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

October 22, 2003

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

SUBJECT: RIVER BEND STATION - NRC INTEGRATED INSPECTION REPORT 05000458/2003005 and 07200049/2003001

Dear Mr. Hinnenkamp:

On September 27, 2003, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your River Bend Station facility. The enclosed integrated inspection report documents the inspection findings, which were discussed on September 30, 2003, 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 finding concerning an air-bound normal service water pump. This issue has a safety significance that is potentially greater than very low significance. No immediate safety concern exists because the condition that caused this pump to be air bound has been corrected. The risk assessment for this issue is ongoing, and you will be notified when the significance is determined. Additionally, this report documents one NRC-identified and two self-revealing issues that were identified and evaluated under the risk significance determination process as having very low safety significance (Green). Two of these were determined to involve violations of regulatory requirements. However, because of the very low safety significance and because they were entered into your corrective action program, the NRC is treating these violations as noncited violations (NCVs) consistent with Section VI.A 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 U.S. Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region IV; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at River Bend Station.

In accordance with 10 CFR 2.790 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 Publically Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

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

Sincerely

/RA/

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

Dockets: 50-458 and 72-049
License: NPF-47

Enclosure:
NRC Inspection Report 05000458/2003005 and 07200049/2003001
w/Attachment: Supplemental Information

cc w/enclosure:
Senior Vice President and Chief Operating Officer
Entergy Operations, Inc.
P.O. Box 31995
Jackson, MS 39286-1995

Vice President
Operations Support
Entergy Operations, Inc.
P.O. Box 31995
Jackson, MS 39286-1995

General Manager
Plant Operations
River Bend Station
Entergy Operations, Inc.
P.O. Box 220
St. Francisville, LA 70775

Director - Nuclear Safety
River Bend Station
Entergy Operations, Inc.
P.O. Box 220
St. Francisville, LA 70775

Wise, Carter, Child & Caraway
P.O. Box 651
Jackson, MS 39205

Mark J. Wetterhahn, Esq.
Winston & Strawn
1401 L Street, N.W.
Washington, D.C. 20005-3502

Manager - Licensing
River Bend Station
Entergy Operations, Inc.
P.O. Box 220
St. Francisville, LA 70775

The Honorable Richard P. Ieyoub
Attorney General
Department of Justice
State of Louisiana
P.O. Box 94005
Baton Rouge, LA 70804-9005

H. Anne Plettinger
3456 Villa Rose Drive
Baton Rouge, LA 70806

President
West Feliciana Parish Police Jury
P.O. Box 1921
St. Francisville, LA 70775

Michael E. Henry, State Liaison Officer
Department of Environmental Quality
Permits Division
P.O. Box 4313
Baton Rouge, LA 70821-4313

Brian Almon
Public Utility Commission
William B. Travis Building
P.O. Box 13326
1701 North Congress Avenue
Austin, TX 78701-3326

Technological Services
Branch Chief
FEMA Region VI
800 North Loop 288
Federal Regional Center
Denton, TX 76201-3698

Electronic distribution by RIV:
Regional Administrator **(BSM1)**
DRP Director **(ATH)**
DRS Director **(DDC)**
Senior Resident Inspector **(PJA)**
Branch Chief, DRP/B **(DNG)**
Senior Project Engineer, DRP/B **(RAK1)**
Staff Chief, DRP/TSS **(PHH)**
RITS Coordinator **(NBH)**
J. Clark **(JAC)**, OEDO RIV Coordinator
Dale Thatcher **(DFT)**
W. A. Maier, RSLO **(WAM)**
RBS Site Secretary **(LGD)**

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<u>ENCLOSURE</u>

U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	50-458, 72-049
License:	NPF-47
Report No:	05000458/2003005 and 07200049/2003001
Licensee:	Entergy Operations, Inc.
Facility:	River Bend Station
Location:	5485 U.S. Highway 61 St. Francisville, Louisiana
Dates:	June 29 through September 27, 2003
Inspectors:	P. J. Alter, Senior Resident Inspector, Project Branch B M. O. Miller, Resident Inspector, Project Branch B J. V. Everett, Senior Inspector, Division of Nuclear Materials Safety L. T. Ricketson, P.E., Senior Health Physicist, Plant Support Branch
Approved By:	D. N. Graves, Chief Project Branch B Division of Reactor Projects

CONTENTS

SUMMARY OF FINDINGS

IR 05000458/2003005 and IR 07200049/2003001; 06/29/2003 - 09/27/2003; River Bend Station; ALARA Planning and Controls.

This report covered a 13-week period of routine inspection by resident inspectors and announced inspections by regional ALARA, security, and independent spent fuel storage inspectors. One unresolved item, that has its risk significance yet to be determined, and one Green finding 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. The inspectors identified a self-revealing violation of Technical Specification 5.4.1 because operators lined up service water to the reactor plant and turbine plant cooling water systems such that an automatic start of standby service water occurred on low system pressure while shifting normal service water pumps. Three heat exchangers in each system were in service when the operating procedures allow only two per system.

  This finding is greater than minor because it was associated with the ability to meet the mitigating systems cornerstone objective and because a plant transient occurred. The inspectors determined that the finding was of very low safety significance (Green), since the finding did not represent an actual loss of safety function of a single train (Section 4OA3).

- Green. The inspectors identified a self-revealing violation for failure to comply with Technical Specification 5.4.1.a. Operators mistakenly racked out the high pressure core spray pump breaker when implementing a clearance order on a standby service water.

  This self-revealing finding was more than minor because the high pressure core spray safety function was made unavailable. The inspectors reviewed the finding using Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." The inspectors determined that the finding was of very low safety significance (Green) because the high pressure core spray pump was not functional for less than one hour. Recovery credit was given for operator actions necessary to restore the equipment lineup and recover the safety function (Section 4OA3).

Cornerstone: Initiating Events

- (TBD). The inspectors identified a self-revealing apparent violation of Technical Specification 5.4.1.a, the significance of which has yet to be determined. A human performance error caused the isolation of the air release valve for normal service water Pump C. The air release valve for a normal service water pump served as a high point vent on the system while the pump was secured. As a result, normal service water Pump C became air bound while in standby and failed to develop discharge pressure when started during a manual swap of running normal service water pumps on September 1, 2003.

  The inspectors determined that the failure to maintain normal service water Pump C discharge air release valve isolation Valve SWP-V3312C open was an apparent violation of normal service water system operating Procedure SOP-0018, Attachment 1A, “Valve Lineup - Normal Service Water,” Revision 32. The issue was more than minor because it was associated with an increase in the likelihood of an initiating event (loss of normal service water). The inspectors reviewed this finding using Inspection Manual Chapter 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations.” The result of the phase one screening process and the inspectors’ review of the increased likelihood of a loss of normal service water was that further review of the risk potential for this condition was necessary (Section 4OA3).

