UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-4005

August 7, 2007

Joseph E. Venable
Senior 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 05000458/2007003

Dear Mr. Venable:

On June 30, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your River Bend Station. The enclosed integrated inspection report documents the inspection findings, which were discussed on July 9, 2007, 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.

This report documents one NRC-identified finding and three self-revealing findings of very low safety significance (Green). Two of these findings were determined to involve violations of NRC requirements. Additionally, a licensee-identified violation which was determined to be of very low safety significance is listed in this report. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these findings as noncited violations (NCVs) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest any NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the 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, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at River Bend Station.

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/

Michael C. Hay, Chief
Project Branch C
Division of Reactor Projects

Docket: 50-458
License: NPF-47

Enclosure:
NRC Inspection Report 05000458/2007003
w/Attachment: Supplemental Information

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SUNSI Review Completed: WCW_ ADAMS: ■ Yes ☐ No Initials: _WCW_____
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RIV:SRI:DRP/C	SRI:DRP/C	SPE:DRP/C	C:DRS/EB1	C:DRS/PSB
MOMiller	PJAlter	WCWalker	DAPowers	MPShannon
T-WCWalker for	***E-WCWalker for***	***/RA/***	***/RA/***	***/RA/***
08/03/07	08/03/07	08/02/07	07/31/07	08/01/07

C:DRS/EB2	C:DRS/OB	C:DRP/C
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08/01/07	08/01/07	08/07/07

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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	50-458
License:	NPF-47
Report:	05000458/2007003
Licensee:	Entergy Operations, Inc.
Facility:	River Bend Station
Location:	5485 U.S. Highway 61 St. Francisville, Louisiana
Dates:	April 1 through June 30, 2007
Inspectors:	M. Miller, Senior Resident Inspector, Project Branch C P. Alter, Senior Resident Inspector, Project Branch C D. Tharp, Resident Inspector, Region III, Clinton W. Sifre, Senior Reactor Inspector, Engineering Branch 1 B. Henderson, Reactor Inspector, Engineering Branch 1 E. Owen, Reactor Inspector, Engineering Branch 1 P. Elkmann, Emergency Preparedness Inspector, Operations Branch
Approved By:	Michael C. Hay, Chief Project Branch C Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000458/2007003; 04/01/2007 - 06/30/2007; River Bend Station; Event Followup.

The report covered a 3-month period of routine inspections by resident inspectors and announced baseline inspections by region-based inspectors. Two Green noncited violations and two Green findings were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or 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: Initiating Events

- Green. A self-revealing noncited violation of Technical Specification 5.4.1.a was identified involving the failure to follow a surveillance procedure for scram discharge instrument volume water level channel calibration. Specifically, on February 9, 2007, an instrument line plug was not replaced following surveillance testing. As a result, on May 5, 2007, following a reactor scram, reactor water sprayed out of the scram discharge instrument volume and contaminated some accessible portions of the containment building causing three inadvertent personnel contamination events. This issue was entered into the licensee's corrective action program as condition Report CR-RBS-2007-01809.

  The finding was more than minor because it was associated with the initiating event cornerstone attribute of equipment performance and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during power operations. A Phase 2 estimation was required, as determined by the Manual Chapter 0609, Appendix A, Phase 1 Worksheet, "SDP Phase 1 Screening Worksheet for Initiating Events, Mitigation Systems, and Barriers Cornerstones," because the associated performance deficiency resulted in a reactor coolant leak greater than the Technical Specification limit for identified reactor coolant system leakage. Using the plant-specific Phase 2 risk-informed notebook, this violation was determined to have very low safety significance because the violation only increased the likelihood of a small-break loss of coolant accident by a very small amount and mitigation capability was unaffected. The cause of the finding was related to the human performance crosscutting component of work practices because neither self nor peer checking actions identified the failure to replace the vent plug (H.4(a)). (Section 4OA3)

- Green. A self-revealing finding was identified involving the failure to implement 1998 vendor recommendations associated with the potential for vibration induced degradation of recirculation loop gate valves. This resulted in the failure to identify and implement timely corrective actions prior to disk to stem separation of recirculation Pump A

discharge gate valve that occurred on May 21, 2007. This issue was entered into the licensee's corrective action program as condition Report CR-RBS-2007-02113.

The finding was more than minor because it was associated with the initiating events cornerstone attribute of equipment performance and affected the associated cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during power operations. Using the Manual Chapter 0609, “Significance Determination Process,” Phase 1 Worksheet, the finding was determined to have a very low safety significance because the finding did not contribute to the likelihood that mitigation equipment or functions would not be available following a reactor trip. (Section 4OA3)

- Green. A self-revealing finding was identified involving inadequate maintenance instructions for opening a stuck closed feedwater regulating Valve A isolation valve. Specifically, the instructions failed to account for the system being pressurized resulting in unexpected valve stem movement while technicians were removing the manual operator from the valve on June 10, 2007. This deficiency could have resulted in personnel harm or an unexpected and uncontrolled plant transient. This issue was entered into the licensee's corrective action program as condition Report CR-RBS-2007-02576.

  The finding was more than minor because it could become a more significant safety concern if left uncorrected. Using the Manual Chapter 0609, “Significance Determination Process,” Phase 1 Worksheet, the finding was determined to have very low safety significance because the deficiency did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available. No violation of NRC requirements occurred. The cause of this finding was related to the human performance crosscutting component of resources because the licensee did not ensure a complete and accurate work package was available prior to the start of the job (H.2(c)). (Section 4OA3)

Cornerstone: Barrier Integrity

- Green. A self-revealing noncited violation of Technical Specification 5.4.1.a was identified involving the failure to follow procedure. Specifically, during control rod withdrawal a reactor engineer noted that reactor power, as calculated by a heat balance, was inconsistent with predicted power. Although this inconsistency was identified the reactor engineers and operators failed to fully evaluate this condition, as required by procedure, and continued with power ascension resulting in an automatic rod withdrawal block. Upon further review the event was caused from feed flow and temperature data not automatically updating resulting in calculated power being less than actual power. This issue was entered into the licensee's corrective action program as condition Report CR-RBS-2007-01691.

  The finding was more than minor because it was associated with the barrier integrity cornerstone attribute of configuration control and it affected the cornerstone objective to provide reasonable assurance that physical design barriers, such as fuel cladding, protect the public from radio-nuclide releases caused by accidents or events. Using the

Manual Chapter 0609, “Significance Determination Process,” Phase 1 Worksheet, the finding was determined to have a very low safety significance because it did not have the potential to affect the integrity of the RCS barrier. The cause of this finding is related to the human performance cross cutting component of work practices because neither self nor peer checking actions prevented the automatic rod withdrawal block (H.4(a)). (Section 4OA3)

B. <u>Licensee-Identified Violations</u>

One violation of very low safety significance, which was identified by the licensee, has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee’s corrective action program. This violation and corrective actions are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status: The plant was operated between 100 percent and 96 percent power from April 1-9, 2007, because of main condenser vacuum concerns. On April 9, reactor power was reduced to 70 percent to support condenser cleaning. On April 11, reactor power was further reduced to 65 percent to support condenser cleaning. On April 19, the reactor was restored to 100 percent power. On April 24, reactor power was reduced to approximately 15 percent to support repairs to a leaking relief valve in the feedwater system. On April 28, the reactor was restored to 100 percent power. On May 4, the reactor was manually scrammed following a loss of main transformer cooling. On May 7, plant restart commenced and on May 10, the reactor was restored to 100 percent power. On May 21, reactor recirculation Loop A flow unexpectedly lowered with a corresponding drop in reactor power from 100 percent to 96 percent. Control room operators balanced flows between recirculation loops resulting in reducing reactor power to 90 percent. On May 22, the reactor was shutdown to support repairs to recirculation pump discharge gate valve A. On June 8, plant restart commenced. Full power was delayed as repairs were made to one of the reactor feedwater pumps. On June 18, the reactor reached 100 percent power and remained at 100 percent for the remainder of the inspection period, except for a normally scheduled down power for a control rod pattern adjustment on June 19, 2007.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

