the licensee’s planned maintenance activities and the predetermined increase in outage risk was not effectively managed by prescribed compensatory measures.

Enforcement: 10 CFR 50.65(a)(4) requires, in part, that before performing maintenance activities, the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities. Contrary to this, the licensee failed to properly manage the highest outage risk condition of RFO-13. On April 26, 2006, the plant entered an Orange outage risk condition for SFC during core offload, when SFC-P1B was deenergized for Division II ESF bus testing. On May 3, 2006, the plant entered an Orange outage risk condition for SFC during core offload, when SFC-P1A was deenergized for a Division I ESF bus outage. WOs were not written and ready for use to have electricians provide alternate power to an SFC pump in the event the running pump failed. The root cause involved the failure of the licensee to ensure that the WO was in place before the plant entered the Orange shutdown risk condition. Corrective action was taken to process the WOs for immediate use. Because the finding was of very low safety significance and was entered into the licensee's CAP as CR-RBS-2006-01937, this violation is being treated as an NCV consistent with Section VI.A of the Enforcement Policy: NCV 05000458/2006003-02, "Failure to adequately manage an increase in plant risk."

1R22 Surveillance Testing

a. Inspection Scope

The inspectors reviewed the USAR, procedure requirements, and TS to ensure that the six surveillance activities listed below demonstrated that the SSCs tested were capable of performing their intended safety functions. The inspectors either witnessed or reviewed test data to verify that the following significant surveillance test attributes were adequate: (1) preconditioning; (2) evaluation of testing impact on the plant; (3) acceptance criteria; (4) test equipment; (5) procedures; (6) jumper/lifted lead controls; (7) test data; (8) testing frequency and method demonstrated TS operability; (9) test equipment removal; (10) restoration of plant systems; (11) fulfillment of ASME Code requirements; (12) updating of performance indicator (PI) data; (13) engineering evaluations, root causes, and bases for returning tested SSCs not meeting the test acceptance criteria were correct; (14) reference setting data; and (15) annunciator and alarm setpoints. The inspectors also verified that the licensee identified and implemented any needed corrective actions associated with the surveillance testing.

- STP-208-3601, "'A' Main Steam Line MSIV's and Outboard Drain Valve Leak Rate Test and Inboard MSIV Inleakage Test," Revision 6, performed on May 2, 2006

- STP-305-1606, "[Division I Battery] ENB-BAT1A Service Discharge Test," Revision 17, performed on May 6, 2006

- STP-050-3601, "Shutdown Margin Demonstration," Revision 27, performed on May 12, 2006

- STP-000-0102, "Power Distribution Alignment Check," Revision 5, performed on May 14 and 15, 2006

- STP-508-4543, “Turbine First Stage Pressure Channel Functional Test,” Revision 7, performed on June 4, 2006

- Reactor coolant sample using Procedures COP-0001, “Sampling via Various Balance-Of-Plant Systems,” Attachment 8, “Reactor Sample Panel Routine Sample Points,” Revision 14, and COP-0305, “Operation of the Countroom Analysis Systems,” Revision 2, performed on June 15, 2006

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed six inspection samples.

h. Findings

Introduction: The inspectors identified an NCV of TS 5.4.1.a for the failure of the licensee to provide an adequate surveillance test procedure to perform TS Surveillance Requirement (SR) 3.8.1.1. Specifically, STP-000-0102, “Power Distribution Alignment Check,” Revision 4, did not include steps to verify the required offsite power circuit breaker alignment and indicated power availability for the Division III 4.16 kV ESF bus as required in Modes 1, 2, and 3.

Description: As discussed in Section 1R15 of this report, operators failed to properly rack in Breaker NNS-ACB23 on April 29, 2006. This condition was discovered on May 22, when the breaker failed to close. During this period, on May 14, 2006, the Division I EDG was removed from service to replace a leaking section of jacket cooling water vent tubing. With the Division I EDG removed from service, TS Required Action 3.8.1.a.1 required that operators perform TS SR 3.8.1.1 within one hour and once every 8 hours until the EDG was operable. TS SR 3.8.1.1 required operators to verify the correct breaker alignment and indicated power for each required offsite power circuit. Operators utilized Procedure STP-000-0102, “Power Distribution Alignment Check,” Revision 4, to satisfy the requirements of TS SR 3.8.1.1; however, the inspectors identified that the procedure did not have steps to verify the correct breaker alignment and indicated power availability to the Division III 4.16 kV ESF bus. As a result, the operators did not identify that Breaker NNS-ACB23 was not racked in. During the period that the Division I EDG was removed from service, the plant was actually in TS Condition 3.8.1.d. TS Condition 3.8.1.d states that with “One required offsite circuit inoperable AND one required [E]DG inoperable,” restore the EDG or the offsite power supply to an operable status in 12 hours or place the plant in Mode 3 within the next 12 hours. The Division I EDG was inoperable for 15 hours and 15 minutes.