Cornerstone: Occupational Radiation Safety

- Green. The inspector identified an ALARA finding because performance deficiencies resulted in a collective dose of the work activity that exceeded 5 person-rem and exceeded the legitimate dose estimation by more than 50 percent. Specifically, radiation work Permit 2003-1800, "RF-11 Refueling Activities," accrued 34.962 person-rem and exceeded the dose estimate (19.939 person-rem) by 75 percent. A primary cause for the unplanned dose was the licensee’s failure to effectively schedule the use of the alternate decay heat removal system, a system which had previously proven to be effective at removing radioactivity from the refueling pool. The licensee also failed to limit the number of personnel on the refueling bridge to the planned number, thus causing the work activity to accrue more collective dose than estimated. A contamination incident during the disassembly of the reactor vessel was caused by poor planning and required additional time for cleanup.

  This finding was more than minor because it was associated with the occupational radiation safety cornerstone attribute (ALARA planning/estimated dose) and affected the associated cornerstone objective (to ensure adequate protection of worker health and safety from exposure to radiation). The finding involved a failure to maintain or implement, to the extent practical, procedures or engineering controls needed to achieve occupational doses that were ALARA and resulted in unplanned, unintended occupational collective dose for a work activity. When processed through the occupational radiation safety significance determination process, this ALARA finding was found to have no more than very low safety significance because the licensee's 3-year rolling average collective dose was not greater than 240 person-rem (Section 2OS2).

REPORT DETAILS

Summary of Plant Status: At the beginning of the inspection period, River Bend Station (RBS) was operating at 100 percent power. Operators changed reactor power only for routine control rod exchanges and tests until September 16, 2003. On that date, operators conducted an unplanned reactor power reduction to 98 percent, held reactor power at 98 percent for 1.5 hours, and then increased reactor power to 100 percent. This was in response to a loss and recovery of the plant process computer. Operators reduced reactor power to 78 percent for rod sequence exchange and turbine control valve (TCV) testing on September 22, 2003, at 7 p.m. At 10:43 p.m., the reactor scrammed on high reactor pressure when the first TCV was tested. The high reactor pressure was caused when an erroneous overspeed signal from the backup turbine speed sensor caused the TCVs to go closed. Operators resynchronized RBS to the grid on September 24, 2003, at 9:59 a.m. RBS attained 100 percent power on September 27, 2003, at 4:02 a.m. Operators began a power reduction to 65 percent for a control rod sequence exchange on September 28, 203, at 8:41 p.m. RBS was returned to 100 percent power on September 29, 2003, at 9:50 a.m. following the rod sequence exchange. RBS remained at 100 percent power for the remainder of the inspection period.

1. REACTOR SAFETY

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

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

Tropical Storm Warning

One weather event was sampled. On June 30, 2003, the inspectors verified the performance of a risk assessment in preparation for the arrival of tropical storm Bill. The inspectors interviewed the duty manager and verified performance of risk assessments, in accordance with administrative Procedure ADM-096, “Risk Management Program Implementation and On-Line Maintenance Risk Assessment,” Revision 04, for the planned maintenance involving structures, systems, or components (SSC) within the scope of the maintenance rule. Specific work activities evaluated included planned work on the following systems and activities:

- Reactor core isolation cooling (RCIC) system
- Main steam line isolation valve logic system
- Reactor plant component cooling water (CCP) Pump CCP-P1A
- Fuel handling in the lower spent fuel pool

b. Findings

No findings of significance were identified.

1R04 Equipment Alignments (71111.04)

a. Inspection Scope

The inspectors performed three partial equipment alignment verifications (partial system walkdowns) during this inspection period. On August 4, 2003, the inspectors walked down residual heat removal (RHR) Train B while RHR Pump A was out of service for scheduled maintenance. On September 11, 2003, the inspectors walked down two qualified circuits between the offsite transmission network and onsite Division III Class 1E electric power distribution system while the Division III diesel generator was out of service for a broken fuel line. On September 11, 2003, the inspectors walked down the RCIC system while the high pressure core spray (HPCS) system breaker was inoperable due to inadequate injection line pressure. In each case, the inspectors verified the correct valve and power alignments by comparing positions of valves, switches, and electrical power breakers to the procedures listed below:

- SOP-0031, “Residual Heat Removal System,” Revision 40
- STP-000-0102, “Power Distribution Alignment Check,” Revision 4
- SOP-0035, “Reactor Core Isolation Cooling System,” Revision 21

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

.1 The inspectors walked down accessible portions of six areas described below to assess: (1) the licensee’s control of transient combustible material and ignition sources; (2) fire detection and suppression capabilities; (3) manual firefighting equipment and capability; (4) the condition of passive fire protection features, such as, electrical raceway fire barrier systems, fire doors, and fire barrier penetration; and (5) any related compensatory measures. The areas inspected were:

- Control building, 98-foot elevation, standby switchgear Room 1B, Fire Zone C-14, on August 4, 2003

- Auxiliary building, 70-foot elevation, RHR B pump room, Fire Area AB-3, on August 4, 2003

- Control building, 98-foot elevation, safety-related cable Chase II, Fire Zone C-2B, on August 4, 2003

- Auxiliary building, 114-foot elevation, west vital motor control center area, Fire Zone AB-1/Z-3, on August 15, 2003

- Fuel building, 95-foot elevation, recirculation pump Motor A switchgear room, Fire Zone FB-1/Z-2, on September 15, 2003

- Normal switchgear building, 98-foot elevation, alternate 4160 VAC supply to emergency switchgear and power supply for reactor feed pumps and condensate pumps, Fire Area NS-98, on September 15, 2003

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

- Pre-Fire Strategy Book
- Updated Safety Analysis Report (USAR) Section 9A.2, “Fire Hazards Analysis”
- RBS postfire safe shutdown analysis
- RBNP-038, “Site Fire Protection Program,” Revision 06A

.2 On August 14, 2003, the inspectors observed one fire brigade drill in the auxiliary building in the vicinity of the reactor recirculation pump fast speed breakers to evaluate the readiness of the licensee’s personnel to prevent and fight fires. The inspectors also verified that the pre-planned drill scenario was followed and that the drill objectives acceptance criteria were met. Specific criteria evaluated included: (1) the proper wear and use of self-contained breathing apparatus, (2) clear communications being used by fire brigade members, (3) proper use and handling of portable fire extinguishers, and (4) hose lines capable of reaching the fire hazzards.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors conducted one periodic flooding assessment to verify that the licensee’s flooding mitigation plans and equipment were consistent with design requirements and risk analysis assumptions. The inspectors conducted a walkdown of the RHR System B equipment room on August 4, 2003. Specifically, the inspectors examined five items: (1) sealing surfaces of watertight doors, (2) sealing of equipment below design flood level, (3) sealing of penetrations in floors and walls, (4) operable sump pumps and level alarm circuits, and (5) sources of potential internal flooding from plant systems. The documents reviewed by the inspectors during this inspection as the bases for acceptability of the plant configuration are listed in the attachment.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

.1 On July 31, 2003, the inspectors observed requalification program simulator training of an operation department staff crew, as part of the operator requalification training program, to assess licensed operator performance and the training evaluator's critique. Emphasis was placed on observing an annual evaluation exercise of high risk licensed operator actions, operator activities associated with the emergency plan, and lessons learned from industry and plant experiences. In addition, the inspectors compared simulator control panel configurations with the actual control room panels for consistency, including recent modifications implemented in the plant. The simulator training scenario observed was RSMS-OPS-805, "Loss of Feedwater Heating/DBA LOCA," Revision 3.

