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

April 23, 2001

EA-01-090

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

SUBJECT: RIVER BEND STATION--NRC INTEGRATED INSPECTION REPORT 50-458/00-16

Dear Mr. Edington:

On March 31, 2001, the NRC completed inspections at your River Bend Station facility. The enclosed integrated inspection report presents the results of these inspections which were discussed with you and other members of your staff on March 30, 2001.

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.

Based on the results of the inspection, the inspectors identified five findings of very low safety significance (Green). Four of these findings were determined to involve violations of NRC requirements. However, because of their very low safety significance and because they have been entered into your corrective action program, the NRC is treating these findings as noncited violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny these noncited violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, 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 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: 50-458
License: NPF-47

Enclosure:
NRC Inspection Report
50-458/00-16

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-458
License:	NPF-47
Report No.:	50-458/00-16
Licensee:	Entergy Operations, Inc.
Facility:	River Bend Station
Location:	5485 U.S. Highway 61 St. Francisville, Louisiana
Dates:	December 24, 2000, through March 31, 2001
Inspectors:	T. W Pruett, Senior Resident Inspector S. M. Schneider, Resident Inspector R. V. Azua, Project Engineer, Branch B C. J. Paulk, Senior Reactor Inspector J. W. Whittemore, Senior Reactor Inspector A. B. Earnest, Senior Physical Security Specialist M. P. Shannon, Senior Health Physicist J. S. Dodson, Health Physicist
Approved By:	W. D. Johnson, Chief, Project Branch B
ATTACHMENT:	Supplemental Information

SUMMARY OF FINDINGS

River Bend Station
NRC Inspection Report 50-458/00-16

IR 05000458-00-16; on 12/24/00-03/31/2001; Entergy Operations, Inc; River Bend Station. Integrated Resident & Regional Report. Adverse Weather Protection, Equipment Alignment, Fire Protection, Maintenance Rule Implementation, and Public Radiation Safety.

The inspections were conducted by resident, reactor, physical security, and radiation protection inspectors. The inspections identified five Green findings, four of which were noncited violations. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using IMC 0609, “Significance Determination Process” (SDP). Findings for which the SDP does not apply are indicated by “No Color” or by the severity level of the applicable violation. The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html.

A. Inspector Identified Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified deficiencies with implementation of cold weather requirements. Specifically, no attempt was made to provide heating to the water treatment room, several room heater switch settings were not in accordance with the respective loop calibration report, and repetitive maintenance tasks did not exist to ensure that room heaters in the fire pump building, the normal switchgear room, or the auxiliary control room were functioning properly.

  The inspectors determined that the safety significance of not implementing cold weather requirements for reduced room temperatures was very low in that actual temperatures in these areas during the time the condition existed did not go low enough to affect the qualification of the equipment located in these areas (Section 1R01).

- Green. The licensee failed to maintain the required inventory of chemicals onsite to support operation of the alternate standby liquid control system. Specifically, the licensee failed to maintain 2500 pounds of sodium borate and 2500 pounds of boric acid onsite for alternate standby liquid control injection as required by the emergency operating procedure.

  The inspectors determined that the safety significance of not maintaining alternate standby liquid control chemicals available was very low in that the standby liquid control system was only determined to be unavailable for a maximum of 12 days over the year during tank sparging evolutions. The failure to maintain adequate chemical inventory in the main warehouse/storeroom for alternate standby liquid control injection is a noncited violation of Technical Specification 5.4.1.a. This violation is in the licensee’s corrective action program as Condition Report 2000-1680 (Section 1R04.7).

- Green. The licensee did not complete annual walkdowns of emergency operating procedure enclosures between November 1996 and October 2000.

  The inspectors determined that the safety significance of not completing annual walkdowns of emergency operating procedure enclosures was very low in that, other than missing alternate standby liquid control chemicals, no significant equipment issues were identified when the enclosures were walked down. Additionally, no actual plant problems occurred which would have required implementation of these enclosures. The failure to perform yearly walkdowns of each emergency operating procedure enclosure, as required by Procedure OSP-0009, is a noncited violation of Criterion V of Appendix B to 10 CFR Part 50. This violation is in the licensee's corrective action program as Condition Report 2000-1723 (Section 1R04.7).