Procedure STP-000-0102, Section 1.1, states, in part, that its purpose is to verify the correct breaker alignment and indicated power availability for each required offsite power circuit in accordance with TS SR 3.8.1.1 in Modes 1, 2, and 3. TS 3.8.1 bases defines an offsite power circuit as follows: “Each offsite circuit consists of incoming breakers and disconnects to the respective preferred station service Transformers 1C and 1D [RSS1 and RSS2], the 1C and 1D preferred station service transformers, and the respective circuit path including feeder breakers to the three 4.16 kV ESF buses.”

NNS-ACB23 is one of the circuit breakers between preferred station service Transformer RTX-XSR1C and the Division III 4.16 kV ESF bus.

Analysis: The performance deficiency associated with this finding involved the licensee's failure to provide operators with an adequate STP to meet the requirements of TS SR 3.8.1.1 to verify correct breaker alignment and indicated power availability to the Division III ESF bus for each required offsite circuit. A review of previous revisions of STP-000-0102 showed that the procedure has never verified the required offsite power circuits for the Division III 4.16 kV ESF bus in Modes 1, 2, and 3. Although this performance deficiency caused the failure to verify the offsite power circuit for an extended period of time, the risk impact was limited to the 10 days from May 12-22, 2006. Therefore, the risk characterization of this finding is the same as that described in Section 1R15 of this inspection report. The cause of the finding was related to the crosscutting aspect of human performance because the licensee did not provide the operators with an adequate STP to complete the TS SR to verify the required offsite power circuits' breaker alignment to all three 4.16 kV ESF buses. Additionally, the cause of the finding was related to the crosscutting aspect of problem identification and resolution in that on two occasions, June 18, 2005, and May 22, 2006, operators entered TS Condition 3.8.1.a for one inoperable offsite power circuit to the Division III 4.16 kV ESF bus and performed STP-000-0102 to meet the Required Action to perform SR 3.8.1.1, but did not recognize that STP-000-0102 did not verify the other offsite power circuit breaker alignment to the Division III 4.16 kV ESF bus.

Enforcement: TS 5.4.1.a requires that written procedures be established, implemented, and maintained covering the activities specified in Appendix A, "Typical Procedures for Pressurized Water Reactors and Boiling Water Reactors," of Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," dated February 1978. Regulatory Guide 1.33, Appendix A, Section 8.a, requires procedures for all TS SRs. Procedure STP-000-0102 states that it verified the correct breaker alignment and power availability for each required offsite circuit in accordance with TS SR 3.8.1.1 in Modes 1, 2, and 3. Contrary to this, Procedure STP-000-0102, Revision 4, did not require verification of the correct breaker alignment for the offsite power circuits to the Division III 4.16 kV ESF bus in Modes 1, 2, and 3. The root cause involved the incorrect interpretation of the Division III 4.16 kV bus SRs as they apply to the unique River Bend Station ESF electrical distribution system. The corrective actions to restore compliance included as an interim measure entering in the control room logs the breaker alignment for and the bus voltage available to the Division III 4.16 kV ESF bus, until STP-000-0102 could be revised. Because the finding was of very low safety significance and has been entered into the licensee's CAP as CR-RBS-2006-02675 and -02402, this violation is being treated as an NCV consistent with Section VI.A of the Enforcement Policy: NCV 05000458/2006003-03, "Inadequate procedure to verify required offsite power breaker alignment."

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the USAR, plant drawings, procedure requirements, and TS to ensure that Temporary Alteration 2006-0011, Off Gas Pretreatment Radiation Monitor Sample Chamber Drain Line Modification, was properly implemented. The inspectors: (1) verified that the modification did not have an affect on system operability/availability; (2) verified that the installation was consistent with modification documents; (3) ensured that the postinstallation test results were satisfactory and that the impact of the temporary modification on the operation of the pretreatment radiation monitor were supported by the test; (4) verified that the modification was identified on control room drawings and that appropriate identification tags were placed on the affected drawings; and (5) verified that appropriate safety evaluations were completed. The inspectors verified that the licensee identified and implemented any needed corrective actions associated with temporary modifications.

The inspectors completed one inspection sample.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation

a. Inspection Scope

On June 20, 2006, the inspectors observed the full scope exercise dress rehearsal, which was used to contribute to “Drill/Exercise Performance” and “Emergency Response Organization Drill Performance” PI. The inspectors: (1) observed the training evolution to identify any weaknesses and deficiencies in classification, notification, and protective action requirements development activities; (2) compared the identified weaknesses and deficiencies against licensee identified findings to determine whether the licensee was properly identifying failures; and (3) determined whether licensee performance was in accordance with the guidance of the NEI 99-02, "Voluntary Submission of Performance Indicator Data," Revision 2, acceptance criteria. The scenario used was RDRL-EP-0602, Tornado/Loss of Offsite Power/Main Steam Line Break, dated June 16, 2006.