.2 On September 2, 2003, an operations department staff crew failed simulator training Scenario RSMS-OPS-622, "Loss of CRD/Loss of Vacuum with MSIV Closure/ATWS," Revision 3. The inspectors interviewed the lead examiner and observed the team debrief, as part of the operator requalification training program, to assess the training examiner's critique and the team's response to this failure. On September 11, 2003, the operations department staff team was re-evaluated in the simulator by examiners using simulator training Scenario RSMS-OPS-0801, "Open SRV/EHC Regulator Failure/ATWS," Revision 2. The inspectors interviewed the lead examiner and reviewed the team and individual evaluations that were documented by the examiners. On September 25, 2003, the inspectors observed the review of the re-evaluation of crew performance.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

The inspectors reviewed System 2 performance problems to assess the effectiveness of the licensee's maintenance efforts for SSC within the scope of the maintenance rule program. The inspectors verified the licensee's maintenance effectiveness by: (1) verifying the licensee's handling of SSC performance or condition problems, (2) verifying the licensee's handling of degraded SSC functional performance or condition, (3) evaluating the role of work practices and common cause problems, and (4) evaluating the licensee's handling of the SSC issues being reviewed under the requirements of the maintenance rule (10 CFR 50.65), 10 CFR Part 50, Appendix B, and the Technical Specifications.

- CR-RBS-2002-1175, Station blackout diesel tripped again on high coolant temperature approximately 4 minutes after starting for testing, reviewed September 4, 2003

- CR-RBS-2003-02673, Failure of both floor drain pumps in RHR equipment Room B

The following documents were reviewed as part of this inspection:

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

- Maintenance rule function list

- Maintenance rule performance criteria list

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

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

.1 Risk Assessment and Management of Risk

On a routine basis, the inspectors verified performance of risk assessments, in accordance with administrative Procedure ADM-096, “Risk Management Program Implementation and on-line Maintenance Risk Assessment,” Revision 04, for planned maintenance activities and emergent work involving SSC within the scope of the maintenance rule. Specific work activities evaluated included planned and emergent work for the week of August 31, 2003.

.2 Emergent Work Controls

The inspectors reviewed licensee activities associated with re-routing of the reactor protection system alternate power supply output to Panel SCM-PNL01A1. The inspectors verified that the licensee took actions to minimize the probability of initiating events, maintained the functional capability of mitigating systems, and maintained barrier integrity. The inspectors also reviewed the activities to ensure the plant was not placed in an unacceptable configuration.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

.1 De-energize Division I Control Room Instrument AC Panel

The inspectors observed performance of operations and electrical maintenance personnel while de-energizing control room instrument AC Panel SCM-PNL01A1 to allow installation and removal of a temporary power feed to the panel for troubleshooting of its power line conditioner supply Transformer SCM-XRC14A1. During the inspection, the inspectors reviewed the plan for the Panel SCM-PNL01A1 outage and observed the prejob briefings conducted in the main control room, de-energizing of the panel on July 11, 2003, and restoration of the normal power supply on July 18, 2003. The inspectors reviewed abnormal operating Procedure AOP-0042, “Loss of Instrument Bus,” Revision 18, used by the operators to develop the contingency procedures for operation of systems affected by the panel outage and the control room logs for proper adherence to Technical Specifications and the Technical Requirements Manual (TRM).

.2 Unscheduled Power Reduction During Loss of Core Monitoring System and Plant Process Computer

The inspectors evaluated operator performance in the control room on September 16, 2003, during an unplanned event. The operators performed an unplanned power reduction of approximately 20 megawatts electric that lasted approximately 1.5 hours. The inspectors determined that operator actions were in accordance with the requirements of general operating Procedure GOP-0005, “Power Maneuvering,” Revision 12. The inspectors evaluated the initiating causes, and the immediate actions taken, in response to failure of the plant process computer as documented in CR-RBS-2003-3158. The inspectors also noted that actions taken were in accordance with the requirements of TRM 3.3.13, “Ultrasonic Feedwater Flow Meters,” and TRM Llimiting condition for operation 3.0.3, “TLCO Not Met and Associated Actions Are Not Met.”

.3 Reactor Scram During TCV Testing

The inspectors observed operations and engineering personnel performance during TCV testing on September 22, 2003. The inspectors observed the prejob briefing and the final preparations for this test. At the conclusion of the test of the first TCV, the reactor scrammed. The inspectors observed operator performance immediately prior to, during, and following the reactor scram. The inspectors reviewed four procedures to assess operator performance during the transient: (1) emergency operating Procedure EOP-001, “RPV Control,” Revision 20; and (2) abnormal operating Procedures AOP-1, “Reactor Scram,” Revision 19; AOP-2, “Main Turbine and Generator Trip,” Revision 16; and AOP-3, “Automatic Isolations,” Revision 18.

The inspectors also evaluated the classification of this event using the criteria established in emergency implementing Procedure EIP-2-001, “Classification of Emergencies,” Revision 12.

.4 Reactor Startup following Forced Outage FO-03-03

The inspectors observed operations and reactor engineering personnel performance during a reactor startup on September 23, 2003. The inspectors evaluated the hot startup approach to criticality, achievement of criticality, reactor power increase to the point of adding heat, and power ascension to the point of one turbine bypass valve opening. The inspectors referred to the following procedures to assess operator performance during the startup: general operating Procedure GOP-001, “Plant Startup,” Revision 41, and system operating Procedure SOP-0071, “Rod Control and Information System,” Revision 71.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed two operability determinations selected on the basis of risk insights. The selected samples are addressed in the condition reports listed below. The inspectors assessed: (1) the accuracy of the evaluations, (2) the use and control of compensatory measures if needed, and (3) compliance with Technical Specifications, TRM, USAR, and other associated design-basis documents. The inspectors’ review included a verification that the operability determinations were made as specified by Procedure RBNP-078, “Operability Determinations,” Revision 7.

