UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

June 12, 2000

Randal K. Edington, Vice President - Operations
River Bend Station
Entergy Operations, Inc.
P.O. Box 220
St. Francisville, Louisiana 70775

SUBJECT: CORRECTION TO NRC REPORT INSPECTION REPORT NO. 50-458/00-09

Dear Mr. Edington:

NRC Inspection Report 50-458/00-09 was issued on May 18, 2000, with an error in the actual decrease in reactor cavity level. Since this error could lead to confusion, we are issuing a corrected page. Please replace page 4 of the "Report Details" section with the revised page 4 included with this letter. We regret any inconvenience this may have caused.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

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

Sincerely,

/RA/

William D. Johnson, Chief
Project Branch B
Division of Reactor Projects

Docket No.: 50-458
License No.: NPF-47

Enclosure:
As stated

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

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

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

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

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

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

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

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

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

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

Ronald Wascom, Administrator
and State Liaison Officer
Department of Environmental Quality
P.O. Box 82135
Baton Rouge, Louisiana 70884-2135

Electronic distribution from ADAMS by RIV:
Regional Administrator **(EWM)**
DRP Director **(KEB)**
DRS Drector **(ATH)**
Senior Resident Inspector **(TWP)**
Branch Chief, DRP/B **(WDJ)**
Senior Project Engineer, DRP/B **(RAK1)**
Branch Chief, DRP/TSS **(LAY)**
RITS Coordinator **(NBH)**

Only inspection reports to the following:
D. Lange **(DJL)**
NRR Event Tracking System **(IPAS)**
RBS Site Secretary **(PJS)**

R:_RB\RB2000-09RP-COR.wpd

RIV:SRI	C:DRP/B			
TWPruett;df	WDJohnson			
E - WDJohnson	***/RA/***			
6/12/00	6/12/00			

OFFICIAL RECORD COPY T=Telephone E=E-mail F=Fax

Valve E12-F064A opened in less than 8 seconds even though the minimum flow of 1100 gpm had been reached. Because the flow was above the setpoint, Valve E12-F064A received an immediate signal to close during the opening stroke. As a result, when Valve E12-F064A reached the full open position, it immediately attempted to close. The sudden reversal of the voltage applied to the valve motor resulted in a current value which exceeded the normal inrush current and caused the breaker for Valve E12-F064A to trip open. With Valve E12-F064A failed open, a flow path existed which resulted in a loss of approximately 3 inches (6,000 gallons) of water inventory from the reactor cavity to the suppression pool.

Following the event, the licensee initiated Condition Report (CR) 1999-0784. The licensee determined the root cause to be that the original design was inadequate in that the RHR minimum flow valve 8 second time delay was not an adequate time to establish flow before the minimum flow valve opened. A contributing cause was determined to be an inadequate review associated with a change in operations practices. Specifically, changes in the operational philosophy regarding the use of human performance tools resulted in more deliberate operation of equipment and the changes were not assessed in relation to specific time sensitive plant evolutions.

On June 26, 1999, the following corrective actions were developed and approved by the licensee:

- Increase the minimum flow valve 8 second time delay to 30 seconds, per Engineering Request (ER) 99-0349, to provide additional time for operations personnel to increase RHR flow,
- Install a 1 second time delay in the electrical circuit for Valve E12-F064A, per ER 99-0450, to prevent sudden motor reversal and a subsequent breaker trip,
- Revise operations standards and expectations to ensure prejob briefs included time sensitive actions, and
- Provide training to operations personnel on breaker trips due to sudden motor reversal.

On March 19 and 20, 2000, Valve E12-MOVF064A again opened (when it should have remained closed) while starting RHR Pump A in the shutdown cooling mode of operation. Valve E12-F064A received an automatic signal to close during each event; however, the valve breaker did not trip even though there was a sudden reversal of voltage. Since Valve E12-F064A closed, the loss of inventory from the reactor vessel to the suppression pool only lasted a few seconds. Consequently, there was not a notable decrease in reactor vessel level. Engineering personnel stated that the sudden reversal in voltage resulted in the potential for a trip of the breaker for Valve E12-F064A and that the breaker could have tripped during the event. Had the breaker for Valve E12-F064A tripped, the loss in inventory from the vessel would have continued until an operator closed Valve E12-F064A, as was the case during the April 30, 1999, event.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

May 18, 2000

Randal K. Edington, Vice President - Operations
River Bend Station
Entergy Operations, Inc.
P.O. Box 220
St. Francisville, Louisiana 70775

SUBJECT: NRC INSPECTION REPORT NO. 50-458/00-09

Dear Mr. Edington:

This refers to the inspections conducted on April 2 through May 6, 2000, at the River Bend Station facility. The enclosed report presents the results of these inspections.

Based on the results of the inspections, the NRC has determined that two Severity Level IV violations of NRC requirements occurred. These violations are being treated as noncited violations (NCVs), consistent with Section VI.A. of the Enforcement Policy. These NCVs are described in the subject inspection report. If you contest these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U. S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U. S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011; the Director, Office of Enforcement, U. S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the River Bend Station facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response, if requested, will be placed in the NRC Public Document Room.