Emergency [plan] implementing procedures reviewed by the inspectors included:

- EIP-2-001, “Classification of Emergencies,” Revision 13
- EIP-2-006, “Notifications,” Revision 32
- EIP-2-007, “Protective Action Guidelines Recommendations,” Revision 21

The inspectors completed one inspection sample.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas

a. Inspection Scope

This area was inspected to assess the licensee's performance in implementing physical and administrative controls for airborne radioactivity areas, radiation areas, high radiation areas, and worker adherence to these controls. The inspector used the requirements in 10 CFR Part 20, TS, and the licensee's procedures required by TS as criteria for determining compliance. During the inspection, the inspector interviewed the radiation protection manager, radiation protection supervisors, and radiation workers. The inspector performed independent radiation dose rate measurements and reviewed the following items:

- PI events and associated documentation packages reported by the licensee in the occupational radiation safety cornerstone
- Controls (surveys, posting, and barricades) of three radiation, high radiation, or airborne radioactivity areas
- Radiation work permits, procedures, engineering controls, and air sampler locations
- Conformation of electronic personal dosimeter alarm setpoints with survey indications and plant policy; workers' knowledge of required actions when their electronic personnel dosimeter noticeably malfunctions or alarms
- Barrier integrity and performance of engineering controls in airborne radioactivity areas
- Adequacy of the licensee's internal dose assessment for any actual internal exposure greater than 50 millirem committed effective dose equivalent
- Physical and programmatic controls for highly activated or contaminated materials (nonfuel) stored within spent fuel and other storage pools.
- Self-assessments, audits, licensee event reports (LER), and special reports related to the access control program since the last inspection
- Corrective action documents related to access controls

- Licensee actions in cases of repetitive deficiencies or significant individual deficiencies

- Radiation work permit briefings and worker instructions

- Adequacy of radiological controls, such as required surveys, radiation protection job coverage, and contamination controls during job performance

- Dosimetry placement in high radiation work areas with significant dose rate gradients

- Changes in licensee procedural controls of high dose rate - high radiation areas and very high radiation areas

- Controls for special areas that have the potential to become very high radiation areas during certain plant operations

- Posting and locking of entrances to all accessible high dose rate - high radiation areas and very high radiation areas

- Radiation worker and radiation protection technician performance with respect to radiation protection work requirements

The inspector completed 21 of the required 21 samples.

b. Findings

1. Unguarded High Radiation Area Boundary

Introduction: The inspector reviewed a self-revealing NCV of TS 5.7.1, resulting from the licensee's failure to control access to a high radiation area. The finding had very low safety significance.

Description: On April 6, 2006, the licensee transferred reverse osmosis system filters from one elevation of the radwaste building to another. Because dose rates on the filter barrels were as high as 600 millirem per hour, the licensee assigned personnel to guard the elevator entrances to prevent workers from entering high radiation areas. On this occasion, the guards were not using radios, as was a common practice. Because of the lack of good communication, a guard prematurely left his post in front of the 123-foot elevation elevator door. Coincidently, two workers attempted to board the elevator on the 123-foot elevation after the guard had left. The elevator carrying the barrels of radioactive filters stopped at the 123-foot elevation, the doors opened, and the electronic dosimeters of the workers alarmed because of the high dose rates. The guard returned and evacuated the workers before they accrued additional radiation dose. The highest dose rate recorded by an electronic alarming dosimeter was 164 millirem per hour. Planned corrective action was still being evaluated by the licensee at the conclusion of the inspection.

Analysis: The failure to control access to a high radiation area was a performance deficiency. The significance of the finding was greater than minor because it was associated with the occupational radiation safety attribute of exposure control and affected the cornerstone objective, in that not controlling access to a high radiation area could increase personal exposure. Using the Occupational Radiation Safety Significance Determination Process, the inspector determined that the finding was of very low safety significance because it did not involve: (1) an as low as is reasonably achievable (ALARA) finding, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. Additionally, this finding had crosscutting aspects associated with human performance in that the failure of the individual to guard the elevator door directly contributed to the violation.

Enforcement: TS 5.7.1 requires each high radiation area, as defined in 10 CFR Part 20, in which the intensity of radiation is greater than 100 millirems per hour but less than 1000 millirems per hour, be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a radiation work permit. The licensee violated TS 5.7.1 when it failed to barricade and conspicuously post the elevator housing the radioactive filter barrels or maintain a guard to ensure workers did not enter a high radiation area. Because this failure to control a high radiation area was of very low safety significance and has been entered into the licensee's CAP as CR-RBS-2006-01294, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy:
NCV 05000458/2006003-04, "Failure to control access to a high radiation area."