- Green. The licensee did not maintain a 3-hour rated fire barrier between two fire areas which contained redundant safe shutdown equipment. Specifically, the inspectors identified an 11.5-inch deep hole in a 12-inch concrete fire barrier between the D-Tunnel and the D-Tunnel cable chase fire areas.

  The inspectors determined that the safety significance of the degraded fire barrier was very low due to the remaining mitigating detection and suppression systems, the fire brigade response, and the low initiating frequency. The failure to maintain a 3-hour rated fire barrier between Fire Areas AB-7 and -18, is a noncited violation of Attachment 4 to Facility Operating License 50-458. This violation is in the licensee's corrective action program as Condition Report 2000-1944 (Section 1R05.1).

- Green. The licensee did not monitor the performance of standby service water station blackout Valve SWP-AOV599 against established goals in a manner sufficient to assure the valve was capable of supplying cooling water to the Division III emergency diesel generator during a station blackout event.

  The inspectors determined that the safety significance of the failure to monitor the station blackout valve was very low due to the high likelihood of success of operator recovery actions. The failure to monitor the performance of Valve SWP-AOV599 is a noncited violation of 10 CFR 50.65(a)(1). This violation (EA-01-090) is in the licensee's corrective action program as Condition Report 1999-0263 (Section 1R12.5).

B. Licensee Identified Violations

Violations of very low safety significance were identified by the licensee and reviewed by the inspectors. Corrective actions taken by the licensee appeared reasonable. These violations are listed in Section 4OA7 of this report.

Report Details

Summary of Plant Status: On December 31, 2000, the first point heater normal level control Valve HDH-LV6A failed closed. The licensee initially reduced power to 97 percent due to lowering feedwater temperature. On January 1, 2001, the licensee lowered reactor power to 58 percent in order to repair Valve HDH-LV6A. On January 3, 2001, the licensee commenced a power ascension following the repairs to Valve HDH-LV6A, scram time testing, and a rod sequence exchange. On January 4, 2001, during the power ascension, level decreased in the first point heater while placing the drain receiver tank into service due to a failure of the first point heater high level dump Valve HDH-LV26A. On January 5, 2001, the licensee reduced reactor power from 87 to 70 percent to support troubleshooting and repair of Valve HDH-LV26A. On January 6, 2001, the licensee recommenced power ascension following the repairs to Valve HDH-LV26A. On January 7, 2001, the licensee achieved 100 percent power.

On February 9, 2001, the licensee reduced power to 64 percent to plug main condenser tubes. On February 11, 2001, the licensee completed repair to the main condenser tubes and commenced a power ascension. On February 12, 2001, the licensee achieved 100 percent power.

On February 28, 2001, the licensee reduced power to 60 percent to repack main steam line steam shutoff Valve B21-F098A. On March 1, 2001, the licensee achieved 100 percent power. The facility operated at essentially 100 percent power throughout the remainder of the inspection period.

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

1R01 Adverse Weather (71111.01)

a. Inspection Scope

The inspectors assessed the licensee's implementation of Procedure OSP-0043, "Freeze Protection and Temperature Maintenance," requirements during periods of cold weather.

b. Findings

The inspectors identified deficiencies of very low safety significance (Green) associated with implementation of cold weather requirements.

Section 4.1.5 of Procedure OSP-0043 specified, in part, to perform cold weather rounds once per shift if the outside temperature is less than or equal to 25°F. If any local area/room temperature is expected to fall below the minimum allowable temperature limit, then contact engineering to request an assessment of that room/area and provide temporary heating to that area. The inspectors determined that the cold weather rounds were implemented when the outside temperature was less than 25°F. However, several deficiencies were identified involving the licensee's response to areas which were determined to be below the minimum allowable temperature limit. Specifically:

- No attempt was made to provide heating to the water treatment room. Following discussions with operations personnel, the inspectors determined that the equipment in the area was no longer utilized. Nevertheless, the licensee acknowledged that operations personnel should have provided heating to the area or revised the procedure to reflect that cold weather monitoring of the water treatment room was no longer required.