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

Sincerely,

/RA/

William D. Johnson, Chief
Project Branch B
Division of Reactor Projects

Docket No.: 50-458
License No.: NPF-47

Enclosure:
NRC Inspection Report No.
50-458/00-09

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

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

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

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

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

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

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

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

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

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

Ronald Wascom, Administrator
and State Liaison Officer
Department of Environmental Quality
P.O. Box 82135
Baton Rouge, Louisiana 70884-2135

Electronic distribution from ADAMS by RIV:
Regional Administrator **(EWM)**
DRP Director **(KEB)**
DRS Director **(ATH)**
Senior Resident Inspector **(TWP)**
Branch Chief, DRP/B **(WDJ)**
Senior Project Engineer, DRP/B **(RAK1)**
Branch Chief, DRP/TSS **(LAY)**
RITS Coordinator **(NBH)**

<u>Only inspection reports</u> to the following:
D. Lange **(DJL)**
NRR Event Tracking System **(IPAS)**
RBS Site Secretary **(PJS)**
Wayne Scott **(WES)**

DOCUMENT NAME: R:_RB\RB2000-09RP-TWP.wpd
To receive copy of document, indicate in box: "C" = Copy without enclosure **"E"** = Copy with enclosure **"N"** = No copy

RIV:RI:DRP		SRI:DRP/B		SPE:DRP/B		HP:DRS/PS		C:DRS/PSB	
SMSchneider		TWPruett		RAKopriva		JSDodson		GMGood	
5/9/00 ***/RA/WDJ***		5/9/00 ***/RA/WDJ***		5/18/00 ***/RA/***		5/18/00 ***/RA/***		5/18/00 ***/RA/***	

C:DRP/B									
WDJohnson									
5/18/00 ***/RA/***									

OFFICIAL RECORD COPY

ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-458
License No.:	NPF-47
Report No.:	50-458/00-09
Licensee:	Entergy Operations, Inc.
Facility:	River Bend Station
Location:	5485 U.S. Highway 61 St. Francisville, Louisiana
Dates:	April 2 through May 6, 2000
Inspectors:	T. W. Pruett, Senior Resident Inspector S. M. Schneider, Resident Inspector R. A. Kopriva, Senior Project Engineer J. S. Dodson, Health Physicist
Approved By:	William D. Johnson, Chief, Project Branch B Division of Reactor Projects

ATTACHMENT: Supplemental Information

SUMMARY OF FINDINGS

River Bend Nuclear Station
NRC Inspection Report 50-458/00-09

The report covers a 5-week period of resident inspection and an announced inspection by a regional radiation specialist. The significance of issues is indicated by their color (green, white, yellow, or red) and was determined by the Significance Determination Process in Inspection Manual Chapter 0609.

Cornerstone: Mitigating Systems

- Green. The inspectors identified a noncited violation for the failure to implement corrective actions as required by Criterion XVI of Appendix B to 10 CFR Part 50. The licensee did not implement corrective actions following a previous occurrence to preclude opening of residual heat removal minimum flow Valve E12-F064A and subsequent loss of approximately 50 gallons of reactor vessel inventory while aligning the residual heat removal system to the shutdown cooling mode of operation. The risk significance of this issue was low because redundant methods of inventory injection were either operating or available. This item was entered in the licensee's corrective action program as Condition Report 2000-0947 (Section 1R12).

- Green. The inspectors identified a noncited violation for three examples of the failure to follow procedures required by Technical Specification 5.4.1.a. Maintenance and engineering personnel did not adequately perform a zone inspection of the drywell as required by Maintenance Action Item 329427, "Drywell Zone Inspection - All Levels." Specifically, the inspectors identified a significant amount of debris during a drywell closeout inspection which had not been identified during the licensee's zone inspection or during a management closeout tour. Maintenance and engineering personnel did not adequately perform a coatings inspection of the drywell as required by Maintenance Action Item 333068, "Drywell Coating Inspection." Specifically, the inspectors identified 400 to 500 square feet of degraded coatings during a drywell closeout inspection which had not been identified during the licensee's coatings inspection or during a management closeout tour. The risk significance of the drywell issues was low because the emergency core cooling system suction strainers would not have been adversely affected. Operations and engineering personnel did not complete a control rod drop accident analysis as required by Procedure STP-500-0705, "Rod Sequence Verification When Rod Pattern Control System Is Bypassed." Specifically, operations personnel withdrew Control Rod 44-13 beyond the banked position withdrawal sequence restraints without having completed a control rod drop accident analysis. The risk significance of this issue was low because the licensee subsequently determined that the plant remained within the boundaries of the control rod drop accident analysis. These items were entered in the licensee's corrective action program as Condition Reports 2000-0911, 2000-0904, and 2000-0941 (Sections 1R20.1, 1R20.2, and 1R22).

- Green. The inspectors identified one function described in the Technical Specification Bases that had not been included in the maintenance rule scope. The function of the residual heat removal minimum flow valves, as described in the bases for Technical Specification 3.3.5.1, "Emergency Core Cooling System Instrumentation," was not included in the list of functions included in the maintenance rule scope for the residual heat removal system. Consequently, maintenance rule functional failures associated with residual heat removal minimum flow Valve E12-F064A opening when aligning the residual heat removal system to the shut down cooling mode of operation were not identified by engineering personnel. The risk significance of this issue was low because the improper characterization of the failure of Valve E12-F064A did not significantly impact implementation of the maintenance rule for the residual heat removal system (Section 1R12).

- Green. The inspectors determined that training personnel demonstrated poor performance in identifying configuration differences between the simulator and main control room. The inspectors identified three simulator fidelity issues during a walkdown of selected panels in the simulator which involved an out-of-service reboiler vent valve, an out-of-service suppression pool temperature indication, and an elevated containment temperature indication. Additionally, the licensee identified four deficiencies during a subsequent audit which involved a feedwater heater controller, a deenergized regenerative evaporator supply shut-off valve, an average power range monitor, and suppression pool cooling Pump 1B. The risk significance of this issue was low because the deficiencies would not have significantly impacted the effectiveness of simulator training (Section 1R11.2).

- Green. The licensee determined that instrument and controls technicians inadvertently caused an engineered safety features isolation of the reactor core isolation cooling system. During the restoration of the reactor core isolation cooling system following the steam supply pressure low channel functional test, an inadvertent engineered safety features actuation resulted in the isolation of the reactor core isolation cooling system. The subsequent investigation of the event by engineering personnel determined that instrument and controls personnel inadvertently contacted an adjacent terminal which caused an engineered safety features actuation of the reactor core isolation cooling system. The risk significance of the issue was low because additional injection systems were operable (Section 1R22).