2. Unanalyzed Airborne Radioactivity Survey

Introduction: The inspector identified an NCV of 10 CFR 20.1501(a) because the licensee failed to survey airborne radioactivity. The finding had very low significance.

Description: On May 2, 2006, during the removal of local power range monitors, the licensee started collecting an air sample of the work area. The air sample spanned two shifts. A health physics technician on the second shift discarded the sample because the first shift had not documented a start time. Therefore, the sample was never analyzed. However, all workers successfully passed through the portal monitors at the exit of the controlled access area without alarm, confirming that no worker experienced an uptake of radioactive material. Planned corrective action is still being evaluated.

Analysis: The failure to survey airborne radioactivity was a performance deficiency. This finding was greater than minor because it was associated with the occupational radiation safety program attribute of exposure control and affected the cornerstone objective in that the lack of knowledge of radiological conditions could increase personnel dose. Using the Occupational Radiation Safety Significance Determination Process, the inspector determined that the finding was of very low safety significance because it did not involve: (1) an ALARA finding, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. Additionally, this finding had crosscutting aspects associated with human performance in that the failure to maintain the sample for analysis directly contributed to the violation.

Enforcement: 10 CFR 20.1501(a) requires that each licensee make or cause to be made surveys that may be necessary for the licensee to comply with the regulations in 10 CFR Part 20 and that are reasonable under the circumstances to evaluate the extent of radiation levels, concentrations or quantities of radioactive materials, and the potential radiological hazards that could be present. Pursuant to 10 CFR 20.1003, a "survey" means 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. In part, 10 CFR 20.1201(a) states that the licensee shall control the occupational dose to individual adults. The licensee violated 10 CFR 20.1501(a) when it failed to perform an evaluation of airborne radioactivity to ensure compliance with 10 CFR 20.1201(a). Because this failure to perform a radiological survey was of very low safety significance and has been entered into the licensee's CAP as CR-RBS-2006-01994, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000458/2006003-05, "Failure to perform airborne radiation survey."

2OS2 ALARA Planning and Controls

a. Inspection Scope

The inspector assessed licensee performance with respect to maintaining individual and collective radiation exposures ALARA. The inspector used the requirements in 10 CFR Part 20 and the licensee's procedures required by TS as criteria for determining compliance. The inspector interviewed licensee personnel and reviewed:

- Current 3-year rolling average collective exposure
- Three outage or on-line maintenance work activities scheduled during the inspection period and associated work activity exposure estimates which were likely to result in the highest personnel collective exposures
- ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements
- Intended versus actual work activity doses and the reasons for any inconsistencies
- Shielding requests and dose/benefit analyses
- Dose rate reduction activities in work planning
- Use of engineering controls to achieve dose reductions and dose reduction benefits afforded by shielding
- Workers use of the low dose waiting areas
- First-line job supervisors' contribution to ensuring work activities are conducted in a dose efficient manner

- Radiation worker and radiation protection technician performance during work activities in radiation areas, airborne radioactivity areas, or high radiation areas

The inspector completed 6 of the required 15 samples and 4 of the optional samples.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

a. Inspection Scope

1. Barrier Integrity Cornerstone

The inspectors sampled licensee submittals for the two PIs listed below for the period October 1, 2004, through March 31, 2006. The definitions and guidance of NEI 99-02, “Regulatory Assessment Indicator Guideline,” Revision 4, were used to verify the licensee’s basis for reporting each data element in order to verify the accuracy of PI data reported during the assessment period. The inspectors: (1) reviewed reactor coolant system (RCS) chemistry sample analyses for dose equivalent Iodine-131 and compared the results to the TS limit; (2) observed a chemistry technician obtain and analyze an RCS sample; (3) reviewed operating logs and surveillance results for measurements of RCS identified leakage; and (4) observed a surveillance test that determined RCS identified leakage.

- RCS Specific Activity
- RCS Leakage

The inspectors completed two inspection samples.

2. Occupational Radiation Safety Cornerstone

The review included corrective action documentation that identified occurrences in locked high radiation areas (as defined in the licensee’s TS), very high radiation areas (as defined in 10 CFR 20.1003), and unplanned personnel exposures (as defined in NEI 99-02), specifically CR-RBS-2006-01910. Additional records reviewed included ALARA records and whole-body counts of selected individual exposures. The inspector interviewed licensee personnel that were accountable for collecting and evaluating the PI data. In addition, the inspector toured plant areas to verify that high radiation, locked high radiation, and very high radiation areas were properly controlled. PI definitions and guidance contained in NEI 99-02, "Regulatory Assessment Indicator Guideline," Revision 3, were used to verify the basis in reporting for each data element.