.1 Readiness For Seasonal Susceptibilities

During the week of June 4, 2007, the inspectors completed a review of the licensee's readiness for seasonal susceptibilities involving high winds and heavy rain at the beginning of hurricane season. The inspectors: (1) interviewed the emergency planning manager and members of the emergency planning staff to discuss changes in readiness since hurricane Katrina in 2005; (2) conducted in-office reviews of Procedure ENS-EP-302, “Severe Weather Response,” Revision 6, abnormal operating Procedure AOP-0029, “Severe Weather Operation,” Revision 19, 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; (3) walked down external 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; (4) evaluated staffing levels to verify the licensee could maintain the readiness of essential systems required by plant procedures; and (5) 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.

1R02 Evaluation of Changes, Tests, or Experiments

a. Inspection Scope

The inspectors reviewed the effectiveness of the licensee's implementation of changes to the facility structures, systems, and components (SSC); risk-significant normal and emergency operating procedures; test programs; and the USAR in accordance with 10 CFR 50.59, "Changes, Tests, and Experiments." The inspectors utilized Inspection Procedure 71111.02, "Evaluation of Changes, Tests, or Experiments," for this inspection.

The inspectors reviewed six safety evaluations performed by the licensee since the last NRC inspection of this area at River Bend Station. The evaluations were reviewed to verify that licensee personnel had appropriately considered the conditions under which the licensee may make changes to the facility or procedures or conduct tests or experiments without prior NRC approval. The inspectors reviewed eight licensee-performed applicability determinations and 10 screenings, in which licensee personnel determined that evaluations were not required, to ensure that the exclusion of a full evaluation was consistent with the requirements of 10 CFR 50.59. Evaluations, screenings, and applicability determinations reviewed are listed in the Attachment to this report.

The inspectors reviewed and evaluated a sample of recent licensee condition reports (CRs) to determine whether the licensee had identified problems related to 50.59 evaluations, entered them into the CAP, and resolved technical concerns and regulatory requirements. The reviewed CRs are identified in the Attachment to this report.

The inspection procedure specifies the inspectors review a minimum sample of five licensee safety evaluations and 10 applicability determinations and screenings (combined). The inspectors completed a review of six licensee safety evaluations and a combination of 18 applicability determinations and screenings.

The inspectors completed 24 inspection samples.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

1. Partial System Walkdowns

a. Inspection Scope

The inspectors: (1) walked down portions of the four risk important systems listed below and reviewed plant procedures and documents to verify that critical portions of the selected systems were correctly aligned; and (2) compared deficiencies identified during the walk down to the licensee's USAR and CAP to ensure problems were being identified and corrected. Documents reviewed by the inspectors are listed in the attachment.

- May 31, 2007, Shutdown Cooling Line-up and Fuel Pool Assist mode of shutdown cooling

- May 31, 2007, Primary Containment Integrity during operations with the possibility of draining the reactor pressure vessel

- June 11, 2007, Division II Standby Service Water

- June 11, 2007, Division II Standby Diesel Generator

The inspectors completed four inspection samples.

b. Findings

No findings of significance were identified.

2. Complete System Walkdown

a. Inspection Scope

The inspectors: (1) reviewed plant procedures, drawings, the USAR, and TS to determine the correct alignment of the on-site Division I 4.16 Kv electrical distribution system; (2) reviewed outstanding design issues, operator workarounds, and USAR documents to determine if open issues affected the functionality of the on-site Division I 4.16 Kv electrical distribution system; and (3) verified that the licensee was identifying and resolving equipment alignment problems. Documents reviewed by the inspectors included:

- SOP-0046, “4.16 KV System,” Revision 031
- USAR Section 8.1.4, “Onsite AC Systems”
- TS Section 3.8.1, “AC Sources - Operating”

The inspectors completed one inspection sample.

b. <u>Findings</u>

No findings of significance were identified.

1R05 <u>Fire Protection</u>

a. <u>Inspection Scope</u>

<u>Quarterly Inspection</u>

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 and hot work activities 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; and (6) reviewed the CAP to determine if the licensee identified and corrected fire protection problems.

- April 16, 2007, Turbine Building, 67-foot level, reactor feed pump area, Fire Area TB-67

- May 17, 2007, Standby Switchgear 1B Room, 98-foot level, Fire Area C-14

- May 17, 2007, Safety Related Cable Chase II, 98-foot level, Fire Area C-2

- June 11, 2007, Safety Related Water Chiller Equipment 1B Room, 98-foot level, Fire Area C-13E

- June 12, 2007, Safety Related Water Chiller Equipment 1A Room, 98-foot level, Fire Area C-13W

- June 13, 2007, Standby Switchgear 1A Room, 98-foot level, Fire Area C-15

Documents reviewed by the inspectors included:

- Pre-Fire Plan/Strategy Book
- USAR Section 9A.2, “Fire Hazards Analysis”
- River Bend Station post-fire 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.

1R11 Licensed Operator Requalification Program

a. Inspection Scope

On June 20, 2007, the inspectors observed one Annual 2007 Dynamic Simulator Examination of senior reactor operators and reactor operators to identify deficiencies and discrepancies in the examination process, to assess operator performance, and to assess the evaluator's critique. The training scenario involved reactor feed Pump A minimum flow failing open, followed by failure of both seals on reactor recirculation Pump A, and a design basis accident loss of coolant accident (LOCA). Documents reviewed by the inspectors included:

- Simulator Examination Scenario RSMS-OPS-0826, "Feedwater Malfunction/Recirc Pump Seal Failure/DBA LOCA With Failure to Isolate," Revision 1

The inspectors completed one inspection sample.

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 two assessment activities listed below to verify: (1) performance of 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 04B, 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 recognizes, and/or enters as applicable, the appropriate licensee-established risk category according to the risk assessment results and licensee procedures; and (4) the licensee identified and corrected problems related to maintenance risk assessments.

- Week of June 18, 2007, Division II work week
- Week of June 25, 2007, Non-Divisional work week

Documents reviewed by the inspectors included:

- Computerized equipment out-of-service risk monitor
- Daily plant status sheets
- Computerized LCO reports
- Computerized RBS daily schedule by system

The inspectors completed two inspection samples.

.2 Emergent Work Control

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.