Cornerstone: Barrier Integrity

- Green. The inspectors identified that engineering personnel did not characterize a failure of reactor core isolation cooling system warmup Valve E51-F076, a containment isolation valve, as a maintenance rule functional failure. The licensee's maintenance rule functional failure review of the failure of Valve E51-F076 to close only considered the affect on reactor core isolation cooling system operation and did not evaluate the affect on the containment isolation function. The risk significance of this issue was low because the improper characterization of the failure of Valve E51-F076 did not significantly impact implementation of the maintenance rule for the reactor core isolation cooling system (Section 1R12).

Report Details

Summary of Plant Status: On April 8, 2000, River Bend completed Refueling Outage 9. On April 14, 2000, the plant achieved 100 percent power. The facility operated at essentially 100 percent power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignment

a. Inspection Scope

The inspectors performed a partial equipment alignment check on the reactor core isolation cooling (RCIC) system.

b. Issues and Findings

There were no findings identified and documented during this inspection.

1R05 Fire Protection

a. Inspection Scope

The inspectors toured the RCIC system pump room, the Division I and Division II standby service water system pump rooms, and the containment building to assess the control of transient combustible material, operational effectiveness of fire protection equipment, and the material condition of fire barriers.

b. Issues and Findings

There were no findings identified and documented during this inspection.

1R06 Flood Protection Measures

a. Inspection Scope

The inspectors verified that the licensee's flooding mitigation plans and equipment were consistent with the licensee's design requirements and the risk analysis assumptions. The areas inspected were Flood Zones AB-114-FL5 and AB-114-FL6 (auxiliary building, 114 foot elevation, Flood Zones 5 and 6). These areas were inspected due to their susceptibility to internal flooding as identified in the Updated Safety Analysis Report, the River Bend Individual Plant Evaluation, and flooding Calculation G13.18.12.3-15-0, "Internal Flooding Screening Analysis."

b. Issues and Findings

There were no findings identified and documented during this inspection.

1R11 Licensed Operator Requalification Program

.1 Operator Performance in the Simulator Control Room

a. Inspection Scope

The inspectors observed the activities of an operating crew in the simulator during an emergency response drill.

b. Issues and Findings

There were no findings identified and documented during this inspection.

.2 Simulator and Control Room Configuration

a. Inspection Scope

The inspectors compared simulator board configurations with actual control room board configurations.

b. Issues and Findings

The inspectors identified three simulator configurations which were different from the main control room. Subsequent to the inspectors' walkdown, the licensee identified an additional four items which needed to be incorporated into the simulator model.

On April 18, 2000, the inspectors observed the activities of an operating crew in the simulator during an emergency response drill. Approximately 10 minutes before the start of the drill, the inspectors conducted a partial walkdown of the simulator panels to determine if there were any fidelity issues. The inspectors identified three configurations/indications which were different from the actual main control room conditions. Specifically, the indication for an out-of-service reboiler vent valve in the main control room was shown energized in the simulator, an out-of-service suppression pool temperature indication in the main control room read normal in the simulator, and a containment temperature indication in the main control room was approximately 10 degrees above the temperature indication in the simulator.

As a result of the inspector identified differences, the operations training supervisor directed training personnel to conduct an audit of the simulator and control room configurations. Four additional differences were identified during this audit which needed to be incorporated into the simulator model. These items included: a feedwater heater controller with an operations hold tag that was in manual, a deenergized regenerative evaporator supply shut-off valve, an average power range monitor which failed to come out of the set-up mode, and suppression pool cooling Pump 1B being in service.

The inspectors discussed the additional fidelity issues with the operations training supervisor who stated that: (1) a program was in place to perform periodic comparisons of the simulator versus the actual control room, (2) training personnel needed to improve their performance in the identification of simulator issues during the periodic comparisons with the main control room, (3) there was a plant/simulator difference database which was used to document and resolve any differences between the simulator and the actual control room, and (4) the specific deficiencies identified by the inspectors before the emergency drill and by training personnel during the audit would be entered into the plant/simulator database.

The inspectors determined that the three deficiencies identified during the small sampling conducted just before the emergency drill scenario and the four additional deficiencies identified by training personnel during the subsequent audit indicated poor performance on the part of training personnel in identifying configuration differences between the simulator and main control room.

The inspectors reviewed the simulator and main control room configuration issues and determined the differences would not have significantly impacted the effectiveness of simulator training. Therefore, this issue was determined to be within the licensee's response band (green).

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors selected three performance problems associated with residual heat removal (RHR) system minimum flow Valve E12-F064A, RCIC system warmup Valve E51-F076, and control room chiller service water Valve SWP-100 and evaluated the effectiveness of the licensee's corrective actions and maintenance rule determinations.

b. Issues and Findings

RHR Minimum Flow Valve E12-F064A Corrective Actions

The inspectors identified a noncited violation for the failure to implement effective corrective actions to prevent the opening of RHR minimum flow Valve E12-F064A and subsequent loss of reactor vessel inventory while placing the RHR system in the shutdown cooling mode of operation.

Valve E12-F064A was designed to open when RHR flow was less than 1100 gpm for more than 8 seconds. As specified in the bases for Technical Specification 3.3.5.1, "Emergency Core Cooling System Instrumentation," the RHR minimum flow valves are time delayed such that the valves will not open for approximately 8 seconds after the flow switches detect low flow. The time delay is provided to limit reactor vessel inventory loss while placing the RHR system in the shutdown cooling mode of operation.