- Occupational Exposure Control Effectiveness

The inspector completed the one required sample in this cornerstone.

3. Public Radiation Safety Cornerstone

The inspector reviewed licensee documents from June 1, 2005, through March 31, 2006. Licensee records reviewed included corrective action documentation that identified occurrences for liquid or gaseous effluent releases that exceeded PI thresholds and those reported to the NRC. The inspector interviewed licensee personnel that were accountable for collecting and evaluating the PI data. PI definitions and guidance contained in NEI 99-02, "Regulatory Assessment Indicator Guideline," Revision 3, were used to verify the basis in reporting for each data element.

- Radiological Effluent Technical Specification/Offsite Dose Calculation Manual Radiological Effluent Occurrences

The inspector completed the one required sample in this cornerstone.

f. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

1. Semiannual Trend Review

g. Inspection Scope

The inspectors completed a semiannual trend review of repetitive or closely related issues related to identify trends that might indicate the existence of more safety significant issues. The inspectors' review consisted of the 6-month period from January 1 to June 30, 2006, of CAP items associated with the three EDG starting air systems documented in 42 CRs. When warranted, some of the samples expanded beyond those dates to fully assess the issue. The inspectors compared and contrasted their results with the results contained in adverse trend CRs for problems related to the starting air compressors and air dryers. Corrective actions associated with a sample of the issues identified were reviewed for adequacy. The CRs reviewed by the inspectors are listed in the attachment.

The inspectors completed one inspection sample.

b. Findings and Observations

There were no findings of significance identified associated with the CRs reviewed.

The inspectors noted that the licensee had identified a long-standing issue related to the performance of the EDG starting air systems' air compressors. Since January 1, 2006,

there were 18 CRs written for high metal wear products in monthly air compressor oil samples. Each of these CRs was closed to CR-RBS-2004-02165. An additional 28 CRs written since August 2, 2004, for high metal wear product concentrations and high moisture content in monthly compressor oil samples were closed to CR-RBS-2004-02165. In addition, operators wrote adverse trend CR-RBS-2006-02407 to detail compressor problems, including excessive run times. The inspectors determined that the licensee is taking appropriate actions to understand the problem with the EDG starting air compressors, including sending the system engineer to observe the vendor's teardown and refurbishment of two of the starting air compressors.

Another four CRs have been written since January 1, 2006, describing problems with starting air system dryers and dryer prefilters. Following a June 29, 2006, meeting held to discuss overall EDG starting air system maintenance problems, the licensee wrote CR-RBS-2006-02799, to look into the relationship between the prefilter and dryer problems. The inspectors noted that this meeting was the first discussion of the overall condition of the EDG starting air systems and to evaluate the interrelationship between compressor, dryer, and prefilter problems.

2. Occupational Radiation Safety

a. Inspection Scope

The inspector evaluated the effectiveness of the licensee's problem identification and resolution process with respect to the following inspection areas:

- Access Control to Radiologically Significant Areas (Section 2OS1)
- ALARA Planning and Controls (Section 2OS2)

b. Findings and Observations

No findings of significance were identified.

3. Inservice Inspection Activities

a. Inspection Scope

The inspector reviewed selected inservice inspection related CRs issued during the current and past refueling outages. The review served to verify that the licensee's CAP was being correctly utilized to identify conditions adverse to quality and that those conditions were being adequately evaluated, corrected, and trended.

b. Findings

No findings of significance were identified.

4OA3 Event Followup

1. (Closed) LER 50-458/2004-003-01, Unplanned Automatic Start of Standby Diesel Generator Due to Loss of Division 1 Switchgear

   On October 31, 2004, technicians caused an unexpected degraded voltage signal, which resulted in a loss of the Division I 4.16 kV ESF bus during preparations for the Division I loss of offsite power/loss of coolant accident test. The Division I EDG automatically started and powered the ESF bus and all equipment operated as expected. Initial inspection of this event was documented in NRC integrated inspection Report 05000458/2004005. During this inspection period, the inspectors reviewed the LER, the root cause analysis, and corrective actions documented in CR-RBS-2004-03518. No additional findings of significance were identified. This LER is closed.

2. (Closed) LER 50-458/2004-004-01, Unplanned Automatic Start of Standby Diesel Generator Due to Loss of Division 2 Switchgear

   On November 1, 2004, technicians inadvertently caused a trip of Transformer RSS2 preferred station service Transformer RTX-XSR1F while troubleshooting a transformer sudden pressure relay trip circuit. As a result, power was also lost to preferred station Transformer RTX-XSR1D and the Division II 4.16 kV ESF bus. The running shutdown cooling, alternate decay heat removal, and plant operating water cleanup systems lost power until the Division II EDG started and restored power to the ESF bus. Shutdown cooling was restored in less than one hour. Initial inspection of this event was documented in NRC integrated inspection Report 05000458/2004005. During this inspection period, the inspectors reviewed the LER, the root cause analysis, and corrective actions documented in CR-RBS-2004-03546. No additional findings of significance were identified. This LER is closed.