- April 1-4, 2007, daily power reductions due to main condenser inefficiency

- Weeks of April 9 and 16, 2007, power reduction to 70 percent and then 65 percent for on-line condenser cleaning

- April 24, 2007, turbine taken off line and the reactor remained critical at 15 percent power with pressure control via the turbine bypass valves for feedwater relief valve maintenance

- April 25, 2007, power ascension following forced Outage 07-01

- May 7, 2007, power ascension following forced Outage 07-02

- May 10 and 17, 2007, residual heat removal shutdown cooling outboard isolation valve logic relay replacement

- June 1, 2007, operations with potential to drain the reactor vessel during repair of recirculation Loop A discharge gate valve

- June 2, 2007, operations with potential to drain the reactor vessel during repair of recirculation Loop B discharge gate valve

- June 10, 2007, power ascension following forced outage 07-03

Documents reviewed by the inspectors included:

- Computerized equipment out-of-service risk monitor
- Daily plant status sheets
- Computerized LCO reports
- Computerized RBS daily schedule by system
- OSP-0034, "Control of Obstructions for Primary Containment/Fuel Building Operability," Revision 3
- OSP-0033, "Operations with a Potential to Drain the Reactor Vessel/Cavity," Revision 6
- OSP-0037, "Shutdown Operations Protection Plan (SOPP)," Revision 16

The inspectors completed nine inspection samples.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors: (1) reviewed plant 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 has identified and implemented appropriate corrective actions associated with degraded components. The licensee operability evaluations were documented in the following CRs:

- CR-RBS-2007-02104, Division 2 diesel generator fuel day level transmitter sensor tubing, reviewed on May 25, 2007
- CR-RBS-2007-02169, Drywell unit Cooler F plugged drain, reviewed on May 29, 2007
- CR-RBS-2007-02180, Low pressure coolant injection Pump C discharge flow transmitter fount out of tolerance low during surveillance test, reviewed on May 30, 2007

- CR-RBS-2007-02230, Division I standby diesel generator excitation cabinet temperature below limits for rounds, reviewed on June 5, 2007

- CR-RBS-2007-02364, Two broken drywell head finger pins (#47 & #48), reviewed on June 15, 2007

Documents reviewed by the inspectors included: Nuclear Management Manual Procedure EN-OP-104, “Operability Determinations,” Revision 2.

The inspectors completed five inspection samples.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

Biennial Review

The inspection procedure requires inspection of a minimum sample size of five permanent plant modifications.

The inspectors reviewed nine permanent plant modification packages and associated documentation, such as implementation reviews, safety evaluation applicability determinations, and screenings to verify that they were performed in accordance with regulatory requirements and plant procedures. The inspectors also reviewed the procedures governing plant modifications to evaluate the effectiveness of the program for implementing modifications to risk-significant SSCs, such that these changes did not adversely affect the design and licensing basis of the facility. Procedures and permanent plant modifications reviewed are listed in the Attachment to this report. Further, the inspectors interviewed the cognizant design and system engineers for the identified modifications as to their understanding of the modification packages and process.

The inspectors evaluated the effectiveness of the licensee’s corrective action process to identify and correct problems concerning the performance of permanent plant modifications by reviewing a sample of related CRs. The reviewed CRs are identified in the Attachment to this report.

The inspectors completed nine inspection samples.

Annual Review

The inspectors reviewed key affected parameters associated with materials/replacement components, timing, control signals, equipment protection from hazards, operations,

flowpaths, ventilation boundary, structural, licensing basis, and failure modes for the modification listed below. The inspectors verified that: (1) modification preparation, staging, and implementation did not impair emergency/abnormal operating procedure actions, key safety functions, or operator response to loss of key safety functions; (2) postmodification testing maintained the plant in a safe configuration during testing by verifying that unintended system interactions would not occur, SSC performance characteristics still met the design basis, the appropriateness of modification design assumptions, and the modification test acceptance criteria had been met; and (3) the licensee had identified and implemented appropriate corrective actions associated with permanent plant modifications.

- Week of May 21, 2007, ER-RB-2004-0131-001, 480 Vac breaker replacement for control building air conditioning Chiller HVK-CHL1C

Documents reviewed by the inspectors included:

- WO 00102689, Replace Breaker EJS-SWG1A-ACB003
- CR-RBS-2007-01189, HVK-CHL1C ghost light indications in “test” position

The inspectors completed one inspection sample.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing

a. Inspection Scope

The inspectors selected the five postmaintenance test activities listed below of risk significant systems or components. For each item, 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 ensure 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. The postmaintenance testing was part of the following work orders (WO):

- WO00085268, Adjust Division II emergency diesel generator governor, reviewed on May 29, 2007

- WO00078503, Replace containment unit cooler supply Breaker EJS-ACB-036, reviewed on April 19, 2007

- WO50690110 03, disassemble, clean, inspect check valves & orifices for penetration valve leakage control Compressor B, LSV-C3B, reviewed on June 4, 2007.
- WO100094-01, replaced RCIC steam line flow high transmitter, E31-PDTN083B, reviewed on June 19, 2007
- WO00091258 01, replace reactor trip Logic A relay, C71-K71, reviewed on June 23, 2007

The inspectors completed five inspection samples.

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities

a. Inspection Scope

During three forced outages: April 24-25, 2007, to repair a leaking feedwater system relief valve; May 4-7, 2007, in response to main transformer cooling system failure; and May 22 through June 10, 2007, in response to recirculation system Loop A discharge gate valve stem/disk separation; the inspectors reviewed the following risk significant outage activities to verify defense in depth commensurate with the outage risk control plan, compliance with the TS, and adherence to commitments in response to Generic Letter 88-17, "Loss of Decay Heat Removal": (1) the risk control plan; (2) tagging/clearance activities; (3) reactor coolant system (RCS) instrumentation; (4) electrical power; (5) decay heat removal; (6) inventory control; (7) containment closure; (8) operations with the potential to drain the reactor pressure vessel; (9) heatup and cooldown activities; (10) restart activities; and (11) licensee identification and implementation of appropriate corrective actions associated with outage activities. The inspectors' drywell inspections included observations of supports, braces, and snubbers for evidence of excessive stress, water hammer, or aging. Specific outage activities observed and reviewed included:

- Installation of recirculation loop level indication systems
- Plugging of jet pump nozzles
- Draining of recirculation loops
- Restoration of recirculation loops
- Plant restart, heatup, and connection to the grid

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed three inspection samples.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors reviewed the USAR, procedure requirements, and TS to ensure that the seven Surveillance Test Procedures (STP) listed below demonstrated that the SSC's 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 data; (13) engineering evaluations, root causes, and bases for returning tested SSC's not meeting the test acceptance criteria to operable status; (14) reference setting data; and (15) annunciators and alarms setpoints. The inspectors also verified that the licensee identified and implemented any needed corrective actions associated with the surveillance testing.

- STP-500-4203, "RPS-Scram Discharge Volume Water Level-High Channel Calibration Test and Logic System Functional Test (C11-LTN012B; C11-N0601B," Revision 12, performed on May 15, 2007

- STP-309-0203, "Division III Diesel Generator Operability Test," Revision 302, performed on May 16, 2007

- STP-302-1201, "ENS-SWG1A Degraded Voltage Channel Functional Test," Revision 11, performed on June 21, 2007

- WO00112179 01, "Change the Range of Transmitters B21-LTN027 and C33-LTN017 for Mode 4 and 5 Operation," Revision 8/13/2006, performed on June 21, 2007

- STP-309-0201, "Division I Diesel Generator Operability Test," Revision 031, performed on June 23, 2007

- STP-505-4517, "RPS/Control Rod Block-APRM Setdown Channel Functional Test (C51-K605A Through C51-K605H)," Revision 04, performed on June 26, 2007

- STP-204-6304, "Div II RHR Quarterly valve Operability Test," Revision 018, performed on December 1, 2006 (Inservice test surveillance)

The inspectors completed seven inspection samples.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the USAR, plant drawings, procedure requirements, and TS to ensure that one temporary 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 post installation test results were satisfactory and that the impact of the temporary modification on permanently installed SSC's was supported by the test; and (4) verified that appropriate safety evaluations were completed. The inspectors verified that the licensee identified and implemented any needed corrective actions associated with the temporary modification.