On April 30, 1999, with the reactor flooded to approximately 23 feet in the reactor cavity, Valve E12-F064A opened (when it should have remained closed) while starting RHR Pump A in the shutdown cooling mode of operation. Specifically, Valve E12-F064A opened in less than 8 seconds even though the minimum flow of 1100 gpm had been reached. Because the flow was above the setpoint, Valve E12-F064A received an immediate signal to close during the opening stroke. As a result, when Valve E12-F064A reached the full open position, it immediately attempted to close. The sudden reversal of the voltage applied to the valve motor resulted in a current value which exceeded the normal inrush current and caused the breaker for Valve E12-F064A to trip open. With Valve E12-F064A failed open, a flow path existed which resulted in a loss of approximately 6 feet of water inventory from the reactor cavity to the suppression pool.

Following the event, the licensee initiated Condition Report (CR) 1999-0784. The licensee determined the root cause to be that the original design was inadequate in that the RHR minimum flow valve 8 second time delay was not an adequate time to establish flow before the minimum flow valve opened. A contributing cause was determined to be an inadequate review associated with a change in operations practices. Specifically, changes in the operational philosophy regarding the use of human performance tools resulted in more deliberate operation of equipment and the changes were not assessed in relation to specific time sensitive plant evolutions.

On June 26, 1999, the following corrective actions were developed and approved by the licensee:

- Increase the minimum flow valve 8 second time delay to 30 seconds, per Engineering Request (ER) 99-0349, to provide additional time for operations personnel to increase RHR flow,
- Install a 1 second time delay in the electrical circuit for Valve E12-F064A, per ER 99-0450, to prevent sudden motor reversal and a subsequent breaker trip,
- Revise operations standards and expectations to ensure prejob briefs included time sensitive actions, and
- Provide training to operations personnel on breaker trips due to sudden motor reversal.

On March 19 and 20, 2000, Valve E12-MOVF064A again opened (when it should have remained closed) while starting RHR Pump A in the shutdown cooling mode of operation. Valve E12-F064A received an automatic signal to close during each event; however, the valve breaker did not trip even though there was a sudden reversal of voltage. Since Valve E12-F064A closed, the loss of inventory from the reactor vessel to the suppression pool only lasted a few seconds. Consequently, there was not a notable decrease in reactor vessel level. Engineering personnel stated that the sudden reversal in voltage resulted in the potential for a trip of the breaker for Valve E12-F064A and that the breaker could have tripped during the event. Had the breaker for Valve E12-F064A tripped, the loss in inventory from the vessel would have continued until an operator closed Valve E12-F064A, as was the case during the April 30, 1999, event.

Following the March 19 and 20, 2000, events, the inspectors reviewed the status of the corrective actions for CR 1999-0784 and determined the following:

- Increasing the minimum flow valve 8 second time delay to 30 seconds to provide operations personnel with additional time to raise RHR flow was never initiated as a specific corrective action in CR 1999-0784. Consequently, no action was taken to develop or perform ER 99-0349.

- The design for installing a 1 second time delay in the electrical circuit for Valve E12-F064A to prevent sudden motor reversal and a subsequent breaker trip was completed on October 27, 1999. The work for the design change was released on February 29, 2000. However, ER 99-0450 was not placed on the schedule to be worked until after the inspectors questioned the status of the modification. ER 99-0450 is currently scheduled to be worked during the weeks of May 15, 2000, for Division I and June 26, 2000, for Division II.

- Revising operations standards and expectations to ensure prejob briefs included time sensitive actions was completed on January 12, 2000.

- Providing training to operations personnel on breaker trips due to sudden motor reversal was completed on September 28, 1999.

A senior reactor analyst evaluated this event using the Significance Determination Process and found that the risk significance of the event was minimal because redundant methods of inventory injection were either operating or available. The inspectors concluded that the safety significance of this issue was very low (green).

Criterion XVI of Appendix B to 10 CFR Part 50 requires, in part, that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition. The failure to implement corrective actions to preclude opening of Valve E12-F064A and a subsequent loss of reactor vessel inventory while aligning the RHR system to the shutdown cooling mode of operation is a violation of Criterion XVI of Appendix B to 10 CFR Part 50, which is being treated as a noncited violation (NCV 50-458/0009-01). The issue was entered into the licensee's corrective action system as CR 2000-0947.

RHR Minimum Flow Valve E12-F064A Maintenance Rule Implementation

The inspectors assessed the licensee’s maintenance rule functional failure review of the opening of Valve E12-F064A. The maintenance rule function for Valve E12-F064A was to provide for pump protection by a minimum flow line and valve which opens when the RHR pump is running and RHR flow is less than required and then auto closes when RHR flow is sufficient. For the occasion where the breaker for Valve E12-F064A tripped open and the valve failed to close, the licensee viewed the event as a maintenance rule functional failure. The licensee determined that a maintenance preventable functional failure had not occurred because the root cause determined the failure to be related to the design of the valve and not the maintenance of the valve. The inspectors agreed with the licensee's maintenance rule determination.

The bases for Technical Specification 3.3.5.1 specified that the minimum flow valves are time delayed such that the valves will not open for approximately 8 seconds after the switches detect low flow. The time delay is provided to limit reactor vessel inventory loss during the startup of the RHR system in the shutdown cooling mode. The inspectors determined that the function described in the Technical Specifications Bases was not included within the scope of maintenance rule functions listed for the RHR system. Consequently, the licensee incorrectly determined that, on occasions where Valve E12-F064A opened and reclosed, a maintenance rule functional failure did not exist and therefore, a maintenance preventable functional failure did not exist. Following discussions with the inspectors, the licensee stated that the function for the RHR minimum flow valve remaining closed to prevent a reduction in reactor vessel inventory would be added to the RHR system list of maintenance rule functions.

The inspectors determined that the licensee had not scoped all of the functions of the RHR system into the maintenance rule. However, not scoping the function of Valve E12-F064A to remain closed did not significantly impact implementation of the maintenance rule for the RHR system. Therefore, this issue was determined to be within the licensee's response band (green).