- CR-RBS-2003-2661, justification for continued operation with a 4.5 inch crack in jet pump support beam for jet Pumps 19 and 20, reviewed during the week of August 25, 2003

- CR-RBS-2003-3014, an unexpected rise in narrow range reactor water level instrument Channel B with reference leg backfill secured for planned maintenance, reviewed during the weeks of August 25 and September 1, 2003

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed five maintenance action items (MAI)/work order packages to assess the adequacy of testing activities to verify system operability and functional capability. The inspectors performed the following: (1) identified the safety function(s) for each system by reviewing applicable licensing basis and/or design-basis documents; (2) reviewed each maintenance activity to identify which maintenance function(s) may have been affected; (3) reviewed each test procedure to verify that the procedure did adequately test the safety function(s) that may have been affected by the maintenance activity; (4) reviewed that the acceptance criteria in the procedure to ensure consistency with information in the applicable licensing basis and/or design-basis documents; and (5) identified that the procedure was properly reviewed and approved. The five postmaintenance tests inspected are listed below:

- MAI 375199, Troubleshoot and repair dual position indication for containment pools to purification system outboard isolation Valve SFC-MOV122, reviewed August 15, 2003

- Work Order Package 00028640 01, Corrective fuel oil leak on Division III diesel generator, reviewed September 11, 20003

- MAI 373268, Preventive maintenance on HPCS pump discharge line fill pump, reviewed September 12, 2003

- MAI 355633, Replacement of the extraction steam inlet nozzle and shell section of low pressure feedwater Heater CNM-E4A, conducted on September 16, 2003.

- MAI 363512, Corrective maintenance on uninterruptible Power Supply SCM-PNL01A following an overheating event, conducted on September 16, 2003

b. Findings

No findings of significance were identified.

1R20 Forced Outage Activities (71111.20)

a. Inspection Scope

The inspectors interviewed the Outage Manager to ascertain the risk assessment conducted related to a forced outage conducted on September 23, 2003, to assess whether the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing the outage and restart plans. The inspectors evaluated control room activities during the forced outage using the criteria documented in general operating Procedure, GOP-0002, "Power Decrease/Plant Shutdown," Revision 28. The inspectors interviewed the General Manager, Director of Licensing, and Engineering Director to appraise the restart decision process. During the forced outage, the following outage activities were observed:

- Initial outage planning meeting
- Outage control center activities and turnover
- Significant event review team activities
- Control room activities at various times during the forced outage

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors assessed, by witnessing and reviewing test data, whether three risk-significant system and component surveillance tests met Technical Specification, USAR, and procedure requirements. The inspectors reviewed whether the surveillance tests demonstrated operational readiness and whether the systems were capable of performing their intended safety functions. The inspectors reviewed the following surveillance test attributes: (1) preconditioning; (2) clarity of acceptance criteria; (3) range, accuracy, and current calibration of test equipment; and (4) that equipment was properly restored at the completion of the testing. The inspectors observed and reviewed the following surveillance tests and surveillance test procedures (STP):

- STP-051-4229, ADS B Timer Channel Functional Test and Channel Calibration (B21C-K5B), Revision 7A, performed on July 30, 2003

- STP-051-4298, ADS A Drywell Pressure Bypass Timer Channel Functional Test and Channel Calibration (B21C-K114A), Revision 6, performed on July 30, 2003

- STP-051-4299, ADS B Drywell Pressure Bypass Timer Channel Functional Test and Channel Calibration (B21C-K114B), Revision 7, performed on July 30, 2003

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

On July 11, 2003, the inspectors reviewed one temporary modification to power control room instrumentation ac Bus SCM-PNL01A from reactor protection system (RPS) alternate power Supply RPS-XRC10A1 in order to troubleshoot the control room instrument ac power supply. On July 18, 2003, the inspectors observed the restoration of normal power supplies to both instrument ac and the RPS. The inspectors conducted the following: (1) reviewed the temporary modification and its associated 10 CFR 50.59 screening against the system design-basis documentation, including the USAR and Technical Specifications; (2) verified that the installation and removal of the temporary modification was consistent with the modification documents; (3) verified that plant drawings and procedures were updated; and (4) reviewed the postinstallation and removal test results to confirm that the actual impact of the temporary modification on both the control room instrument ac and the RPS power supplies had been adequately verified.

b. Findings

No findings of significance were identified.

Emergency Preparedness [EP]

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors observed two emergency preparedness simulator training exercises conducted on July 31 and August 7, 2003, to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation development activities. The inspectors also evaluated the licensee assessment of classification, notification, and protective action recommendation development during the exercises, in accordance with plant procedures and NRC guidelines. The following procedures and documents were reviewed during the assessment:

- EIP-2-001, “Classification of Emergencies,” Revision 11

- EIP-2-006, “Notifications,” Revision 27

- RSMS-OPS-804, “Main Turbine Trip/ATWS with SLC Failure/SRV Relief Failure,” Revision 3, on July 31, 2003

- RSMS-OPS-803, “Trip of RPS MG Set/Relief Valve Fails Open/Steam Leak in Drywell,” Revision 3, on August 7, 2003

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

The inspector interviewed radiation protection staff and other radiation workers to determine the level of planning, communication, ALARA practices, and supervisory oversight integrated into Refueling Outage 11 work activities. The inspector focused on work activities completed since March 28, 2003. Additionally, the following items were reviewed and compared with regulatory requirements to assess the licensee’s program to maintain occupational exposures as low as reasonably achievable (ALARA):

- ALARA program procedures

- Processes, methodology, and bases used to estimate, justify, adjust, track, and evaluate exposures

- Radiation Work Permit (RWP) packages, including ALARA prejob, in-progress, and postjob reviews for RWP 2003-1800, “Refueling Activities”; RWP 2003-1936, “Installation and Removal of Temporary Shielding in the Drywell”; and RWP 2003-1950, “Scaffolding in the Drywell”

- The use and result of administrative and engineering controls to achieve dose reductions

- Plant source term evaluation and control strategy/program

- ALARA Committee meeting minutes and presentations

- Quality Assurance Surveillances (RBS QA Surveillance Reports QS-2003-RBS-008 and QS-2003-RBS-009)

- Radiation Protection Self-Assessment/ALARA Program (June 2-5, 2003)
- Implementation of 10 CFR 20.1703(f)

b. Findings

Introduction. The inspector identified a Green ALARA finding because performance deficiencies resulted in the collective dose of a work activity that exceeded 5 person-rem and exceeded the dose estimation by more than 50 percent.

Description. The licensee estimated that RWP 2003-1800, "RF-11 Refueling Activities," would accrue 19.939 person-rem of collective dose. Instead, the actual dose for the work activity was 34.962 person-rem or 175 percent of the original dose estimate. A primary cause for the unplanned dose was the licensee’s failure to effectively schedule the use of the alternate decay heat removal system (ADHRS) to remove radioactivity from the refueling pool water.