3. (Closed) LER 50-458/2004-005-01, Unplanned Automatic Scram Due to Loss of Non-Vital 120 Volt Instrument Bus

   On December 10, 2004, an automatic scram occurred due to a loss of power to nonsafety-related instrumentation Bus VBN-PNL01B1. A capacitor on the control board for the nonsafety-related Inverter BYS-INV01B static switch failed, which caused a loss of power to Bus VBN-PNL01B1, a subsequent downshift of the plant operating recirculation pumps and a lockup of the main feedwater regulating valves. The result was an automatic plant scram complicated by a loss of normal feedwater. Inspection of this event was documented in NRC integrated inspection Report 05000458/2004005. Additional inspection was documented in NRC supplemental inspection Report 05000458/2005012. During this inspection period, the inspectors reviewed the LER, the root cause analysis, and corrective actions documented in CR-RBS-2004-04289. No additional findings of significance were identified. This LER is closed.

4. (Closed) LER 50-458 /2005-001-01, Unplanned Manual Scram Due to Indication of Ground Fault in Main Generator

On January 15, 2005, while the plant was at 100 percent power, a main generator field ground fault alarm was received. Control room operators tripped the turbine in accordance with alarm response Procedure ARP-680-09. The licensee later determined that one of the five rectifier banks in the generator excitation control system was the source of the ground and removed it from service. In addition, the licensee tested the relay that causes the main generator ground fault alarm and found it to be out of calibration such that it alarmed before the ground current reached its setpoint. The alarm response procedure requirement to trip the turbine was revised to allow validation of the alarm before tripping the main turbine. Inspection of this event was documented in NRC integrated inspection Report 05000458/2005002. Additional inspection was documented in NRC supplemental inspection Report 05000458/2005012. During this inspection period, the inspectors reviewed the LER, the root cause analysis, and corrective actions documented in CR-RBS-2005-00140. No additional findings of significance were identified. This LER is closed.

4OA5 Other Activities

Implementation of Temporary Instruction 2515/165 - Operational Readiness of Offsite Power and Impact on Plant Risk

a. Inspection Scope

The objective of Temporary Instruction 2515/165, "Operational Readiness of Offsite Power and Impact on Plant Risk," was to gather information to support the assessment of nuclear power plant operational readiness of offsite power systems and impact on plant risk. During this inspection, the inspectors interviewed licensee personnel, reviewed licensee procedures, and gathered information for further evaluation by the Office of Nuclear Reactor Regulation.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

Exit Meetings

On May 5, 2006, the inspector presented the occupational radiation safety inspection results to Mr. D. Vinci, General Manager, Plant Operations, and other members of his staff who acknowledged the findings. The inspector confirmed that proprietary information was not provided or examined during the inspection.

On May 5, 2006, the inspector presented the results of this inspection of inservice inspection activities to Mr. P. Russell, Manager, System Engineering, and other

members of licensee management. The inspector confirmed that proprietary information was not provided or examined during the inspection.

On July 5, 2006, the resident inspectors presented the integrated baseline inspection results to Mr. P. Hinnenkamp, Vice President - Operations, and other members of licensee management. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

T. Baccus, Acting Supervisor, ALARA Planning
L. Ballard, Manager, Quality Programs
D. Burnett, Superintendent, Chemistry
C. Bush, Manager, Outage
J. Clark, Assistant Operations Manager - Training
T. Coleman, Manager, Planning and Scheduling/Outage
M. Davis, Manager, Radiation Protection
C. Forpahl, Manager, Corrective Action Program
T. Gates, Manager, Equipment Reliability
H. Goodman, Director, Engineering
K. Higginbotham, Assistant Operations Manager - Shift
P. Hinnenkamp, Vice President - Operations
B. Houston, Manager, Plant Maintenance
A. James, Superintendent, Plant Security
K. Jenks, Supervisor, Engineering Codes and Standards
N. Johnson, Manager, Engineering Programs & Components
R. King, Director, Nuclear Safety Assurance
J. Leavines, Manager, Emergency Planning
D. Lorfing, Manager, Licensing
J. Maher, Superintendent, Reactor Engineering
W. Mashburn, Manager, Design Engineering
J. Miller, Manager, Training and Development
P. Russell, Manager, System Engineering
C. Stafford, Manager, Operations
D. Vinci, General Manager - Plant Operations

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000458/2006003-01	NCV	Failure to identify Division III ESF bus supply breaker not racked in
05000458/2006003-02	NCV	Failure to adequately manage an increase in plant risk
05000458/2006003-03	NCV	Inadequate procedure to verify required offsite power breaker alignment
05000458/2006003-04	NCV	Failure to control access to a high radiation area
05000458/2006003-05	NCV	Failure to perform airborne radiation survey