- The inspectors determined that several room heater switch settings were not in accordance with the respective loop calibration report. Following discussions with the licensee, the inspectors determined that operations personnel were not aware that room heater setpoints existed. Although the setpoints were incorrect, the inspectors determined that the temperature in the affected spaces did not impact the qualification of components.

- The inspectors determined that repetitive maintenance tasks did not exist to ensure that room heaters in the fire pump building, the normal switchgear room, or the auxiliary control room were functioning properly. The licensee initiated actions to ensure all of the heaters were adjusted to the correct setpoint and initiated a review to determine if the heaters needed to be added to the preventive maintenance program.

The inspectors determined that the failure to ensure that room heater setpoints were in accordance with the loop calibration report and the failure to provide maintenance tasks to ensure the room heaters were functioning properly had a credible impact on safety and could have affected the availability or reliability of equipment important to safety located in the fire pump building, the normal switchgear room, and the auxiliary control room. The safety significance of this finding was evaluated using the significance determination process which screened the finding as Green. Since the actual room temperatures did not fall below any levels that would have affected the qualification of this equipment during the time the condition existed, the safety significance of not implementing cold weather requirements for reduced room temperatures was determined to be very low.

1R04 Equipment Alignment (71111.04)

.1 Alignment Check of the Division III Emergency Diesel Generator

a. Inspection Scope

The inspectors completed a partial walkdown of the Division III emergency diesel generator following testing on January 31, 2001, to verify that the system was properly configured and to identify any discrepancies that might impact the function of the system and thereby potentially increase risk. The inspectors reviewed Procedure SOP-0052, “HPCS Diesel Generator,” during the assessment.

b. Findings

No findings of significance were identified.

.2 Alignment Check of the Reactor Protection System

a. Inspection Scope

The inspectors completed a partial walkdown of the reactor protection system on February 1, 2001, to verify that the system was properly configured and to identify any discrepancies that might impact the function of the system and thereby potentially increase risk. The inspectors reviewed Procedure SOP-0079, “Reactor Protection System,” during the assessment.

b. Findings

No findings of significance were identified.

.3 Alignment Check of the Division I Emergency Diesel Generator

a. Inspection Scope

The inspectors completed a partial walkdown of the Division I emergency diesel generator while the Division II emergency diesel generator was out of service to verify that the system was properly configured and to identify any discrepancies that might impact the function of the system, and thereby, potentially increase risk. The inspectors reviewed Procedure SOP-0053, “Standby Diesel Generator and Auxiliaries,” during the assessment.

b. Findings

No findings of significance were identified.

.4 Alignment Check of Residual Heat Removal System

a. Inspection Scope

The inspectors completed a partial walkdown of Residual Heat Removal System C on February 23, 2001, to verify that the system was properly configured and to identify any discrepancies that might impact the function of the system, thereby potentially increasing risk. The inspectors reviewed Procedure SOP-0031, “Residual Heat Removal System,” during the assessment.

b. Findings

No findings of significance were identified

.5 Alignment Check of Station Blackout Diesel Generator

a. Inspection Scope

The inspectors completed a partial walkdown of the station blackout diesel generator on February 15, 2001, to verify that the system was properly configured and to identify any discrepancies that might impact the function of the system, thereby potentially increasing risk. The inspectors reviewed Procedures SOP-0054, “Station Blackout Diesel Generator,” and AOP-0051, “Loss of Decay Heat Removal,” during the assessment.

b. Findings

No findings of significance were identified.

.6 (Closed) Violation 50-458/9915-01: failure of operating personnel to be aware of plant indications. This issue involved the repeated failure of licensee personnel to implement corrective actions to minimize the frequency at which operations personnel were not aware of suspect indications. The inspectors verified the implementation of corrective actions specified in the licensee’s response to the Notice of Violation dated April 18, 2000, and in Condition Report (CR) 1999-1448.