According to the licensee, the ADHRS demineralizers had been very effective in removing cobalt from the refueling pool water and lowering dose rates during previous outages. The use of the ADHRS was discussed before the outage during ALARA Committee Meeting 02-06, conducted August 22, 2002, but the ALARA committee failed to take assertive action to ensure that the use of the system was included on the outage schedule. Consequently, because the use of ADHRS was not on the outage schedule, the importance of having the system available was not recognized by the operations staff. Work on valves necessary to operate the system was conducted early in the outage, delaying the systems availability. When the ADHRS would have been of most benefit, the system was not available. Instead of putting ADHRS into service on the 4th day of the outage as originally discussed during ALARA Committee Meeting 02-06, the system was placed into service on the 8th day. The licensee estimated that this failure resulted in approximately 9 person-rem of additional, unplanned collective dose. The licensee adjusted the dose estimate for RWP 2003-1800 to account for the effect of the increased source term. However, the inspector concluded that the increased dose was the result of a performance deficiency (ineffective planning and scheduling) and that the revision to the dose estimate was not valid.

Additional performance deficiencies contributed to the unplanned dose accrued by RWP 2003-1800. The licensee’s in-progress and postjob reviews documented that more workers than planned were allowed to stay on the refueling bridge, thus adding to the dose total. Fuel bundles were mispositioned and had to be moved again because of control issues with the fuel movement plans. An event discussed in NRC Inspection Report 50-458/03-03 spread contamination throughout the containment building and required decontamination personnel to spend more time than planned on cleanup activities. The event also impacted the collective radiation dose received by the radiation protection organization. Because of the perceived need for greater oversight following the event, the radiation protection control point was moved to the refueling floor, resulting in increased dose. The dose contribution attributable to each performance deficiency individually was not known.

Analysis. This finding was more than minor because it was associated with the Occupational Radiation Safety Cornerstone attribute (ALARA planning/estimated dose) and affected the associated cornerstone objective (to ensure adequate protection of worker health and safety from exposure to radiation). The finding involved a failure to maintain or implement, to the extent practical, procedures or engineering controls needed to achieve occupational doses that were ALARA and that resulted in unplanned, unintended occupational collective dose for a work activity. When processed through the Occupational Radiation Safety Significance Determination Process, this ALARA finding was found to have no more than very low safety significance because the licensee's 3-year rolling average collective dose was not greater than 240 person-rem. The finding was documented in the licensee's corrective action program as CR-RBS-2003-1213 (FIN 05000458/2003005-01).

Enforcement. No violation of regulatory requirements occurred.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

The inspectors reviewed submissions for the five performance indicators (PI) listed below spanning the period from July 2002 through June 2003. To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Indicator Guideline," Revision 2, were used to verify the basis in reporting for each data element.

.1 Mitigating Systems Cornerstone

- Heat Removal System (RCIC)

The inspector reviewed the performance indicator technique sheets to determine whether the licensee satisfactorily identified the required data reporting elements. This information was compared to the information reported for the PI during the inspection period for accuracy. The inspectors also sampled the maintenance rule database, portions of operator log entries, and portions of limiting conditions for operation log entries to verify the accuracy of the data reporting elements, the licensee's basis for crediting system availability, and the calculation of the average system unavailability for the previous 12 quarters. The inspectors also interviewed licensee personnel associated with the PI data collection, evaluation, and distribution.

.2 Barrier Integrity Cornerstone

- Reactor Coolant System (RCS) Leakage

The inspector reviewed the PI technique sheets to determine whether the licensee satisfactorily identified the required data reporting elements. This information was compared to the information reported for the PI during the inspection period for accuracy. The inspectors also reviewed shift logs and report outputs from the leakage computer for RCS leakage data to verify the accuracy of the data reporting elements for the previous 12 quarters on a sampling basis. The inspectors also interviewed licensee personnel associated with the PI data collection, evaluation, and distribution.

.3 Physical Protection Cornerstone

- Protected area security equipment
- Personnel screening program performance
- Fitness-For-Duty/Personnel Reliability program performance

The inspectors reviewed the licensee's security program for collection and submittal of PI data. Specifically, a random sampling of security event logs and corrective action reports from June 1, 2002, through May 30, 2003, were reviewed.

b. Findings

RCS identified leakage data was collected at the Technical Specification frequency of every 12 hours. However, RCS identified leakage data was calculated by the leakage monitoring computer and was available on a more frequent basis. The data was printed at 15-minute intervals. Given that the data was available more frequently than at the 12-hour frequency of the Technical Specifications, a more accurate value may be available.

NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 2, also states that the maximum RCS leakage was to be used in calculating the PI. The licensee was using the 24-hour average leakage rate instead of identifying and using a maximum leakage rate as determined on a more frequent basis.

The inspector questioned whether NEI 99-02 guidance was being appropriately applied regarding RCS leakage calculations. This issue will remain open until further clarification is obtained (URI 05000458/200305-02).

4OA2 Identification and Resolution of Problems (71152)

1. Occupational Radiation Safety Sample Review

a. Inspection Scope

The inspector reviewed selected corrective action documents involving exposure tracking, higher-than-planned exposure levels, radiation worker and radiation protection practices, and repetitive deficiencies since the last inspection in this area in March 2003. The selected corrective action documents are listed in the attachment to this inspection report. The inspector used regulatory and procedural requirements as criteria for determining the adequacy of the licensee's problem identification and resolution results.

b. Findings and Observations

No findings of significance were identified.

2. Physical Security Annual Sample Review

a. Inspection Scope

The inspectors evaluated licensee activities to determine whether the licensee appropriately resolved conditions adverse to quality, which included identifying the root cause, implementing appropriate corrective actions, and trending lower level deficiencies. The inspectors reviewed 12 condition reports related to compensatory measures, listed in the attachment to this report. In addition, the inspectors reviewed the multi-site Quality Assurance Audit of the Security Program, Report QA-16-2001-W3-1-Multi-site, dated December 21, 2001, and Quality Assurance Audit Report, QA-16-2002-GGNS-1-Multi-site, dated November 4 through December 10, 2002. The inspectors also reviewed RBS QA Surveillance Reports QS-2002-RBS-001, QS-2002-RBS-013, and QS-2002-RBS-019.

b. Findings and Observations

No findings of significance were identified.

4OA3 Event Followup (71153)

1. (Closed) Licensee Event Report (LER) 05000458/2003-002-01, Secondary Containment Door Failure Due to Malfunction of Door Assist Device

On March 7, 2003, with the plant in Mode 1, a personnel access door to secondary containment failed open when its door assist device (DAD) failed. Each of three normal access doors to the auxiliary building from the turbine building were equipped with a DAD to allow access when the standby gas treatment system was running, because of the high differential pressure across the doors. The DAD was removed from the door and secondary containment was reestablished in 78 minutes, well within the allowed time permitted by Technical Specifications. The failure was internal to the operating mechanism of the DAD. The DAD was repaired and all of the other DADs were inspected. The inspectors reviewed the LER and the root cause analysis and corrective actions documented in CR-RBS-2003-0865. No findings of significance were identified. This LER is closed.