Closed

50-458/2004-003-01	LER	Unplanned Automatic Start of Standby Diesel Generator Due to Loss of Division 1 Switchgear
50-458/2004-004-01	LER	Unplanned Automatic Start of Standby Diesel Generator Due to Loss of Division 2 Switchgear
50-458/2004-005-01	LER	Unplanned Automatic Scram Due to Loss of Non-Vital 120 Volt Instrument Bus
50-458 /2005-001-01	LER	Unplanned Manual Scram Due to Indication of Ground Fault in Main Generator

LIST OF 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:

Section 1R06: Inservice Inspection Activities

Procedures

CEP-NDE-0400, “Ultrasonic Examination,” Revision 0

CEP-NDE-0404, “Manual Ultrasonic Examination of Ferritic Piping Welds (ASME XI),” Revision 1

CEP-NDE-0407, “Straight Beam Ultrasonic Examination of Bolts and Studs (ASME XI),” Revision 1

CEP-NDE-0423, “Manual Ultrasonic Examination of Austenitic Piping Welds (ASME XI),” Revision 1

CEP-NDE-0424, “Manual Ultrasonic Examination of the Reactor Vessel Flange Ligament Areas (ASME XI),” Revision 1

CEP-NDE-0428, “Manual Ultrasonic Throughwall Sizing in Piping Welds (ASME XI),” Revision 1

CEP-NDE-0641, “Liquid Penetrant Examination for ASME Section XI,” Revision 1

CEP-NDE-0731, “Magnetic Particle Examination (ASME Section XI),” Revision 0

SPP-7010, “Preparation of Weld Data Documents,” Revision 9

Miscellaneous Documents

7228.000-701-131A, “Risk Informed Break Exclusion Region Evaluation for River Bend Station,” Revision 0

Liquid Penetrant Examinations

BOP-PT-06-024	BOP-PT-06-025	BOP-PT-06-026	BOP-PT-06-029

UT Calibration Reports

CAL -06-015	CAL -06-016	CAL-06-017

UT Pipe Weld Examinations

ISI-UT-06-003	ISI-UT-06-006	ISI-UT-06-009	ISI-UT-06-012
ISI-UT-06-004	ISI-UT-06-007	ISI-UT-06-010	ISI-UT-06-013
ISI-UT-06-005	ISI-UT-06-008	ISI-UT-06-011	ISI-UT-06-014

Condition Reports

CR-RBS-2005-00065	CR-RBS-2005-00067	CR-RBS-2005-00100	CR-RBS-2005-01379

Section 1R15: Operability Evaluations

Primary Containment Purge Exhaust Line Operability

CR-RBS-2006-00964, primary containment purge exhaust line leak rate test results showing negative trend

ADM-0050, “Primary Containment Leakage Rate Testing Program,” Revision 8

SEP-APJ-001, “Primary containment Leakage Rate Testing (Appendix J) Program,” Revision 0G

STP-403-7301, “Containment Purge System Isolation Valve Leak Rate Test,” Revisions 0, 1, 2, and 3

RBS-ER-00-0589, “Post RF-09 LLRT Testing Interval Determination,” dated January 25, 2001

RBS TS Amendment 81, dated July 20, 1995

RBS TS Bases Revision 126, dated March 31, 206

NNS-ACB23 Not Functional

Electrical Drawings

EE-001AC, “Startup Electrical Distribution Chart,” Revision 33

ESK-05NNS03, “Elementary Diagram - 4.16 kV Switchgear Bus 1C Normal Supply ACB,” Revision 13

Corrective Action Documents

CR-RBS-2006-02402 CR-RBS-2006-0235 CR-RBS-2006-02337
CR-RBS-1998-00190

Procedures

OSP-0022, “Operations General Administrative Guidelines,” Revision 01
GOP-0001, “Plant Startup,” Revision 47, performed on May 12, 2006
STP-000-0102, “Power Distribution Alignment Check,” Revision 4, performed on May 9, 2006
STP-000-0102, “Power Distribution Alignment Check,” Revision 4, performed on May 22, 2006

Work Requests

WR 76625 WR 77441 WR77478

Miscellaneous Documents

Main Control Room Logs
TS LCO Records: 1-OPT-06-0187 1-TS-06-0694
RBS Tagout Record: 1-302-NNS-SWG1A-006-A

Section 1R20: Refueling and Other Outage Activities

Procedures

RSP-0217, “Auxiliary Access Control Functions,” Revision 27

GOP-0003, “Scram Recovery,” Revision 14A, post scram report, dated April 23, 2006