RCIC Valve E51-F076

The inspectors determined that engineering personnel had incorrectly concluded that the failure of RCIC system warmup Valve E51-F076, a containment isolation valve, was not a maintenance rule functional failure.

The inspectors reviewed CR 1999-1098 during an assessment of system event failure determinations associated with the RCIC system. CR 1999-1098 described a failure of Valve E51-F076 to isolate on a Division II low pressure isolation signal during surveillance testing. The licensee had determined that the failure of Valve E51-F076 was due to a defective starter reversing contactor closing coil.

Engineering personnel determined that the failure of Valve E51-F076 to isolate was not a maintenance rule functional failure of the RCIC system because Valve E51-F076 was only opened when the RCIC system was being warmed up before placing the system into service. Engineering personnel also determined that a maintenance rule functional failure of the switchgear system had not occurred because there was not a maintenance rule functional failure of the RCIC system.

On April 13, 2000, the inspectors questioned engineering personnel to determine why the failure of containment isolation Valve E51-F076 to close on an engineered safety features actuation signal was not considered a functional failure. In response, engineering personnel stated that an incorrect maintenance rule functional failure determination was made in that the determination did not consider switchgear Function F-303-004, which specified that the switchgear system provided 480VAC power to Division II standby load centers and motor control center loads. Additionally, engineering personnel did not consider RCIC system Function F-209-015, which specified that the RCIC system provided an isolation function on low reactor pressure vessel pressure.

On May 3, 2000, engineering personnel determined that the failure of Valve E51-F076 was a maintenance rule functional failure, but not a maintenance preventable functional failure in that the contactor failure was due to a random component failure. The inspectors determined that the licensee did not properly characterize the failure of Valve E51-F076 as a maintenance rule functional failure. However, not properly characterizing the failure of Valve E51-F076 did not significantly impact implementation of the maintenance rule for the RCIC system. Therefore, this issue was determined to be within the licensee's response band (green).

Control Room Chiller Service Water Valve SWP-100

There were no findings identified and documented during this inspection.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors evaluated the effectiveness of risk assessments performed by the licensee for work weeks beginning on April 9, 16, and 30, 2000.

b. Issues and Findings

There were no findings identified and documented during this inspection.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed two operability evaluations associated with emergency core cooling system strainer performance due to the inspectors' identification of debris in the drywell and unqualified coatings in the drywell.

b. Issues and Findings

There were no findings identified and documented during this inspection. See Section 1R20 for additional details concerning the closeout inspections of the drywell.

1R19 Postmaintenance Testing

a. Inspection Scope

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

MAI 323585, "Replace Fuel Booster Pump (Division I EDG),"
MAI 333777, "Replace Discharge Header Check Valve (Division I EDG),"
STP-309-6306, "Division III Emergency Diesel Generator Air Start Valve Operability."

b. Issues and Findings

There were no findings identified and documented during this inspection.

1R20 Refueling and Outage Activities

.1 Drywell Debris

a. Inspection Scope

The inspectors toured the drywell following the management closeout tour to identify debris which could impact emergency core cooling system strainer performance.

b. Issues and Findings

The inspectors identified a noncited violation for the inadequate completion of a maintenance repetitive task associated with drywell zone inspections. Specifically, a significant number of drywell deficiencies were identified by the inspectors following the licensee's performance of a drywell zone inspection and a management closeout tour.

On March 29, 2000, the inspectors toured the drywell in preparation for drywell closeout. The inspectors identified a significant amount of debris and other deficiencies on the 141 foot elevation (the first level inspected) and secured their inspection. The inspectors discussed the condition of the 141 foot elevation with the drywell coordinator who stated that he felt this level was ready for closeout. The inspectors then discussed the condition of the 141 foot elevation with the drywell closeout manager who stated that his expectations were higher than that of the drywell coordinator and that he had not yet completed the drywell management closeout tour.

On April 3, 2000, the drywell closeout manager informed the inspectors that the management closeout tour of the drywell was complete and that the inspectors could tour the drywell. The inspectors toured the drywell and identified numerous deficiencies on each level. These deficiencies included: tools left in the drywell, a cardboard sign taped to the drywell wall, a 3-inch by 8-inch piece of wood, a loose electronic alarming device, and a significant amount of general debris such as loose tape, nails, plastic tie wraps, paper identification tags, and rope. Material deficiencies included a loose electrical conduit connection, a missing fastener from a pipe hanger, an inoperable exit sign which was hanging by its wires, and an electrical cable wrapped with material with instructions that specified the wrapping should be removed before system operation. As a result of these inspector identified deficiencies, the licensee repeated the drywell zone inspection and removed approximately 3 cubic feet of debris from the drywell.

On April 4, 2000, the inspectors reentered the drywell and found the drywell condition for debris satisfactory.

Maintenance Action Item (MAI) 329427, "Drywell Zone Inspection - All Levels," was initiated to perform a zone inspection of the drywell during Refueling Outage 9 and to document any deficiencies and corrective actions implemented as a result of the inspection. The inspectors determined that the licensee did not adequately perform the drywell zone inspection as required by repetitive task MAI 329427. The inspectors also determined that the initial management tour of the drywell failed to identify the significant number of deficiencies existing in the drywell before the NRC inspection.

The inspectors compared the amount of debris removed from the drywell with the amount of material that would be required to adversely affect the emergency core cooling system suction strainers. The inspectors determined that the strainers would not have been impacted and that the safety functions of the emergency core cooling systems would not have been impaired. Therefore, this issue did not meet the initial Significance Determination Process screening and is considered to be green.

Technical Specification 5.4.1.a requires, in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Section 9 of Appendix A of Regulatory Guide 1.33 requires the licensee to have procedures for performing maintenance. Repetitive task MAI 329427 required that a zone inspection of the drywell be performed to identify, document, and correct deficiencies. The failure to adequately perform the drywell zone inspection repetitive task is a violation of Technical Specification 5.4.1.a, which is being treated as a noncited violation (50-458/0009-02). This violation is in the licensee’s corrective action program as CR 2000-0911.