2. (Closed) LER 50-458 /200306-01, Automatic Initiation of Standby Service Water (SSW) System Due to Inadequate Control of System Operation

a. Inspection Scope

Inspectors reviewed the LER and CR-RBS-2003-2054, which documented this event in the corrective action program, to verify that the cause of the May 7, 2003, automatic initiation of Division II SSW was identified and that corrective actions were reasonable. The automatic initiation of Division II SSW was caused by having too many heat exchanger flowpaths open in the normal service water (NSW) while swapping NSW pumps, causing the system pressure to decrease to the standby service water system automatic initiation setpoint. The inspectors reviewed plant parameters and verified that licensee staff properly implemented the appropriate plant procedures, and that plant equipment performed as required.

b. Findings

Introduction. The inspectors identified a self-revealing, Green, noncited violation for failure to comply with Technical Specification 5.4.1.a by failing to correctly implement system operating procedures.

Description. On May 7, 2003, at approximately 9:05 a.m., an unplanned initiation of Division II SSW occurred while swapping running NSW pumps due to an unexpected low pressure condition in the NSW system. This caused an automatic start of the Division II SSW system.

The inspectors reviewed the LER, the root cause analysis, and corrective actions documented for this issue in CR-RBS-2003-2054. The licensee found that three heat exchangers in the reactor plant CCP system and three heat exchangers in the turbine plant component cooling water (CCS) system were incorrectly aligned such that NSW was flowing through all six heat exchangers. System operating Procedure (SOP) SOP-0016, “Reactor Plant Component Cooling Water System,” Revision 20A, required that two heat exchangers be placed in service in the CCP system. SOP-0017, “Turbine Plant Component Cooling Water System,” Revision 13A, required that two heat exchangers are placed in service in the CCS system. This is a performance deficiency (human performance error).

Because six heat exchangers for CCP and CCS provided flowpaths for NSW instead of the required four heat exchangers, the flow through the NSW system was higher than normal and the system pressure was lower than normal. Normally, two of the three NSW pumps were running. The operators shutdown one of the NSW pumps to swap running NSW pumps. NSW pressure dropped below the SSW initiation setpoint, and Division II SSW system automatically initiated.

Analysis. This finding was more than minor because it was associated with the ability to meet the mitigating systems cornerstone objective and because a plant transient occurred. The inspectors reviewed the finding using Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." The inspectors determined that the finding was of very low safety significance (Green), since the finding did not represent an actual loss of safety function of a single train.

Enforcement. System operating Procedures SOP-0016, "Reactor Plant Component Cooling Water System," Revision 20A, and SOP-0017, "Turbine Plant Component Cooling Water System," Revision 13A, were not properly implemented. This issue is a violation of Technical Specification 5.4.1.a which requires that written procedures be established, implemented, and maintained according to the applicable recommendations in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Item 4.1 of Regulatory Guide 1.33, Revision 2, Appendix A, recommends procedures for closed cooling water systems and this system is applicable to RBS.

This human performance error is associated with an inspection finding that is characterized by the significance determination process (SDP) as having very low risk significance (Green) and is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC enforcement policy. This issue is in the licensee's corrective action program as CR-RBS-2003-02054 (NCV 05000458/2003005-03).

3. (Closed) LER 05000458/200307-00, HPCS Inadvertently Disabled Due to Personnel Error During Installation of Clearance Order

a. Inspection Scope

The inspectors reviewed the LER and CR-RBS-2003-02437, which documented this event in the corrective action program, to verify that the cause of the June 17, 2003, HPCS function being disabled was identified and that corrective actions were reasonable. The inspectors reviewed plant parameters and verified that licensee staff properly implemented the appropriate plant procedures and that plant equipment was restored as required.

b. Findings

Introduction. The inspectors identified a Green noncited violation for failure to comply with Technical Specification 5.4.1.a by failing to correctly implement a clearance order that resulted in the inoperability of the HPCS pump.

Description. On June 17, 2003, with the plant at 100 percent power, assignments were made to rack out SSW Pump SWP-P2C and hang Clearance RB-03-0862, Tag 1, on SWP-P2C. The breaker for HPCS Pump E22-PC001 was racked out instead of the breaker for SWP-P2C. This was a human performance error. Annunciators in the main control room alerted the control room team to the improper electrical equipment lineup for the HPCS system. The breaker for HPCS pump breaker was racked back into the switchgear and a breaker operability test was performed. The HPCS pump breaker was racked out a total of 16 minutes.

Analysis. The inspectors concluded that racking out the HPCS pump breaker was a failure to correctly implement Clearance RB-03-0862. This performance deficiency affected the mitigating systems cornerstone.

This self-revealing finding was more than minor because the HPCS safety function was made unavailable. The inspectors reviewed the finding using Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." Based on the results of the phase one screening of the finding, the inspectors conducted a safety significance determination. The inspectors determined that the finding was of very low safety significance (Green) because the HPCS pump was not functional for less than one hour. Recovery credit was given for operator actions necessary to restore the equipment lineup and recover the safety function.

The dominant accident sequences identified during the risk-informed SDP process were (1) transients with a loss of the power conversion system, (2) a stuck relief valve, (3) loss of offsite power, and (4) loss of normal service water.

Enforcement. Clearance RB-03-0862 was not properly implemented. That is a violation of Technical Specification 5.4.1.a. Technical Specification 5.4.1.a requires that written procedures be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Revision 2, Appendix A, Item 4.f. recommends procedures for equipment control (e.g., tagging). Administrative Procedure ADM-27, "Protective Tagging," Revision 20, step 7.7.1.3, required that tags be placed in the sequence shown on the clearance. Tag 1 of Clearance RB-03-0862 was incorrectly placed following opening of the HPCS pump breaker instead of the SSW Pump 2C breaker.

This violation is associated with an inspection finding that is characterized by the SDP as having very low risk significance (Green) and is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC enforcement policy. This violation is in the licensee's corrective action program as CR-RBS-2003-02437 (NCV 05000458/2003005-04).