OSP-0031, “Shutdown Operations Protection Plan,” Revision 16

OSP-0041, “Alternate Decay Heat Removal,” Revision 8A

AOP-0051, “Loss of Decay Heat Removal,” Revision 18

OSP-0034, “Control of Obstructions for Primary Containment/Fuel Building Operability,” Revision 3

GOP-0001, “Plant Startup,” Revision 47, performed on May 12, 2006

Corrective Action Documents

CR-RBS-2006-00691 CR-RBS-2006-01937

Miscellaneous Documents

Control Room Logs

TS LCO Logs

Daily Refueling Outage Updates

ORAT Report

WO 50340401 and 81284

ER-RB-2005-0157-000, “Install new relays on the output of EOC-RPT optical output cards C71A-AT17 and C71A-AT18,” dated May 16, 2006

WO 5034041's task outline to configure the station blackout diesel to supply power to spent fuel pool cooling Pump SFC-P1A

WO 5034041, Configure the station blackout diesel to supply power to spent fuel pool cooling Pump SFC-P1A, written May 3, 2006

Section 1R22: Surveillance Testing

Drawing EE-001AC, “Startup Electrical Distribution Chart,” Revision 33

TS Section 3.8.1 and Bases 3.8.1, Revision 0

USAR Section 8.2.1.2.1, “General Design Criteria,” Revision 16

NUREG-0989, “Safety Evaluation Report Related to the Operation of River Bend Station,” dated May 1984

TS LCO Logs

1-TS-06-0694 I-TS-06-0685 1-TS-05-0386

Corrective Action Documents

CR-RBS-2006-02675 CR-RBS-2006-02402 CR-RBS-2005-02331

Section 4OA2: Identification and Resolution of Problems

Semiannual Trend Review

CR-RBS-2004-02165	CR-RBS-2006-01270	CR-RBS-2006-02469
CR-RBS-2006-00159	CR-RBS-2006-01324	CR-RBS-2006-02484
CR-RBS-2006-00226	CR-RBS-2006-01333	CR-RBS-2006-02540
CR-RBS-2006-00279	CR-RBS-2006-01429	CR-RBS-2006-02544
CR-RBS-2006-00296	CR-RBS-2006-01464	CR-RBS-2006-02550
CR-RBS-2006-00434	CR-RBS-2006-01489	CR-RBS-2006-02558
CR-RBS-2006-00663	CR-RBS-2006-01490	CR-RBS-2006-02559
CR-RBS-2006-00798	CR-RBS-2006-02269	CR-RBS-2006-02651
CR-RBS-2006-00799	CR-RBS-2006-02348	CR-RBS-2006-02661
CR-RBS-2006-00928	CR-RBS-2006-02349	CR-RBS-2006-02682
CR-RBS-2006-00993	CR-RBS-2006-02356	CR-RBS-2006-02683
CR-RBS-2006-01131	CR-RBS-2006-02375	CR-RBS-2006-02732
CR-RBS-2006-01132	CR-RBS-2006-02406	CR-RBS-2006-02733
CR-RBS-2006-01205	CR-RBS-2006-02407	CR-RBS-2006-02799
CR-RBS-2006-01261		

Section 2OS1: Access Controls to Radiologically Significant Areas

Corrective Action Documents

CR-RBS-2006-00090 CR-RBS- 2006-01294 CR-RBS-2006-01787 CR-RBS- 2006-01950

Radiation Work Permits

2006-1915	RFO-13, Remove and Replace LPRMs, Including Support Activities
2006-1921	RFO-13, Flow Control Valve Maintenance, Including Support Activities
2006-1929	RFO-13, Recirc Pump Work, Including Support Activities

Procedures

RP-103	Access Control, Revision 2
RP-106	Radiological Survey Documentation, Revision 1
RP-108	Radiation Protection Posting, Revision 2
RPP-0006	Performance of Radiological Surveys, Revision 19

Section 2OS2: ALARA Planning and Controls (71121.02)

Corrective Action Documents

CR-RBS-2006-01746

Procedures

ENS-RP-105 Radiation Work Permits, Revision 7

LIST OF ACRONYMS

CDF	core damage frequency
ALARA	as low as is reasonably achievable
ASME	American Society of Mechanical Engineers
CAP	corrective action program
CFR	*Code of Federal Regulations*
CR-RBS	River Bend Station condition report
EDG	emergency diesel generator
LER	licensee event report
MC	inspection manual chapter
NCV	noncited violation
NDE	nondestructive examination
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
ORAT	outage risk assessment team
PI	performance indicators
RCS	reactor coolant system
RFO	refueling outage
SFC	spent fuel pool cooling system
SOP	system operating procedures
SR	surveillance requirement
SSC	structures, systems, or components
STP	surveillance test procedure
TS	Technical Specifications
USAR	Updated Safety Analysis Report
WO	work order
WR	work request