- Temporary thermocouples to monitor core exit temperatures, reviewed May 13, 2007

Documents reviewed by the inspectors included:

- GMP-0102, "Reactor Vessel Disassembly," Revision 16
- EQIS M-553, "Analysis of teflon insulated thermocouple wire for temporary reactor coolant temperature monitoring instrumentation"

The inspectors completed one inspection sample.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP2 Alert and Notification System Testing

a. Inspection Scope

The inspector discussed with licensee staff the status of offsite siren and tone alert radio systems to determine the adequacy of licensee methods for testing the alert and notification system in accordance with 10 CFR Part 50, Appendix E. The licensee's alert and notification system testing program was compared with criteria in NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1, Federal Emergency Management Agency (FEMA) Report REP-10, "Guide for the Evaluation of Alert and Notification Systems for Nuclear Power Plants," and the licensee's current

FEMA-approved alert and notification system design report, "River Bend Station Prompt Notification System Design Report," Revision 1, December 2001. The inspector also reviewed procedures EPP-2-401, “Inadvertent Siren Sounding,” Revision 7, EPP-2-502, “Emergency Communications Equipment Testing,” Revision 22, and EPP-2-701, "Prompt Notification System Maintenance and Testing," Revisions 18 and 19. The inspector completed one inspection sample.

b. Findings

No findings of significance were identified.

1EP3 Emergency Response Organization Augmentation

a. Inspection Scope

The inspector discussed with licensee staff the status of primary and backup systems for augmenting the on-shift emergency response staff to determine the adequacy of licensee methods for staffing emergency response facilities. The inspector reviewed the references listed in the Attachment to this report related to the emergency response organization augmentation system to evaluate the licensee’s ability to staff the emergency response facilities in accordance with the licensee emergency plan and the requirements of 10 CFR Part 50, Appendix E.

The inspector completed one inspection sample.

b. Findings

No findings of significance were identified.

1EP4 Emergency Action Level and Emergency Plan Changes

a. Inspection Scope

The inspector performed an in-office review of Revisions 15 and 16 to Emergency Plan implementing Procedure EIP-2-001, “Classification of Emergencies,” submitted April 19, 2007. These revisions implemented a scheme of emergency action levels based on Nuclear Engineering Institute (NEI) 99-01, "Methodology for Development of Emergency Action Levels," Revision 4, as approved by the NRC by letter dated October 25, 2005.

The revisions were compared to the NRC Safety Analysis Report dated October 25, 2005, the criteria of NRC Bulletin 2005-002, "Emergency Preparedness and Response Actions for Security Based Events," the criteria of NEI 99-01, Revision 4, and to the standards in 10 CFR 50.47(b) to determine if the revisions were adequately conducted following the requirements of 10 CFR 50.54(q). This review was not documented in a Safety Evaluation Report and did not constitute approval of licensee changes; therefore, these revisions are subject to future inspection.

The inspector completed two inspection samples.

b. Findings

No findings of significance were identified.

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies

a. Inspection Scope

The inspector reviewed the licensee's CAP requirements in Procedures EN-LI-102, "Corrective Action Process," Revision 2, and EN-LI-119, "Apparent Cause Evaluation Process," Revision 3. The inspector reviewed summaries of 180 CRs assigned to the emergency preparedness department between November 2005 and May 2007, and selected 13 for detailed review against the program requirements. The inspector evaluated the response to the corrective action requests to determine the licensee's ability to identify, evaluate, and correct problems in accordance with the licensee program requirements and 10 CFR 50.47(b)(14) and 10 CFR 50 Appendix E. The inspector reviewed the licensee's audit program requirements in Procedure EN-QV-109, "Audit Process," Revision 9, the 2006 quality assurance audit, quality assurance surveillances conducted in 2006 and 2007, and licensee self-assessments of emergency preparedness. The inspector also reviewed other documents listed in the attachment to this report.

The inspector completed one inspection sample.

b. Findings

No findings of significance were identified.

1EP6 Exercise Evaluation

a. Inspection Scope

For the exercise below, the inspectors: (1) observed the evolution to identify any weaknesses and deficiencies in classification, notification, and Protective Action Requirements development activities; and (2) reviewed the identified weaknesses and deficiencies against licensee-identified findings to determine whether the licensee is properly identifying deficiencies.

- June 6, 2007 Force-On-Force Drill, Day 2

Documents reviewed by the inspectors included:

- AOP-0054, "Security Events," Revision 007
- Technical support center lead controller notes
- Completed notification forms
- ENS-NS-215, "Conduct of Security Force Exercises and Drills," Revision 1

- EIP-2-001, “Classifications of Emergencies,” Revision 14
- EIP-2-006, “Notifications,” Revision 32

The inspectors completed one inspection sample.

8. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

a. Inspection Scope

Cornerstone: Emergency Preparedness

The inspector reviewed licensee evaluations for the three emergency preparedness cornerstone PIs of Drill and Exercise Performance, Emergency Response Organization Participation, and Alert and Notification System Reliability, for the period July 1, 2006, through March 31, 2007. The definitions and guidance of NEI 99-02, “Regulatory Assessment Indicator Guideline,” Revisions 2 through 4, and the licensee PI Procedure EN-LI-114, “Performance Indicator Process,” Revision 2, were used to verify the accuracy of the licensee’s evaluations for each PI reported during the assessment period.

The inspector reviewed a 100 percent sample of drill and exercise scenarios and licensed operator simulator training sessions, notification forms, and attendance and critique records associated with training sessions, drills, and exercises conducted during the verification period. The inspector reviewed drill participation records for key emergency responders. The inspector reviewed alert and notification system testing procedures, maintenance records, and a 100 percent sample of siren test records. The inspector also reviewed other documents listed in the Attachment to this report.

The inspector completed one inspection sample.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

a. Inspection Scope

.1 Routine Review of Identification and Resolution of Problems

The inspectors performed a daily screening of items entered into the licensee's CAP. This assessment was accomplished by reviewing CRs and WOs and attending corrective action review and work control meetings. The inspectors: (1) verified that equipment, human performance, and program issues were being identified by the licensee at an appropriate threshold and that the issues were entered into the CAP; (2) verified that corrective actions were commensurate with the significance of the issue; and (3) identified conditions that might warrant additional follow-up through other baseline inspection procedures.

The inspectors completed one inspection sample.

.2 Semiannual Trend Review

The inspectors completed a semiannual trend review of repetitive or closely related issues that were documented in corrective maintenance documents, metrics, and trend reports to identify trends that might indicate the existence of more safety significant issues. The inspectors review consisted of the six month period of July - December 2006. When warranted, some of the samples expanded beyond those dates to fully assess the issue. The inspectors also reviewed CAP items associated with work control. The inspectors compared and contrasted their results with the results contained in the licensee's quarterly trend reports. Corrective actions associated with a sample of the issues identified in the licensee's trend report were reviewed for adequacy. Documents reviewed by the inspectors included:

- EN-MA-123, “Identification and Trending of Rework,” Revision 0
- EN-WM-102, “Work Implementation and Closeout,” Revision 0
- ADM-0080, “Post-Maintenance Testing,” Revision 4A
- CAP Search results on terms “KW-Vendor” and “KW- Planning,” executed on March 23, 2007

The inspectors completed one inspection sample.

.3 Annual Sample Review

The inspector selected 13 CRs for detailed review. The reports were reviewed to ensure that the full extent of the issues were identified, an appropriate evaluation was performed, and appropriate corrective actions were specified and prioritized. The inspectors evaluated the CRs against the requirements of Procedure EN-LI-102, “Corrective Action Process."