The corrective actions involved, in part, an assessment of operations department performance, development of a watchstanding practices improvement plan, completing additional management observations of operations activities, and added emphasis on management expectations for operations department watchstanders. The inspectors also observed that watchstanders were typically aware of the reasons and corrective actions for suspect indications when questioned by inspectors during routine walkdowns of control room panels. The inspectors determined that the corrective actions taken and planned adequately resolved the concerns associated with the violation.

.7 (Closed) Unresolved Item 50-458/0014-01: safety-significance determination for the storage of chemicals for alternate standby liquid control (SLC) injection and walkdowns of equipment needed to implement emergency operating procedure enclosures.

The inspectors identified two noncited violations which involved the failure to maintain chemicals for alternate SLC injection and emergency operating procedure walkdowns. The issues were determined to be of very low safety significance (Green).

On September 14, 2000, the licensee discovered plastic material in the SLC tank during the SLC system monthly sample. The licensee declared both subsystems of SLC inoperable and removed the pieces of plastic from the tank. The licensee determined that any remaining material that could be introduced into the SLC pump suction piping was less than 1 square inch and would not preclude the system from injecting sodium pentaborate into the reactor.

The licensee initially determined that the incremental risk associated with the SLC system being out of service for 1 year was 3.0E-8. The only time period the licensee could not postulate the system’s behavior was during 10-minute air sparges before the monthly chemistry sample of the SLC tank. The licensee concluded that the SLC system may not have been able to perform its intended safety function during the sparge evolution.

On September 25, 2000, the inspectors conducted a walkdown with Procedure EOP-0005, Enclosure 15, “Alternate SLC Injection and SLC TK GAL to LB Conversion,” and identified that the sodium borate and boric acid chemicals were not available in the warehouse or onsite for alternate SLC use. CR 2000-1680 was generated to document this condition.

On October 2, 2000, in response to the inspectors' observation of the missing chemicals, operations personnel completed a review of their documentation of emergency operating procedure enclosure audits and yearly walkdowns required by Procedure OSP-0009, "Authors Guide/Control and Use of Emergency Operating and Severe Accident Procedures." The review determined that the yearly walkdowns of each emergency operating procedure enclosure had not been performed since November 26, 1996. The licensee initiated CR 2000-1723 and Licensee Event Report (LER) 50-458/0013 to document this condition.

On November, 29, 2000, the licensee issued a letter to the NRC documenting supplemental information related to the SLC foreign material issue. The licensee had completed a more detailed evaluation of the condition in the SLC tank during operation of the SLC system with the plastic materials present in the tank. The licensee could not predict the response of the system during the monthly sparging evolutions; however, assuming the tank was out of service for the entire day following sparging (for a total of 12 days of inoperability over the year), the licensee determined the incremental risk was 9.9 E-10.

The inspectors determined that the unavailability of the SLC system and the alternate SLC system had an actual impact on safety and affected the safety function of a mitigating system. The inspectors completed a Phase One Significance Determination Process (SDP) which identified the need to perform a Phase Two evaluation because the finding represented an actual loss of safety function of a system. The Phase Two SDP review indicated that the loss of both the normal and alternate SLC functions, with no remaining mitigation capability, could have substantial safety significance. The licensee's probabilistic safety analysis determined that the loss of the SLC functions would be of very low safety significance.

An NRC senior reactor analyst (SRA) reviewed the risk significance of the SLC system being inoperable for a period of up to 12 days per year (based on the unpredictability of the system response with plastic material present during the monthly sparging and sampling of the SLC tank). The SRA utilized the draft River Bend Station SDP plant specific anticipated transient without scram (ATWS) work sheet and the guidance in NRC Manual Chapter 0609, "Significance Determination Process," dated April 21, 2000, including Table 1 - Estimated Likelihood for Initiating Event Occurrence During Degraded Period. In addition, the SRA considered the initiating event frequencies in NUREG/CR-5750, "Rates of Initiating Events at U.S. Nuclear Power Plants: 1987-1995, the licensee's probabilistic safety assessment calculation file, G13.18.12.3*135-1, "ATWS Event Tree for Level 1 PRA," dated April 20, 1999, and their letter "Supplemental Information to SLC Foreign Material Unresolved Item River Bend Station-Unit 1," dated November 29, 2000.