4. The standby NSW pump failed to develop discharge pressure when started during a manual swap of running NSW pumps

a. Inspection Scope

The inspectors reviewed the circumstances surrounding the failure of NSW Pump C to develop discharge pressure when started during a manual swap of running NSW pumps on September 1, 2003. The inspectors interviewed engineering, maintenance, and operations personnel, walked down portions of the NSW system, reviewed system operating Procedure SOP-0018, “Normal Service Water,” Revision 32; MAI 352248 for the overhaul of NSW Pump C; and clearance Order RB-03-0584, which was used to perform the work.

b. Findings

Introduction. The inspectors identified an apparent self-revealing violation of Technical Specification 5.4.1.a, the significance of which has yet to be determined. A human performance error caused the isolation of the air release valve for NSW Pump C. As a result, NSW Pump C became airbound and failed to develop discharge pressure during a planned swap of running NSW pumps on September 1, 2003.

Description. In June 2003, NSW Pump C was removed from service for a planned overhaul of the pump, including impeller replacement. The machine work on the pump casing was performed at an approved vendor’s off-site machine shop. The pump was returned to the site and licensee mechanical maintenance technicians realigned and re-coupled the pump to its motor. On June 14, 2003, the pump was filled and vented and run successfully for postmaintenance testing. The pump remained in service for the next 16 days. On September 1, 2003, while swapping running NSW pumps, NSW Pump B was secured and NSW Pump C was started. NSW Pump C did not develop its expected discharge pressure when its motor-operated discharge valve came completely open. Running NSW Pump A indication showed that it was supplying all system flow. NSW Pump C was secured and NSW Pump B was restarted. System operating parameters returned to normal for two pump operation.

On September 2, 2003, engineering, maintenance, and operations personnel examined NSW Pump C in an effort to determine the reason for its failure to develop normal discharge pressure. NSW Pump C discharge air release valve isolation Valve SWP-V3312C was found closed. The air release valve for NSW Pump C served as a high point vent on the system while the pump was secured. As a result, NSW Pump C became air bound while in standby, and failed to develop discharge pressure when started the previous day. The licensee documented the improper valve lineup in CR-RBS-2003-03042. Later that day, the licensee successfully test ran NSW Pump C and swapped running pumps to NSW Pumps A and C in service with NSW pump B secured. Final NSW system parameters were normal for two pump operation.

Analysis. The inspectors determined that this human performance error was more than minor because it was associated with an increase in the likelihood of an initiating event. The inspectors reviewed this finding using IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations.” The result of the phase one screening process and the inspectors’ review of the increased likelihood of a loss of NSW was that further review was required to determine the overall risk significance of this event.

Enforcement. The inspectors determined that the failure to maintain NSW Pump C discharge air release valve isolation Valve SWP-V3312C open was an apparent violation of NSW SOP-0018, Attachment 1A, “Valve Lineup - Normal Service Water,” Revision 32. As such, the human performance error was a violation of Technical Specification 5.4.1.a. which requires that written procedures be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Appendix A, Item 4.q, covers procedures for the startup, operation, and shutdown of the service water system. This finding does not present an immediate safety concern because the system lineup has been corrected. Pending determination of its risk significance, the apparent violation is identified as URI 05000458/2003005-05.

4OA4 Crosscutting Aspects of Findings

Three performance deficiencies with human performance crosscutting aspects were identified in Section 4OA3 of this report. The three items included: a failure to properly align service water such that an SSW system automatic start occurred when shifting normal service water pumps; a failure to rack out the correct breaker when implementing a protective clearance tagout (operators racked out the HPCS pump break instead of an SSW pump breaker); and a failure to properly align the air release isolation valve on a normal service water pump that resulted in air binding of the pump.

4OA5 Other Activities

1. On-site Fabrication of Components and Construction of an Independent Spent Fuel Storage Installation (ISFSI) (60853)

a. Inspection Scope

On August 27, 2003, concrete placement was completed for the first of three sections of the concrete pad at the ISFSI. The pad was designed for storage of 40 spent fuel casks. The licensee plans to use the Holtec vertical cask system under a general license in accordance with the Holtec Certificate of Compliance #72-1014. Design parameters and seismic criteria for the construction of the pad provided in the Holtec Final Safety Analysis Report were reviewed against the construction specifications for the pad and the reactor’s Part 50 Final Safety Analysis Report. The construction specifications and the soil testing results for the backfill under the pad were also reviewed to verify the stability of the pad area. The actual pouring of the first section of the pad was observed, including observation of the quality control tests performed on the concrete. Qualifications of personnel performing the quality control tests were verified by reviewing their certifications. The documents reviewed are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

4OA6 Management Meetings

Exit Meetings

The inspectors presented the security inspection results to Mr. R. King, Director - Nuclear Safety Assurance, and other members of licensee management on August 14, 2003.

The inspectors presented the ALARA inspection results to Mr. P. Hinnenkamp, Vice President, Operations, and other members of licensee management at the conclusion of the inspection on August 22, 2003.

The inspectors presented the ISFSI inspection results for docket 72-049 to Mr. T. Hoffman, Senior Project Manager, and other members of licensee management on August 27, 2003.

The inspectors presented the integrated inspection results for docket 50-458 to Mr. P. Hinnenkamp, Vice President - Operations, and other members of licensee management on September 23, 2003.

The licensee acknowledged the information presented. The licensee indicated that none of the information provided to the inspectors was proprietary.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

B. Allen, Manager, Emergency Planning
M. Boyle, Superintendent, Radiation Protection
W. Brian, Director - Engineering
D. Burnett, Superintendent, Chemistry
C. Bush, Assistant Operations Manager
J. Fowler, Manager, Quality Programs
A. James, Superintendent - Plant Security
T. Gates, Manager, System Engineering
H. Goodman, Manager, Nuclear Engineering
R. Goodwin, Manager - Training and Development
J. Heckenberger, Manager, Planning and Scheduling/Outage
P. Hinnenkamp, Vice President - Operations
R. King, Director - Nuclear Safety Assurance
J. Leavines, Manager, Licensing
T. Lynch, Manager, Operations
J. Malara, Manager, Design Engineering
W. Mashburn, Manager, Programs and Components
J. McGhee, Manager, Plant Maintenance
T. Trepanier, General Manager - Plant Operations
W. Trudell, Manager, Corrective Actions

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000458/2003005-02	URI	RCS Leakage PI data collection may be less than adequate (Section 4OA1)
05000458/2003005-05	URI	NSW pump found to be air-bound when called upon to run (Section 4OA3)

Opened and Closed

05000458/2003005-01	FIN	Failure to maintain collective doses associated with RWP 2003-1800 ALARA (Section 2OS2)
05000458/2003005-03	NCV	Failure to follow procedure resulted in automatic initiating of SSW (Section 4OA3)

05000458/2003005-04	NCV	Procedure violations resulted in HPCS inoperability (Section 4OA3)

Closed

05000458/2003-002-01	LER	Secondary Containment Door Failure Due to Malfunction of Door Assist Device (Section 4OA3)
05000458/2003-006-01	LER	Automatic Initiation of Standby Service Water System Due to Inadequate Control of System Operation (Section 4OA3)
05000458/2003-007-00	LER	Automatic Initiation of SSW System Due to Inadequate Control of System Operation (Section 4OA3)

Discussed

None.