The inspector completed one inspection sample.

b. Findings and Observations

There were no findings identified associated with the review of licensee corrective actions in that the full extent of issues were identified and the licensee identified appropriate corrective actions; however, the inspector identified a developing trend in that 7 of 13 CRs were closed prior to completing the assigned corrective actions. The inspector determined that the corrective actions had been appropriately completed in all cases.

4OA3 Event Followup

.1 Missing scram discharge instrument volume vent plug

Introduction. A Green self-revealing NCV of TS 5.4.1.a was identified involving the failure to follow a surveillance procedure for scram discharge instrument volume water level channel calibration. Specifically, a vent plug was not replaced following surveillance testing. As a result, on May 4, 2007, following a reactor scram, reactor coolant sprayed out of the scram discharge instrument volume (SDIV) and contaminated some accessible portions of the containment building causing three inadvertent personnel contaminations.

Description. On May 4, the reactor was manually scrammed when a main transformer cooling system failed. Following the scram, licensee non-destructive examination inspectors performed post-scram inspections of the hydraulic control unit piping. When they completed the inspection while processing through personnel contamination monitors they were found to be contaminated. The licensee investigation found that a vent plug was not installed on a vent connection on the west side SDIV as required resulting in leakage of water following the scram.

The missing vent plug resulted in an opening in the SDIV. When the reactor was scrammed, the SDIV filled with reactor water at full reactor pressure causing the SDIV to become part of the RCS pressure boundary. Reactor coolant sprayed through the vent opening in the SDIV until the scram was reset. The operators reset the scram which isolated the SDIV from the RCS and opened vent and drain valves, which returned the SDIV to atmospheric conditions.

The previous documented manipulation of that vent was on February 9, 2007, during the performance of STP-500-4203, “RPS-Scram Discharge Volume Water Level - High Channel Calibration Test and Logic System Functional Test (C11-LTN012B; C11-N601B).” The inspectors determined most plausible cause of the vent plug not being installed was that during the performance of STP-500-4203 the technicians failed to reinstall the plug as required by the surveillance procedure. Specifically, procedure STP-500-4203, Section 7.3, “Restoration,” Revision 12, states, “Reinstall vent plugs.”

Analysis. The finding was more than minor because it was associated with the initiating event cornerstone attribute of equipment performance and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during power operations. A Phase 2 estimation was required, as determined by the Manual Chapter 0609, Appendix A, Phase 1 Worksheet, "SDP Phase 1 Screening Worksheet for Initiating Events, Mitigation Systems, and Barriers Cornerstones," because the associated performance deficiency resulted in a reactor coolant leak greater than the Technical Specification limit for identified reactor coolant system leakage. Using the plant-specific Phase 2 risk-informed notebook, this violation was determined to have very low safety significance because the violation only increased the likelihood of a small-break loss of coolant accident by a very small amount and mitigation capability was unaffected. The cause of the finding was related to the human performance crosscutting component of work practices because neither self nor peer checking actions identified the failure to replace the vent plug (H.4(a)).

Enforcement. TS 5.4.1.a requires that written procedures be established, implemented, and maintained covering the activities specified in Regulatory Guide 1.33, Revision 2, Appendix A, dated February 1978. Regulatory Guide 1.33, Appendix A, section 9, requires procedures for performing maintenance activities. Contrary to this, maintenance technicians failed to implement STP-500-4203, "RPS-Scram Discharge Volume Water Level - High Channel Calibration Test and Logic System Functional Test (C11-LTN012B; C11-N601B)," that required reinstallation of the scram discharge instrument volume vent plugs. Because the finding is of very low safety significance and has been entered into the licensee's CAP as Condition Report CR-RBS-2007-01809, this violation is being treated as an NCV consistent with Section VI.A of the Enforcement Policy:
NCV 05000458/2007003-01, "Missing vent plug caused breach of scram discharge instrument volume."

.2 Reactor Recirculation Loop 'A' Flow Anomaly

a. Inspection Scope

The inspectors assessed the circumstances related to the separation of reactor recirculation system Loop A discharge gate valve disk from its stem. The inspectors reviewed the actions taken by the licensee to control recirculation system loop flow mismatch, the overall risk assessment and management during the subsequent forced outage, the adequacy of contingencies implemented for high risk plant configurations, and the scope and conduct of the maintenance on the recirculation loop discharge gate valves. In addition, the inspectors developed a detailed sequence of events, reviewed the licensee's extent of condition review, and evaluated pertinent industry operating experience and potential precursors to the event, including the effectiveness of licensee actions taken in response to applicable operating experience.

b. Findings and Observations

Introduction. A Green self-revealing finding was identified involving the failure to implement vendor recommendations concerning vibration induced wear on the recirculation system suction and discharge gate valves. This deficiency resulted in the failure to identify and implement timely actions prior to disk to stem failure of recirculation Pump A discharge gate valve that occurred on May 21, 2007.

Description. On May 21, 2007, reactor recirculation loop Train A flow unexpectedly lowered by 4,000 pounds mass per hour, without any change in the reactor recirculation flow control valve Train A position or operator action. There was a corresponding drop in reactor power from 100 percent to 96 percent. The licensee balanced flows between the two reactor recirculation loops. The licensee held reactor power at 90 percent while they evaluated the flow anomaly. The licensee concluded that the most likely cause was stem/disk separation of recirculation Loop A discharge gate valve. On May 22, 2007, the reactor was shutdown to further investigate the flow anomaly.

The licensee conducted a problem analysis, prior to the shutdown, and concluded that the disk had likely separated from the stem of recirculation discharge gate valve (RDGV) Loop A, B33-MOVF067A, and that disk partially blocked flow through recirculation Loop A,

producing the indications received on May 21, 2007. Upon disassembly of B33-MOVF067A, the licensee confirmed disk to stem separation of the valve. The failure mechanism was due to turbulent flow induced vibrations resulting in wear of the valve stem, upper wedge threads, and wedge pin. The licensee replaced the disk and stem assembly in both B33-MOVF067A and B.

The licensee's root cause investigation team determined that several opportunities to prevent the unexpected failure were missed. It had been known as early as 1988 that there was the possibility of RDGV damage due to turbulent flow. An important missed opportunity occurred in 1998 when the licensee failed to implement vendor recommendations. In 1998, the licensee stated that system engineers would implement vendor recommendations to monitor the performance of the recirculation loop suction and discharge gate valves by periodically checking vibration, flow, and checking the loose parts monitoring system for abnormal noise patterns. The licensee determined these recommendations were not effectively implemented resulting in the failure to identify the degrading condition prior to the failure experienced on May 21, 2007.

<u>Analysis</u>. The performance deficiency associated with this finding involved the licensee's failure to implement vendor recommendations to monitor the performance of the recirculation loop suction and discharge gate valves. The finding was more than minor because it was associated with the initiating events cornerstone attribute of equipment performance and affected the associated cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown and power operations. Using the Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the finding was determined to have very low safety significance because the finding did not contribute to the likelihood that mitigating equipment or functions would not be available following a reactor trip. This issue was entered into the licensee's CAP as condition Report CR-RBS-2007-02113

<u>Enforcement</u>. No violation of NRC requirements occurred. FIN 05000458/2007003-02, "Failure to Implement Vendor Recommendations."

.3 <u>Load Line Analysis Limit Exceeded</u>

<u>Introduction</u>. A Green self-revealing NCV of TS 5.4.1.a was identified involving the failure to follow procedure resulting in an average power range monitor rod block.