.2 Drywell Coatings

a. Inspection Scope

The inspectors toured the drywell following the management closeout tour to identify any degraded coatings which could impact emergency core cooling system strainer performance.

b. Issues and Findings

The inspectors identified a second example of a noncited violation of Technical Specification 5.4.1.a for the inadequate performance of repetitive task MAI 333068, "Drywell Coating Inspection." Specifically, the inspectors identified several hundred square feet of degraded drywell coatings (peeling and blistering paint) following the licensee performance of a drywell coatings inspection and a management closeout tour.

Repetitive task MAI 333068 was initiated to perform a drywell coatings inspection for evidence of peeling, chipping, rusting, or mechanical damage in the drywell during Refueling Outage 9. Repetitive task MAI 333068 required the licensee to inspect coatings in the drywell and document any deficiencies found and the corrective actions taken. Repetitive task MAI 333068 also required notifying the design engineering supervisor civil/structural before the start of the repetitive task to determine if the coatings engineer would participate in the task. The inspectors determined that the coatings engineer did not participate in the performance of the drywell coatings inspection during Refueling Outages 8 or 9.

On April 3, 2000, the drywell closeout manager informed the inspectors that the management closeout tour of the drywell was complete and that the inspectors could tour the drywell. The inspectors toured each level of the drywell and identified several areas with degraded coatings (areas of peeling and blistering paint). The inspectors discussed the issue with the supervisor civil/structural design engineering who informed the inspectors that MAI 333068 had already been performed and would have documented the condition of the coatings in the drywell. The inspectors reviewed the results of MAI 333068 and determined that most of the areas identified by the inspectors had not been identified during the drywell coatings inspection.

On April 4, 2000, the inspectors reentered the drywell and identified the areas of degraded coatings to the drywell closeout manager. Following additional discussions with the inspectors, the licensee initiated Condition Report 2000-0904 and repeated the drywell coatings inspections with the coatings engineer present. The licensee identified several areas within the drywell which required recoating, including an area of approximately 400 to 500 square feet on the floor of the drywell. Engineering personnel determined that the recoating of these areas could be deferred to the next refueling outage.

The inspectors compared the amount of unqualified coatings in the drywell with the amount of unqualified coatings that would be required to adversely affect the emergency core cooling system suction strainers and determined that the strainers would not have been impacted. Additionally, the inspectors aggregated the effect of the debris in the drywell with the unqualified coatings and determined that the strainers would not be adversely affected and that the safety functions of the emergency core cooling systems would not be impaired. Therefore, this issue did not meet the initial Significance Determination Process screening and is considered to be green.

Technical Specification 5.4.1.a requires, in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Section 9 of Appendix A of Regulatory Guide 1.33 requires the licensee to have procedures for performing maintenance. Repetitive task MAI 333068 required that a coatings inspection of the drywell be performed to identify, document, and correct deficiencies. The failure to adequately perform the drywell coatings inspection repetitive task is a second example of a violation of Technical Specification 5.4.1.a. This additional example was entered in the licensee's corrective action program as CR 2000-0904.

.3 Reactor Startup

a. Inspection Scope

The inspectors observed heatup and startup activities, verified that Technical Specification conditions were met before changing modes, and ensured containment integrity was established.

b. Issues and Findings

There were no findings identified and documented during this inspection.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors reviewed the below listed surveillance tests to verify that systems were capable of performing their intended safety functions and to ensure that requirements for Technical Specifications, the Updated Safety Analysis Report, and procedures were met:

STP-207-4539, "RCIC Isolation - RCIC Steam Supply Pressure - Low Channel Functional Test,"

STP-052-3701, "Control Rod Scram Testing,"

STP-203-6305, "Division III Quarterly High Pressure Core Spray Pump and Valve Surveillance."

b. Issues and Findings

RCIC Testing

On April 12, 2000, during the restoration of the RCIC system low pressure test, an inadvertent engineered safety features actuation resulted in isolation of the RCIC system. The inspectors observed that the instrument and controls technicians had followed the procedure as written and determined that the procedure was appropriate for the knowledge, skills, and abilities of the instrument and controls technicians involved.

Engineering personnel reviewed the event and determined that the most probable cause was that instrument and controls technicians inadvertently contacted an adjacent terminal which resulted in an engineered safety features actuation of the RCIC system.

The inspectors observed that the instrument and controls technicians were careful during the testing activity and had placed electrical tape over exposed adjacent terminals. Nevertheless, it was plausible that the instrument and controls technicians may have inadvertently contacted an adjacent terminal. Even though an inadvertent engineered safety features actuation occurred, the inspectors determined that there was no violation of procedural requirements.

The licensee made the appropriate notifications following the event and planned to submit a licensee event report. The inspectors determined that the risk significance of the event was low because additional injection systems were operable. Therefore, this issue was determined to be in the licensee's response band (green).

Control Rod Scram Testing

A third example of a noncited violation of Technical Specification 5.4.1.a was identified during the performance of control rod scram testing. Specifically, operations and reactor engineering personnel failed to implement a caution statement in a procedure which required a control rod drop accident analysis for the control rod sequence used during scram time testing of Control Rod 44-13.

On April 8, 2000, operations and reactor engineering personnel performed Procedure STP-052-3701, "Control Rod Scram Testing." The plant was in Mode 1 at approximately 10 percent thermal power. Control Rod 44-13, a Group 9 control rod, was positioned at step 00, and was the next control rod to be withdrawn for scram time testing.