The SDP Phase Two assessment using the draft ATWS worksheet identified that there was a cutset involving an ATWS and then loss of all mitigation capability (common cause failure of the SLC system). Using the initiating event frequency established in Table 1 of Manual Chapter 0609 and the period of greater than 3 days and less than 30 days, the condition appeared to be of low to moderate safety significance. The SRA then considered the actual maximum 12 days the SLC system could have been unavailable as well as the revised initiating event frequency in NUREG/CR-5750. This revised analysis concluded that the condition was of very low safety significance. This finding of very low safety significance was supported using the risk achievement worth developed using the licensee's plant specific model, the core damage frequency, and the maximum 12-day period. Therefore, the NRC staff concluded that the unavailability of the SLC system for a maximum of 12 days in a year resulted in a finding of very low safety significance (Green).

Technical Specification 5.4.1.a requires, in part, that written procedures be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Section 6.n of Appendix A requires the licensee to have procedures for combating emergencies and other significant events for conditions requiring the use of SLC. Enclosure 15 of Procedure EOP-0005, specified, in part, that approximately 2500 pounds each of sodium borate and boric acid would be available in the main warehouse/storeroom for transit to the auxiliary building in the event alternate SLC injection was required. The inspectors determined that the failure to maintain adequate chemical inventory in the main warehouse/storeroom for alternate SLC injection was a violation of Technical Specification 5.4.1.a (NCV 50-458/0016-01). This violation is associated with an inspection finding that is characterized by the SDP as having very low safety 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 2000-1680.

The inspectors determined that the failure to perform yearly walkdowns of each emergency operating procedure enclosure had a credible impact on safety and could have affected the function of mitigating systems addressed in these procedures. The licensee's failure to perform the walkdowns failed to provide the opportunity to identify degraded, nonfunctioning, or missing equipment necessary for the performance of these emergency operating procedures. The safety significance of not completing these annual walkdowns was evaluated using the SDP which screened the violation as Green. The safety significance was determined to be very low in that no additional significant equipment issues were identified when the enclosures were walked down and because no actual plant problems occurred during this time which would have required the implementation of these enclosures.

Criterion V of Appendix B to 10 CFR Part 50 required, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. The inspectors determined that the failure to perform yearly walkdowns of each emergency operating procedure enclosure, as required by Procedure OSP-0009, was a violation of Criterion V of Appendix B to 10 CFR Part 50 (NCV 50-458/0016-02). This violation is associated with an inspection finding that is characterized by the SDP as having very low safety 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 2000-1723.

1R05 Fire Protection (71111.05)

.1 Tours of Plant Areas

a. Inspection Scope

The inspectors toured the D-Tunnel and D-Tunnel cable chase fire areas, residual heat removal pump Room A, the control room ventilation room, the spent fuel pool area, Auxiliary Building Elevation 170, the Division III switchgear room, the diesel-driven fire Pump 1A room, the diesel-driven fire Pump 1B room, and the motor-driven fire pump room to assess the control of transient combustible material and ignition sources, operational effectiveness of fire protection equipment, and the material condition of fire barriers. The following procedures were reviewed during the assessment:

- FPP-0030, "Storage of Combustibles"
- FPP-0050, “Handling of Flammable Liquids and Gases"
- FPP-0040, “Control of Transient Combustibles”
- Fire strategies for the associated areas
- Updated Safety Analysis Report fire hazards analysis
- River Bend postfire safe shutdown analysis
- Calculation G13.18.12.2-132, “Evaluate Fire Wall Separating Fire Areas AB-7 and AB-18"
- STP-000-3602, “Fire Barrier Visual Inspection”
- Calculation G13.18.12.2-133, “Fire Protection SDP for Void Found in Wall Separating Fire Area AB-7 and AB-18"

b. Findings

The inspectors identified a noncited violation of Attachment 4 to Facility Operating License 50-458, which involved a degraded fire barrier between Fire Areas AB-7 (D-Tunnel) and AB-18 (D-Tunnel cable chase). These areas contained redundant equipment for safe shutdown. The issue was determined to be of very low safety significance (Green).