<u>Description</u>. On April 26, 2007, during power ascension the reactor engineer noted that reactor power, calculated by the heat balance, did not rise as fast as predicted. The reactor engineer failed to determine that the feedwater flow and temperature input data to the heat balance was not updating; therefore, the reactor power level calculated by the heat balance program was inaccurate.

The reactor engineer was monitoring core parameters during control rod withdrawal, and noted that reactor power indicated 75 percent instead of the expected 77-78 percent. He also noted effective multiplication factor (Keff) and the rod-line was indicated differently than expected; however, the thermal limits and margin to preconditioning were consistent with the indicated power level. He discussed these observations with another reactor engineer

and concluded the Keff change was curious but not an indication of further problems that needed to be investigated. The control room supervisor and reactor engineer discussed the difference between expected conditions and indicated conditions. Control rod pull sheets were discussed and approved for implementation by the reactor engineer. The control room supervisor ordered rod withdrawal to continue power ascension. During the subsequent withdrawal of control rods an average power range monitor rod block stopped the rod withdrawal. Following this automatic safety feature stopping the withdrawal of control rods the reactor engineer identified that the heat balance was inaccurate and requested that the control room supervisor halt the power ascension.

The inspectors noted that the licensee evaluated the safety significance in consultation with the fuel vendor. They reached the conclusion that no thermal limits or preconditioning limits were exceeded therefore no potential for fuel damage occurred.

The inspectors reviewed the licensee's root cause analysis report, personnel statements, and discussed the issue with reactor engineers and station management. The inspectors reviewed reactor engineering Instruction 08, "Reactor Engineering Standards and Expectations," Revision 09, which provided guidelines and department expectations for reactor engineering personnel such that activities are carried out in an effective and consistent manner. The inspectors noted that the licensee's investigation team determined that the root cause of the load line excursion was that the reactor engineering Instruction 08, "Reactor Engineering Standards and Expectations," Revision 09., was not followed correctly.

Reactor engineering Instruction 08, "Reactor Engineering Standards and Expectations," Revision 09, requires that when unexpected conditions arise (e.g., significantly higher/lower rodlines than planned, unplanned or unanticipated core parameter changes, etc.) put the reactor in a safe condition and evaluate the condition so that it is fully understood before proceeding. Contrary to this, when the anomaly presented itself and the actual power and Keff deviated from the predicted power and Keff, the reactor engineer did not investigate thoroughly enough to fully understand the condition. He proceeded forward with further rod withdrawals to achieve desired load line and did not recognize that he was in a position of uncertainty. The result was an average power range monitor rod block and the load line analysis limit was exceeded by 2.4 percent for 15 minutes at 80 percent power.

The inspectors concluded that the combined failures of self checking and peer checking demonstrated a lack of engineering rigor and resulted in the failure to conclude the heat balance was inaccurate in time to prevent the load line excursion event.

Analysis. The performance deficiency associated with this finding involved the licensee's failure to comply with the requirements of reactor engineering Instruction 08, "Reactor Engineering Standards and Expectations," Revision 09. The finding was more than minor because it was associated with the barrier integrity cornerstone attribute of configuration control and it affected the cornerstone objective to provide reasonable assurance that physical design barriers, such as fuel cladding, protect the public from radio-nuclide releases caused by accidents or events. Using Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the finding was determined to have a very low safety significance because it did not have the potential to affect the integrity of the reactor coolant system barrier.

The cause of this finding is related to the human performance crosscutting component of work practices because neither self nor peer checking actions prevented the automatic rod withdrawal block (H.4(a)).

Enforcement. TS 5.4.1.a requires that written procedures be established, implemented, and maintained covering the activities specified in Regulatory Guide 1.33, Revision 2, Appendix A, dated February 1978. Regulatory Guide 1.33, Appendix A, section 2.b, requires that procedures for power operation and process monitoring be written and implemented. Contrary to this requirement, the requirements of reactor engineering Instruction 08, “Reactor Engineering Standards and Expectations,” Revision 09, were not implemented. Specifically, on April 26, 2007, following unexpected conditions during power ascension the reactor was not placed in a safe condition and the condition evaluated and understood before proceeding. This deficiency resulted in exceeding the load line analysis limit and actuation of a rod block protection feature during control rod withdrawal. Because the finding is of very low safety significance and has been entered into the licensee’s CAP as Condition Report CR-RBS-2007-01691, this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000458/2007003-03, “Failure to Follow Instructions Resulted in Exceeding Load Line Analysis Limit.”

.4 Inadequate Maintenance Instructions

Introduction. The inspectors identified a Green finding involving the failure to provide adequate instructions in a WO. Specifically, the WO failed to provide instructions to safely remove the manual actuator from feedwater system isolation valve FWS-V1 while it was subjected to the high energy conditions of the feedwater system normal operating pressure and temperature.

Description. On June 10, 2007, while attempting to put a third feedwater regulating valve in service during power ascension following forced outage 07-03, the operators were unable to open manual isolation Valve FWS-V1. The licensee prepared WO 00113850-06 for mechanical maintenance technicians to open FWS-V1 while it was subjected to feedwater system normal operating pressure and temperature. WO 00113850-06 provided instructions to: (1) remove the manual actuator; (2) install a stem lifting tool; (3) open FWS-V1; (4) rebuild the manual actuator; (5) remove the lifting tool and; (6) reinstall the manual actuator. The inspectors noted the WO failed to provide the instructions necessary to restrain FWS-V1 stem and prevent unexpected valve stem movement or ejection.

On June 10, 2007, while removing the manual actuator from FWS-V1, the mechanics observed that as they loosened the eight actuator mounting bolts the stem was moving out of the valve body on its own. FWS-V1 stem had moved out approximately one-quarter inch. The mechanics stopped and asked for and received permission to open the valve at that time because they no longer needed to remove the damaged manual actuator and install the stem lifting tool prior to opening FWS-V1.

The mechanics removed four of the eight actuator mounting bolts and replaced them with long threaded rods. Next, the mechanics began manipulating the remaining four actuator mounting bolts and the four long threaded rods to allow the valve stem to travel out of the valve body. The inspectors noted that these steps were not in the original WO instructions.

By the time the valve stem had traveled three inches out of the valve body, the mechanics received instructions to stop the job so that the work activity could be re-evaluated.

The WO was revised to install a valve stem clamp on FWS-V1, remove and repair the damaged manual actuator, reinstall the actuator, and open FWS-V1 manually with that actuator. These activities were successfully implemented.

The inspectors interviewed the mechanics and found that the prejob brief did not mention the fact that FWS-V1 was at full system pressure and temperature of approximately 1,100 psig and 350°F. The prejob brief also did not cover operating experience associated with valve stem ejection accidents. The mechanics stated that they were not aware those conditions existed until the inspectors informed them of the conditions during the interview. The inspectors concluded that even though the mechanics did not recognize that a valve stem ejection accident was possible on this job, their actions prevented what could have been a valve stem ejection accident.

Analysis. The performance deficiency associated with this finding involved inadequate maintenance instructions for opening feedwater regulating Valve A isolation valve, FWS-V1. The finding was more than minor because it could become a more significant safety concern if left uncorrected. Using the Manual Chapter 0609, “Significance Determination Process,” Phase 1 Worksheet, the finding was determined to have very low safety significance because the deficiency did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available. This issue was entered into the licensee's CAP as condition Report CR-RBS-2007-02576.

The cause of this finding was related to the human performance crosscutting component of resources in that the licensee did not ensure a complete and accurate work package was available prior to the start of the job (H.2(c)).

Enforcement. No violation of NRC requirements occurred. FIN 05000458/2007003-04, “Inadequate Work Instructions.”