The rod control and information system limits control rod movements that enforce the banked position withdrawal sequence restrictions below the 27.5 percent low power setpoint. The approved banked position withdrawal sequence positions for any Group 9 control rod were steps 00, 02, or 04 during the performance of Procedure STP-052-3701. Withdrawing a Group 9 control rod beyond step 04 would be in violation of the banked position withdrawal sequence requirements.

A note in Section 7.3 of Procedure STP-052-3701 specified that bypassing the control rod in the rod action control system may be required if reactor power was less than the lower power setpoint. Additionally, the note referred operations personnel to Procedure STP-500-0705, "Rod Sequence Verification When Rod Pattern Control System Is Bypassed." A caution statement for Section 7.3 of Procedure STP-500-0705 specified that a special control rod drop accident analysis was required if thermal power was less than the low power setpoint and bypassed control rods are to be moved outside the restraints of the banked position withdrawal sequence.

On April 8, 2000, operations and reactor engineering personnel bypassed Control Rod 44-13 in the rod action control system, withdrew Control Rod 44-13 to step 48, and then scrammed Control Rod 44-13 to step 00. The test personnel involved failed to implement the caution statement in Procedure STP-500-0705, which required a special control rod drop accident analysis when thermal power was less than the low power setpoint and the rod to be tested was moved outside of its banked position withdrawal sequence restraints. Additionally, the withdrawal of Control Rod 44-13 beyond the banked position withdrawal restraints resulted in the unplanned entry into Technical Specifications 3.10.7, "Control Rod Testing - Operating," and 3.1.6, "Control Rod Pattern."

Technical Specification 3.10.7 specified, in part, that the control rod pattern limiting condition for operation constraints discussed in Technical Specification 3.1.6 may be suspended and control rods bypassed in the rod action control system to allow performance of control rod scram time testing as long as conformance to the approved control rod sequence is verified. If the conditions of the limiting condition for operation are not met, then the test and exception to Technical Specification 3.1.6 must be suspended immediately.

Technical Specification 3.1.6 specified that operable control rods shall comply with the requirements of the banked position withdrawal sequence in Modes 1 and 2 when less than 20 percent thermal power. With one or more control rods not in compliance with the banked position withdrawal sequence, operations personnel must move the control rod to the correct position or declare the control rod inoperable within 8 hours.

The inspectors determined that Control Rod 44-13 was outside of the banked position withdrawal sequence restraints for approximately 6 minutes before being scrammed to step 00. Therefore, the completion times associated with Technical Specifications 3.1.6 and 3.10.7 were met. Following the unplanned entry into the Technical Specifications, the licensee requested that General Electric conduct a control rod drop accident analysis for the plant conditions during the performance of Procedure STP-052-3701. General Electric determined that the licensee remained within the boundaries of the control rod drop accident analysis during scram time testing on April 8, 2000. Therefore, the inspectors determined that this issue did not meet the initial Significance Determination Process screening and is considered to be green.

Technical Specification 5.4.1.a requires, in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Section 8 of Appendix A of Regulatory Guide 1.33 requires the licensee to have procedures for surveillance tests including control rod operability and scram time tests. Section 7.3 of Procedure STP-052-3701, “Control Rod Scram Testing,” specified that control rods may require bypassing in the rod action control system in accordance with Procedure STP-500-0705, “Rod Sequence Verification When Rod Pattern Control System Is Bypassed,” when reactor power was below the low power setpoint. Section 7.3 of Procedure STP-500-0705 required a special control rod drop accident analysis when thermal power was below the low power setpoint for control rods that were to be moved outside the restraints of the banked position withdrawal sequence. The failure of operations and reactor engineering personnel to obtain a control rod drop accident analysis as required by Procedure STP-500-0705 before withdrawing Control Rod 44-13 beyond the banked position withdrawal sequence restraints is a third example of a violation of Technical Specification 5.4.1.a. This item was entered in the licensee’s corrective action program as CR 2000-0941.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS3 Radiological Monitoring Instrumentation

a. Inspection Scope

The inspector interviewed licensee personnel and reviewed the following items:

- Calibration, operability, and alarm setpoints, when applicable, of portable radiation detection instrumentation, temporary area radiation monitors, continuous air monitors, whole body counting instrumentation, and personnel contamination monitors

- Calibration and source response check documentation for radiation detection instruments staged for use, whole body counting instrumentation, and personnel contamination monitors

- Radiation protection technician instrument selection and self-verification of instrument operability prior to use

- The status and surveillance records of self-contained breathing apparatuses staged and ready for use in the plant

- The licensee's capability for refilling and transporting self-contained breathing apparatus air bottles to and from the control room and operations support center during emergency conditions

- Control room operator and emergency response personnel training and qualifications for use of self-contained breathing apparatuses

- Licensee self-assessments and audits, focusing on radiological incidents that involved personnel internal exposures

- Selected exposure significant radiological incidents that involved radiation monitoring instrument deficiencies since the last inspection in this area

- Licensee self-assessments and audits, focusing on radiological incidents that involved personnel internal exposures

- Selected exposure significant radiological incidents that involved radiation monitoring instrument deficiencies since the last inspection in this area

b. Issues and Findings

There were no significant findings identified and documented during this inspection.

4. OTHER ACTIVITIES

4OA6 Management Meetings

.1 Exit Meeting Summaries

The health physicist inspector presented the inspection results to Mr. Rick King and other members of licensee management on April 20, 2000. The licensee acknowledged the findings presented.

The resident inspectors presented the inspection results to Mr. Rick King and other members of licensee management on May 9, 2000. The licensee acknowledged the findings presented.