On November 6, 2000, the inspectors identified a 3.5-inch wide by 1.5-inch high by 11.5-inch deep void in a 12-inch deep interior wall in the D-Tunnel. The licensee inspected the affected wall, determined the fire barrier was inoperable, and initiated CR 2000-1944. The licensee also concluded that the condition was likely in existence since facility construction. The void was subsequently repaired and the fire barrier returned to an operable condition.

The licensee conducted periodic fire barrier inspections as required by Technical Requirement 3.7.9.6, "Fire-rated Assemblies," Surveillance Requirement 3.7.9.6.6, every 18 months. Procedure STP-000-3602 provided guidance on conducting floor, wall, and ceiling inspections, including checking that fire barriers are free of damage or defects such as cracks, separations, gouges, holes, or openings. The most recent inspection of fire areas (AB-7 and AB-18) was conducted in September and October of 1999 with no identified deficiencies.

On December 5, 2000, the licensee completed Calculation G13.18.12.2-132 to evaluate the degraded fire barrier. The calculation concluded that the concrete wall separating Fire Areas AB-7 and AB-18 was adequate to withstand the hazards associated with each fire area with the identified void in the wall. The calculation also identified that the 3-hour rating for the fire barrier separating Fire Areas AB-7 and AB-18 was questionable.

On January 8, 2001, the licensee performed an assessment of the issue and determined that the condition of the 3-hour rated fire barrier with the identified void would be characterized as very low safety significance.

The inspectors determined that the hole in the fire barrier had a credible impact on safety and involved degradation of a fire protection feature. Since the finding involved the degradation of a fire barrier which separated cables supplying both Divisions I and II safe shutdown equipment, the inspectors conducted an Inspection Manual Chapter 0609, “Significance Determination Process,” Appendix F, review for findings related to fire barrier or suppression features. The Appendix F, Phase One SDP identified the need to perform a Phase Two SDP because the finding represented an impairment or degradation of a fire barrier and because the 3-hour rated fire barrier separated redundant safe shutdown components. The Phase Two SDP identified that the issue should be characterized as very low safety significance (Green) due to the remaining detection and suppression equipment, fire brigade response, and the low initiating event frequency.

Attachment 4 to Facility Operating License 50-458, specified that the licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility and as approved in the Safety Evaluation Report. Sections 9A.2.5.1.8 and 9A.2.5.1.14 of the Updated Safety Analysis Report (USAR) fire hazards analysis address fire protection requirements for Fire Areas AB-7 and AB-18. Both of these sections required a 3-hour rated reinforced concrete fire barrier separating the fire area from adjacent areas. The inspectors determined that the failure to maintain a 3-hour rated fire barrier between Fire Areas AB-7 and AB-18, was a violation of Attachment 4 to Facility Operating License 50-458 (NCV 50-458/0016-03). This violation is associated with an inspection finding that is characterized by the SDP as having very low safety 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 2000-1944.

.2 Fire Brigade Drill

a. Inspection Scope

The inspectors observed a fire brigade drill in order to assess the effectiveness of corrective actions for a previous unsatisfactory drill.

b. Findings

No findings of significance were identified.

.3 (Closed) Violation 50-458/0011-03: failure to complete monthly inspections of portable fire extinguishers. The issue involved the failure of the licensee to implement corrective actions for a minor violation involving fire extinguisher inspections.

The inspectors verified the implementation of corrective actions specified in the licensee’s response to the Notice of Violation dated September 21, 2000, and in CRs 2000-0969 and -1459. The corrective actions involved, in part, inspection of the affected fire extinguishers, a review of other fire protection equipment to ensure the required testing was completed, and a review of CRs initiated in response to noncited violations to ensure corrective actions restored the original noncompliance in a timely manner. The inspectors determined that the corrective actions taken and planned adequately resolved the concerns associated with the violation.