4OA4 Crosscutting Aspects of Findings

Section 4OA3 describes one finding and two NCVs related to crosscutting area of human performance:

- The finding was associated with the human performance crosscutting component of resources in that the licensee did not ensure a complete and accurate work package was available prior to the start of the job (H.2(c))

- One NCV was associated with the human performance crosscutting component of work practices in that neither self or peer checking actions prevented an automatic rod withdrawal block (H.4(a))

- One NCV was associated with the human performance crosscutting component of work practices in that neither self or peer checking actions identified a failure to reinstall a vent plug (H.4(a))

4OA6 Management Meetings

On April 20, 2007, the inspectors presented the safety evaluation and permanent plant modifications inspection results to Mr. J. Venable, Senior Vice President, and other members of licensee management who acknowledged those results. No proprietary information was included in this report.

On May 18, 2007, the inspector presented the emergency preparedness inspection results to Mr. E. Olson, General Manager, Plant Operations, and other members of licensee management who acknowledged the findings. The inspector confirmed that proprietary information was not retained following the inspection.

On July 9, 2007, the inspectors presented the integrated baseline inspection results to Mr. J. Venable, Senior Vice President, and other members of licensee management who acknowledged the findings. The inspector confirmed that proprietary information was not retained following the inspection.

4OA7 Licensee-Identified Violations

The following violation of very low safety significance (Green) was identified by the licensee and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Policy for being dispositioned as an NCVs.

- 10 CFR Part 50.47(b)(2) states, ". . . adequate staffing to provide initial facility accident response in key functional areas is maintained at all times. . . ."
  10 CFR Part 50.47(b)(15) states, "Radiological emergency response training is provided to those who may be called on to assist in an emergency." Contrary to the above, one chemistry technician whose emergency response organization qualifications had expired stood 11 watches as the required on-shift dose assessor between January 15 and August 5, 2006. Although the licensee's on-shift staffing process allowed more than two shifts during a 30-day period to go below emergency plan requirements, this performance deficiency has been evaluated as being of low safety significance (Green) because the finding was not a functional failure of planning standard 50.47(b)(2), in that, the technician was present and may have been capable of performing their required emergency plan function, and other trained licensee personnel not usually assigned dose assessment responsibilities were present and could have assisted if necessary. This issue was identified in the licensee's CAP as CRs 2006-03264 and 2007-02023.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

L. Ballard, Manager, Quality Programs
C. Bush, Manager, Outage
M. Chase, Manager, Training and Development
J. Clark, Assistant Operations Manager - Training
C. Forpahl, Manager, Corrective Action Program
B. Heath, Acting Superintendent, Chemistry
K. Higginbotham, Assistant Operations Manager - Shift
B. Houston, Manager, Radiation Protection
A. James, Superintendent, Plant Security
N. Johnson, Manager, Engineering Programs & Components
J. Laque, Manager, Plant Maintenance
J. Leavines, Manager, Emergency Planning
D. Lorfing, Manager, Licensing
J. Maher, Superintendent, Reactor Engineering
W. Mashburn, Manager, Design Engineering
B. Matherne, Manager, Planning and Scheduling/Outage
J. Miller, Manager, Operations
E. Olson, General Manager - Plant Operations
J. Roberts, Director, Nuclear Safety Assurance
P. Russell, Manager, System Engineering
J. Venable, Site Vice President
D. Wiles, Director, Engineering

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000458/2007003-01	NCV	Failure to Install Scram Discharge Instrument Volume Vent Plug
05000458/2007003-02	FIN	Failure to Implement Vendor Recommendations
05000458/2007003-03	NCV	Failure to Follow Instructions Resulted in Exceeding Load Line Analysis Limit
05000458/2007003-04	FIN	Inadequate Work Instructions

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 1R02: Evaluation of Changes, Tests, or Experiments

10 CFR 50.59 Evaluations

Number	Title	Revision
EC-0000000252	Continuous Backfill/Flush on E31-PDTN084A/B Transmitter	0
ER-RB-2006-0250	Add time delay relay to standby service water initiation logic	0
ER-RB-2006-0111	Removal of DER-PS44A(B) input to control room annunciator	0
ER-RB-2005-0350	Delete Condensate Filter Thermal Relief Valves	0
ER-RB-2004-0131	Standby 480V Load Centers, Division I & II	0
ER-RB-2004-0210	Fabrication and Installation of vented steel floor plugs for reactor water cleanup system demineralizer cubicles	0

10 CFR 50.59 Screenings

Number	Title	Revision
ER-RB-2005-0088	Replace STX-XS2B	0
ER-RB-2004-0367	Replace SCA-PNL2D3, Breakers 7 through 15 with Ground Fault Interrupting Breakers	0
ER-RB-2004-0466	Provide Welded Patch Details for DTM-014-040-4	0
ER-RB-2005-0269	Replace Aged Chemical Storage Tanks	0
ER-RB-2003-0237	Evaluate the removal of by-pass valve in gland steam exhaust condenser	0
ER-RB-2002-0509	Isolation valves for generator core monitor	0
ER-RB-2002-0342	Minor mod required for WTH	0
ER-RB-2001-0529	Offgas condenser level control panel rearrangement	0
ER-RB-2002-0509	Isolation valves for generator core monitor	0
AOP-0016	Loss of Standby Service Water	14

Applicability Determinations

Number	Title	Revision
ER-RB-2005-0345	Replacement valves for N64-VF004A&B	0
ER-RB-2005-0346	Replacement valve for FWS-V29	0
ER-RB-2004-0307	Change elastomer materials for Fisher AOV O-Rings to VITON	0
ER-RB-2003-0547	Replace existing EHC isolation valves	0
SOP-0079R16PR-17	SOP-0079, Reactor Protection System	17
AOP-0001	Reactor Scram	23
SOP-0011R21PR-22	Main Steam System	22
GOP-0003	Scram Recovery	19

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CR-RBS-2005-01103
CR-RBS-2005-02123
CR-RBS-2005-03261

1R04: Equipment Alignment

Applicability Determinations

Number	Title	Revision
SOP-0031	Residual Heat Removal	301
OSP-0017	Normal Control Board Lineups for Safety Related Systems	301
OSP-0037	Shutdown Operations Protection Plan (SOPP)	16
STP-000-0702	Primary Containment Shutdown Verification	14B
CR-RBS-2007-02326	Audible air leak on control panel for JRB-DRA2 (171' Airlock)	May 31, 2007
SOP-0042	Standby Service Water System	026
SOP-0053	Standby Diesel Generator and Auxiliaries	304

Section 1R17B: Permanent Plant Modifications

Engineering Changes

Number	Title	Revision
ER-RB-1999-0161	Removing Support to Inspect the Pipe Under the Support Collar	0
ER-RB-1999-0728	Limitorque Actuator to Increase Torque Output	0
ER-RB-1999-0794	Scheme to allow the use of normal station transformers	0
ER-RB-2004-0131	Replacement - cycle14 online design and implementation	0
ER-RB-2004-0367	Replace breakers 7 through 15 with ground fault interrupting breakers	0
ER-RB-2005-0087	Perform design to reinstall original transformer as STX-XS5A	0
ER-RB-2005-0088	Replace STX-XS2B	0
ER-RB-2005-0350	Delete condensate filter thermal relief valves	0
EC-00000252	Continuous backfill/flush on E31-PDTN084A/B	0