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: Flood Protection Measures

USAR Section 3.4.1, “Flood Protection”

RBS individual plant examination, section 3.3.6, “Internal Flooding Analysis,” Revision 0

Calculation G13.18.12.3*15-0, “Internal Flooding Screening Analysis,” Revision 0

Calculation G13.18.12.3*16-0, “Quantitative Analysis of Cases that Survived the Screening Analysis,” Revision 0

Calculation G13.18.12.3*13-0, “Miscellaneous Internal Flooding Calculations,” Revision 0

Calculation G13.2.3 PN-317-Addendum 0B, “Max Flood Elevations for Moderate Energy Line Cracks in Cat I Structures,” Revision 0

Section 2OS2: ALARA Planning and Controls

CR-RBS-2003-01016	CR-RBS-2003-01213	CR-RBS-2003-01506	CR-RBS-2003-02062
CR-RBS-2003-01179	CR-RBS-2003-01442	CR-RBS-2003-01691	CR-RBS-2003-02623

Section 3PP2: Access Control

Procedure PSP-4-101, “Administration (Document Control),” Revision 16
Procedure PSP-4-300, “Operations (Access Control),” Revision 22
Procedure STI-301-07-02, “X-Ray Equipment Image Test,” Revision 9
Procedure STI-301-07-04, “Metal Detector Operability Test,” Revision 11
Procedure STI-301-07-05, “Explosive Detector Operability Test,” Revision 13
Procedure STI-301-90-06, “Biometrics Hand Reader Quarterly Performance Test,” Revision 3
Procedure SPI-04, “Access Control Officer(s),” Revision 41
Procedure SPI-09, “Vehicle/Material Search,” Revision 28
Drill Report Records for the period July 1, 2002, through January 31, 2003

Section 4OA1: Performance Indicator Verification

Procedure LI-107, ”NRC Performance Indicator Process,” Revision 1
NEI 99-02, ”Regulatory Assessment Performance Indicator Guideline,” Revision 2
Alarm History Records for the period January 1 through July 15, 2003
Security Work Orders for the period August 1, 2002, through July 11, 2003

Section 4OA2: Identification and Resolution of Problems

CR-RBS-2002-00587	CR-RBS-2002-01692	CR-RBS-2002-01753	CR-RBS-2003-01049
CR-RBS-2002-01203	CR-RBS-2002-01695	CR-RBS-2003-00241	CR-RBS-2003-01291
CR-RBS-2002-01265	CR-RBS-2002-01752	CR-RBS-2003-00631	CR-RBS-2003-02352

Section 4OA5: Other Activities

- Holtec Report HI-2002444, “Final Safety Analysis Report,” Docket 72-1014, Revision 1
- RBS Updated Safety Analysis Report, Revision 15
- Quality Control Report VCD VA-1125.928-002-004A, “Quality Assurance/ Quality Control Report for the ISFSI Subgrade Project,” Revision 00
- Certificate Test Report 50001R03 for the tests performed on the reinforcing bars used in the ISFSI pad, dated August 25, 2003
- Engineering Request ER 00-0391, “Site Soil Preparations Required for Dry Cask Storage Pad,” Revision 0
- Engineering Request ER 00-0392-000, “ISFSI Site Development and Pad Installation,” Revision 0, and associated 50.59 review dated June 12, 2003
- Drawing KA-EY-090A, “ISFSI Storage Area,” Revision B
- Drawing KA-EC-090A, “ISFSI Slab Plan and General Notes,” Revision B

- E-mail from Mark Walton (Entergy) to Vincent Everett providing slump and temperature results of the Quality Assurance/Quality Control tests on the concrete, dated September 2, 2003

- Condition Report CR-RBS-2003-03218 concerning the 28-day break test results for the ISFSI pad concrete samples, dated September 24, 2003

- Condition Report CR-RBS-2003-02980 concerning fine aggregate sieve analysis, dated August 23, 2003

- Condition Report CR-RBS-2000-02023 concerning adherence to procedures related to soil testing by the contractor, dated November 22, 2000

- MAI 358547, “Dry Fuel Storage Slab,” printed July 18, 2003

- Nonconformance and Disposition Action Report NCR 1210B-01 concerning the hydrometer analysis required by ASTM D 422, Revision 2

- Nonconformance and Disposition Action Report NCR 00-1210B-02 concerning use of ASTM D 4253 and D 4254 for determining maximum and minimum drying densities, Revision 2

- Nonconformance and Disposition Action Report NCR 00-1210B-03 concerning the use of USBR 5755-89 for determining elastic modulus for sand, Revision 2

- Nonconformance and Disposition Action Report NCR 00-1210B-04 concerning the required number of field moisture and density tests per lift, Revision 0

- Nonconformance and Disposition Action Report NCR 00-1210B-05 concerning confirmatory samples related to grain size distribution in the backfill, Revision 0

- Nonconformance and Disposition Action Report NCR 00-1210B-06 concerning qualification testing of samples, Revision 0

- Nonconformance and Disposition Action Report NCR 00-1210B-07 concerning grain size analysis requirements of ASTM D 422, Revision 0

- Nonconformance and Disposition Action Report NCR 00-1210B-08 concerning erosion problems on the north slope of the backfill after a 5" rainfall on January 16, 2001, Revision 0

- Entergy Procedure QV-111, Attachment 9.1 “Certification Form,” and Certification Statements for the personnel performing the Quality Assurance/Quality Control tests during the concrete placement of the first pad

- Soil Testing Engineers, Inc., “Report of Siting and Geotechnical Evaluation of the Dry Cask Storage Facility at the River Bend Station,” dated September 25, 2000

- Compilation of test data results for the field density tests for each lift of the ISFSI pad backfill, no date given

LIST OF ACRONYMS

ADHRS	alternate decay heat removal system
ALARA	As Low As Reasonably Achievable
CCP	reactor plant component cooling water
CCS	turbine plant component cooling water
CFR	*Code of Federal Regulations*
CR-RBS	Condition Report-River Bend Station
DAD	door assist device
FIN	finding
HPCS	high pressure core spray
ISFSI	Independent Spent Fuel Storage Installation
LER	licensee event report
MAI	maintenance action item
NCV	noncited violation
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
NSW	normal service water
PI	performance indicators
RBS	River Bend Station
RCIC	reactor core isolation cooling
RCS	reactor coolant system
RHR	residual heat removal
RPS	reactor protection system
RWP	radiation work permit
SDP	Significance Determination Process
SOP	system operating procedure
SSC	structures, systems, or components
SSW	standby service water
STP	surveillance test procedure
TCV	turbine control valve
TRM	Technical Requirements Manual
URI	unresolved item
USAR	Updated Safety Analysis Report