1R07 Heat Sink Performance (71111.07)

Inspection Scope

The inspectors interviewed licensee maintenance and engineering personnel and reviewed selected documents related to the inspection, maintenance, cleaning, and testing of safety-related heat exchangers and heat sinks. The inspectors also observed heat exchanger testing. The purpose of this effort was to identify any possibility of heat sink performance problems with a potential to increase risk at the River Bend Station. A secondary purpose of this effort was to identify any conditions or deficiencies which could mask degraded performance of safety-related heat exchangers or heat sinks. The heat sinks and heat exchangers chosen for review were:

- Standby service water system cooling tower
- Residual heat removal system

.1 Performance of Testing, Maintenance, and Inspection Activities

a. Inspection Scope

The inspector held discussions with system and process engineers, and then reviewed the licensee's test methodology and procedures for the selected heat sinks and heat exchangers. The inspector then reviewed testing requirements for the standby service water pumps and fans as well as mandatory inspection requirements and criteria for tower fill and tower basin debris. In addition, the inspector reviewed requirements for the standby service water system chemistry analysis, chemistry treatment, testing for biological contamination, and makeup water purity. Finally, the inspector reviewed results of testing and inspections. Through these results, the inspector determined that the currently required periodic performance of inspection, maintenance, and testing provided assurance that the cooling tower remained operable and would perform its safety function.

The inspector reviewed the design basis of the residual heat removal system and compared the design requirement for heat removal performance of Residual Heat Removal Division I or II with the design capacity of the heat exchangers in a division. The inspector then reviewed the procedures for and the results of required performance testing of the residual heat removal system heat exchangers. Specifically, the inspector verified that the test procedures established appropriate test conditions, required the use of the proper measuring and test instrumentation that adequately accounted for test instrument inaccuracies, and resulted in a valid test of heat exchanger thermal performance. The inspector then verified that test results were trended, the causes of degrading trends were identified, and that necessary action in the form of chemical cleaning was taken when performance limits were approached. The inspector determined that the performance of the residual heat removal heat exchangers was appropriately monitored and maintained to assure their safety function would be performed. The inspector also verified that the nonsafety-related service water system that normally supplied the heat exchangers were continually analyzed and treated to control biological or scale deposition fouling to ensure continuing heat exchanger performance.

b. Findings

No findings of significance were identified.

.2 Verification of Conditions and Operations Consistent with Design Bases

a. Inspection Scope

The inspector verified that the heat sink and heat exchanger test acceptance criteria were consistent with the design bases. The inspector then reviewed the applicable heat sink and heat exchanger performance calculation revisions to support a planned 5 percent power uprate. This review verified that the licensee's staff had correctly identified the performance conditions and parameters to be attained and, using conservative assumptions, had validated the existence of adequate thermal margin performance for the 30-day postevent period. Through additional review of the appropriate procedures and guidance, the inspector verified that all assumed configuration changes and operator actions required to preserve the assumptions of the analysis, were in fact required by approved procedures and guidance. Finally, the inspector verified that calculations supported multiple design bases such as adequacy of cooling tower basin inventory for 30 days and tower heat removal capacity for the highest predicted heat load for the design basis event.

b. Findings

No findings of significance were identified.

.3 Identification and Resolution of Problems

a. Inspection Scope

The inspector used Inspection Procedure 71152 as a guide for reviewing these issues and subsequently verified that the actions taken for these problems were appropriate and sufficient to prevent recurrence.

The inspector examined the licensee's corrective action program for significant problems with the selected structures, systems, and components. Selected CRs that identified potential issues which could degrade heat sink or heat exchanger performance were reviewed. CRs and a sample of test results identifying issues that could mask decreased performance were also reviewed. The inspector determined that conditions related to single or common cause failures which could lead to degraded heat sink or heat exchanger performance were being identified. The licensee staff had also been successful in the identification of conditions with the potential to mask degraded performance. However, the inspector observed and informed the licensee that the application of corrective action toward resolving these conditions was often untimely. Additional examination did not identify any questions of operability or challenges to safety.

b. Findings

No findings of significance were identified.