Procedures

Number	Title	Revision
EN-LI-101	10 CFR 50.59 Review Program	3
EN-LI-100	Process Applicability Determination	4
EDP-PE-09	Engineering Request Part Interchangeability Evaluation	8
EN-DC-112	Engineering Change Request and Project Initiation Process	0
EN-DC-114	Project Management	3
EN-DC-115	Engineering Change Development	2
EN-DC-116	Engineering Change Installation	0
ENS-DC-115	Engineering Request Response Development	10
FHP-0003	Refuel Platform Operation Procedure	20
Spec. No. 210.502	Field Application of Protective Coatings Inside Containment	3

Drawings

Number	Title	Revision
SK-EC 252	Panel H22-P004	0
PID-25-01G	System 051 Reference Leg Backfill System	2
PID-27-06A	System 209 Reactor Core Isolation Cooling	42
05-1678	Jib Crane	
R-STM-118	Service Water System	

Condition Reports

CR-RBS-2005-01732
CR-RBS-2005-02003
CR-RBS-2005-03255

Miscellaneous

Number	Title	Revision/Date
LBDC 09.01-091	Dry Fuel Storage System Considerations	Nov. 2005
LCN 9.01-051	Removal of JIB Crane from FSAR	0
RBC-50225	River Bend Station Unit 1 - Issuance of Amendment RE: Deletion of Shield Building Annulus Mixing System Technical Specifications	Oct. 2004
RBG-46183	License Amendment Request Deletion of Technical Specification 3.6.4.4 Shield Building Annulus Mixing System; and Revision of Main Steam Isolation Valve Surveillance Requirement SR 3.6.1.3.10	Oct. 2003
RBG-46222	Supplement to Amendment Request Deletion of Technical Specification 3.6.4.4 Shield Building Annulus Mixing System and Revision of Main Steam Isolation Valve Surveillance Requirement SR 3.6.1.3.10	Feb. 2004
RBG-46303	Supplement to Amendment Request Deletion of Technical Specification 3.6.4.4 Shield Building Annulus Mixing System and Revision of Main Steam Isolation Valve Surveillance Requirement SR 3.6.1.3.10	Aug. 2004

SDC-309	River bend System Design Criteria Standby Diesel Generator Division I and II, 309 Diesel Generator Building Ventilation, 405	3

Section 1R20: Refueling and Other Outage Activities

Number	Description	Revision
OSP-0034	Control of Obstructions for Primary Containment/Fuel Building Operability	3
OSP-0033	Operations with a Potential to Drain the Reactor Vessel/Cavity	16
OSP-0037	Shutdown Operations Protection Plan	16
CR-RBS-2007-02113	Recirculation Loop A flow decrease/Power decrease to 95 percent without operator action	May 27, 2007
CR-RBS-2007-02260	Actions taken in response to GE SILs 528 and 620	May 28, 2007
WO 00112116-01	Job Plan to Repair B33-MOVF067A	June 1, 2007
Evaluation # EN-2007-002	10 CFR 50.59 Evaluation for installation of jet pump plugs	
GOP-3	Scram Recovery	19
GMP-0102	Reactor Vessel Disassembly	16
STP-000-0702	Primary Containment Shutdown Verification	14B
E-mails	River Bend Forced Outage 07-03 hourly updates	Hourly
Temporary Procedure, TP07-0002	Reactor Recirc Loop A/B Discharge Valve OPDRV,	0
Temporary Procedure, TP07-0003	Draindown to Support B33-MOVF067A, Recirc Pump A Disch VLV – Maintenance	0
Assembly Drawing 94-13570	20X16X20-90M Venturi Welding Ends Outside Screw & Yoke 316 Stainless Steel Gate Valve with Smb-1 Limitorque Valve Control Discharge Valve	K

IPTE Briefing Sheets	Senior Manager	no date or revision
Primavera computer printouts	System “FO0703 RECIRC Flow Mismatch,” Schedule	per shift
	Engineering Matrix of requirements stated in EC-1541 to Controlling Document	no date or revision
Engineering P & I Diagram, PID-25-01C	Reactor Recirculation	26
Licensing Basis Document Change Request for USAR 5.4.1.3	Reactor Recirculation System Description	May 28, 2007
Contingency Action Plan (CAP)	Recirculation Pump discharge Valve	no date or revision
PNO-IV-07-005	Preliminary Notice of Event or Unusual Occurrence	ML071430449
	Operator Logs	every shift
WO 112116	Transport plan for B33-MOVF0067A/B	Undated
AZ Marine Contract PO Number 5270001	Container Testing and Certification Documentation	0
EC-0000001541	Operations with a Potential to Drain the Reactor Vessel	0

Section 1EP3: Emergency Response Organization Augmentation

EIP-2-016, "Operations Support Center," Revision 23
EIP-2-018, "Technical Support Center," Revision 29
EIP-2-020, "Emergency Operations Facility," Revision 28
EPP-2-202, “Emergency Response Organization,” Revision 11
EPP-2-502, "Emergency Communications Equipment Testing," Revision 22

Pager Test Checklists for Drills Conducted

2005 - November 11
2006 - March 9, June 13, September 11, November 6
2007 - March 20

Section 1EP5: Correction of Emergency Preparedness Weaknesses and Deficiencies

Procedure EIP-2-101, “Periodic Review of the Emergency Plan,” Revision 21
Procedure EIP-2-102, “Training, Drills, and Exercises,” Revision 25
Procedure EN-QV-109, “Audit Process,” Revision 9

Procedure EPP-2-201, “River Bend Station Emergency Preparedness Organization and Responsibilities,” Revision 19

Audit QA-7-2006-RBS-1, Emergency Preparedness Program, April 3 through 20, 2006

QA Surveillance Report QS-2006-RBS-005, August 21 through September 19, 2006
QA Surveillance Report QAS-2006-RBS-008, "Review of the Plant Paging System," October 21 through November 3, 2006

QA Surveillance Report QS-2007-RBS-005, June 2006 through April 2007
Emergency Planning Program Assessment, RLO-2006-00001
River Bend Station Emergency Preparedness Corporate Assessment, June 5-6, 2006
Emergency Planning Program Assessment, LO-RLO-2007-00038 CA 00001
Standing Order #196, “Interim Actions for Sensitivity to Systems with Risk Impact and Diagnosis Actions,” Revision 3

Training Evaluation Action Request 2007-14

Drills and Exercises Conducted

2005 - October 18
2006 - March 15, May 23, June 20, August 16, October 3
2007 - February 20

Condition Reports

2006-01141	2006-02837	2006-03543	2006-04348	2007-02020
2006-01486	2006-02921	2006-03599	2006-04645	2007-02023
2006-02605	2006-03003	2006-04270	2007-00110	2007-02051
2006-02730	2006-03264			

Section 4OA1: Performance Indicator (PI) Verification

EIP-2-001, "Classification of Emergencies," Revisions 14 to 16
EIP-2-006, "Notifications," Revisions 32, 33
EIP-2-007, "Protective Action Recommendations," Revision 21

Station Drill schedules for 2005, 2006, and 2007

Miscellaneous Documents

River Bend Station Emergency Plan, Revision 31

LIST OF ACRONYMS

ASME	American Society of Mechanical Engineers
CAP	corrective action program
CFR	*Code of Federal Regulations*
CR	condition report
CR-RBS	River Bend Station condition report
LEFM	leading edge flow meter
LOCA	loss of coolant accident
NCV	noncited violation
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
PI	Performance Indicators
RCS	reactor coolant system
RDGV	recirculation discharge gate valve
SDIV	scram discharge instrument volume
SSC	structures, systems, and components
STP	surveillance test procedure
TS	Technical Specifications
USAR	Updated Safety Analysis Report
WO	work order