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

ATTACHMENT

PARTIAL LIST OF PERSONS CONTACTED

Licensee

E. Bush, Superintendent Operations
M. Cantrell, Supervisor, Operations Training
R. Edington, Vice President-Operations
H. Goodman, Superintendent, Reactor Engineering
D. Heath, Supervisor, Health Physics Shift
T. Hildebrandt, Manager, Maintenance
H. Holmes, Specialist, Health Physics / Chemistry
J. Holmes, Manager, Radiation Protection and Chemistry
V. Huffstatler, Supervisor, Health Physics Shift
R. King, Director, Nuclear Safety and Regulatory Affairs
M. Laiche, Master Technician, Radiation Protection
J. McGhee, Manager, Operations
C. Miller, Superintendent, Composite Team
D. Mims, General Manager, Plant Operations
D. Myers, Senior Licensing Specialist, Nuclear Safety Assurance
D. Pace, Director, Engineering
J. Reeves, Specialist, Health Physics
A. Shahkarami, Manager System Engineering
D. Wells, Superintendent, Radiation Protection

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

50-458/0009-01	NCV	Failure to implement corrective actions to prevent recurrence of inadvertent opening of residual heat removal minimum flow valve (Section 1R12)
50-458/0009-02	NCV	Three examples of a failure to follow procedures involving debris in the drywell, unqualified coatings in the drywell, and scram time testing (Section 1R20)

LIST OF ACRONYMS AND INITIALISMS USED

CFR	Code of Federal Regulations
CR	condition report
EDG	emergency diesel generator
ER	engineering request
NCV	noncited violation
RCIC	reactor core isolation cooling
RHR	residual heat removal

LIST OF DOCUMENTS REVIEWED

Calculations

G13.18.12.3*13, Revision 0, Miscellaneous Internal Flooding Calculations

G13.18.2.0*35, Revision 1, Auxiliary Building Flooding Level 141, 15,000 gallons Service Water

G13.2.3 (PN-314), Revision 0, Maximum Flood Elevations for Moderate Energy Line Cracks in Category I Structures

Calibration and Instrument Response Packages

Contamination and portal monitor calibration data packages

Contamination monitor, portal monitor, and portable survey instrument response test documentation packages

Calibration data packages, calibration verification and background documentation for the Accuscan II and Fastscan whole body counters

Calibration Criteria Sheet for Eberline PCM and TCM
Calibration Criteria Sheet for Eberline Personnel Monitor PM7
Calibration Criteria Sheet for the Merlin Gerin Tool Monitor (CPO)
Calibration Criteria Sheet for Portable Radiological Instruments
Calibration Criteria Sheet for Portable Air Samplers
Calibration Criteria Sheet for Eberline Model AMS-3 Continuous Air Monitor

Condition Reports

CR 1999-0602	Valve E12-F064B Opened After Minimum Flow Through Pump was Reached
CR 1999-0605	Valve E12-F064A Opened Earlier Than Expected
CR 1999-0842	Valve E12-F064B Failed to Open Within 8 Seconds
CR 1999-0860	Electrical Transient Caused Initiation of Division I Diesel Generator, Low Pressure Core Spray Injection, and Residual Heat Removal A Injection
CR 1999-0863	Valve E12-F064A Failed to Open
CR 1999-0966	Valve E12-F064A Did Not Open
CR 1999-1242	Valve E12-F064C Closed
CR 1999-1701	RHR A Flow Meter and RHR Minimum Flow Valve Trip Unit Reading Downscale
CR 1999-1963	Repetitive Failure of Valve SWP-100
CR 2000-0704	Valve E12-F064A Stroked Open and Closed
CR 2000-0856	Valve E12-F064A Opened After Minimum Flow Through Pump was Reached

Condition Reports initiated between April 2 and May 6, 2000

List of Condition Reports involving radiation monitoring instruments (1/1/99 - 4/11/2000)

Lesson Plans

RBS-1-LP-GET-00300	Radiological Respiratory Training Lesson Plan, Revision 12/2/99
RBS-1-LP-GET-00301	Radiological Respiratory Training Lesson Plan, Revision 12/2/99
RBS-1-LP-GET-00302	Radiological Respiratory Training Lesson Plan, Revision 12/2/99
RBS-1-LP-GET-00303	Radiological Respiratory Training Lesson Plan, Revision 12/2/99
RBS-1-SIM-STG-400007.00	Simulator Instructor Guide Module 7 2000

Maintenance Rule

Listing of fixed area and postaccident monitors included in the maintenance rule program

Maintenance rule functions database for the reactor core isolation cooling system, residual heat removal system, and service water system

Plant Procedures

ADM-0018	Plant Housekeeping, Revision 11
ADM-0080	Post Maintenance Testing, Revision 2
EP-00-148	Site Drill No. 00-02
FPP-0095	Fire Extinguisher Inspection and Maintenance, Revision 6
GOP-0001	Plant Startup, Revision 31
REP-0052	Startup Reactivity Controls, Revision 3
RSP-0202	Radiation Protection Instrument Program, Revision 08
RHP-0105	Operation of the Canberra Accuscan II and Fastscan Whole Body Counters, Revision 02
RPP-0111	Operation of the Eberline ASP-1 with Attached NRD Detector, Revision 1A
RPP-0113	Operation and Calibration of the Eberline ACM-100A, Revision 07
SOP-0035	Reactor Core Isolation Cooling, Revision 20
STP-050-3601	Shutdown Margin Demonstration, Revision 18
STP-309-0201	Division I Diesel Generator Operability Test, Revision 21

Equipment Out Of Service Schedules

Fire Hazards Analysis

Operations, Electrical, Mechanical, I&C, Radiation Protection, Outside, Fire Protection, and Maintenance Support Weekly Schedules

River Bend Online Maintenance Guidelines

Quality Assurance Audits and Assessments

Radiation Protection Program Assessment/Audit, March 2-6, 1998
Radiation Protection Program Assessment/Audit, January 17-21, 2000
Quality Assurance Audit Report 99-03-1-CHEM, March 1-22, 1999
Quality Assurance Audit Report 99-06-I-REMP/ENV/EFF, June 7-11, 1999
Quality Assurance Audit Report 99-07-I-FEPL, July 26 through September 23, 1999