.4 Heating Ventilation and Air Conditioning Chiller C Performance Testing

a. Inspection Scope

The inspectors reviewed test acceptance criteria to ensure differences between testing conditions and design conditions were appropriately considered. Test results were compared with acceptance criteria to determine if they were acceptable. The frequency of testing was reviewed to ensure degradation was identified before losing design heat removal capabilities. Test equipment calibration data in the test procedure was reviewed to determine if the test equipment met calibration requirements before commencing the test. The following procedures were reviewed during the assessment:

- STP-410-3603, “Performance Monitoring Program for Control Building Chiller HVK-CHL1C (Division I)”

- ER-98-0012, “Recommended Test Conditions to Measure Heat Removal Capacity of the Control Building Chillers”

- Calculation G13.18.2.1*078-00 Add. D, “Evaluation of Control Building Chillers Performance Test Data”

b. Findings

No findings of significance were identified.

.5 Residual Heat Removal B Heat Exchanger Performance Testing

a. Inspection Scope

The inspectors observed Residual Heat Removal B heat exchanger performance testing, test personnel performance, test control stations, and installed test equipment. The inspectors reviewed test acceptance criteria to ensure differences between testing conditions and design conditions were appropriately considered. Test results were compared with acceptance criteria to determine if they were acceptable. The frequency of testing was reviewed to ensure degradation was identified before losing design heat removal capabilities. Test equipment was reviewed to determine if the equipment met calibration requirements. The following procedures were reviewed during the assessment:

- PEP-0240, “Performance Monitoring Program for the Residual Heat Removal Heat Exchangers E12-EB001B and E12-EB001D (Division II)”
- EPRI NP-7552, “Heat Exchanger Performance Monitoring Guidelines”

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

The inspectors observed the testing of operations personnel in the simulator on January 29, 2001. The observation was performed to determine if there were deficiencies or discrepancies with the training and if the licensee's evaluators conducted an adequate critique of the training. The following procedures were reviewed as part of the assessment:

- EOP-1, “Reactor Pressure Vessel Control,” Revision 16
- EOP-1A, “Reactor Pressure Vessel Control - Anticipated Transient Without Scram,” Revision 16
- EOP-2, “Primary Containment Control,” Revision 12
- EOP-3, “Secondary Containment and Radioactive Release Control,” Revision 11
- EOP-4, “Contingencies - Reactor Pressure Vessel Flooding,” Revision 8
- EOP-4A, “Contingencies - Anticipated Transient Without Scram,” Revision 8

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

.1 Periodic Evaluation Reviews

a. Inspection Scope

The inspectors reviewed the licensee's reports documenting the performance of the last three Maintenance Rule periodic effectiveness assessments. These periodic evaluations covered the period from July 1997 through December 1999.

The inspector verified that the licensee's program had identified and monitored risk-significant functions associated with structures, systems, and components using reliability and unavailability. Additionally, the performance of nonrisk-significant functions were monitored using plant level criteria.

The inspector reviewed the conclusions reached by the licensee with regard to the balance of reliability and unavailability for specific Maintenance Rule functions. This review was conducted by examining the licensee's evaluation of all risk-significant functions that had exceeded performance criteria during the evaluation periods.

The inspector also examined the licensee's evaluation of program activities associated with placement of Maintenance Rule Program risk-significant functions in Categories (a)(1) and/or (a)(2). This review was conducted by the examination of 16 maintenance preventable functional failure evaluations, which licensee personnel had concluded were not maintenance preventable functional failures. Also, the inspector reviewed the periodic evaluation conclusions reached by the licensee for functions of the feedwater, service air, instrument air, service water cooling, automatic depressurization and safety relief valves, residual heat removal, main steam line isolation valve positive leakage control, 120 Vac electric distribution, feedwater heater, control building chilled water, suppression pool cleanup, and process and digital radiation monitoring systems.

b. Findings

No findings of significance were identified.

.2 Effectiveness of Maintenance Rule Program

a. Inspection Scope

The inspector reviewed the Maintenance Rule expert panel meeting minutes for those meetings listed in the attachment with an emphasis on issues associated with functions of the feedwater, service air, instrument air, service water cooling, automatic depressurization and safety relief valves, residual heat removal, main steam line isolation valve positive leakage control, 120 Vac electric distribution, feedwater heater, control building chilled water, suppression pool cleanup, and process and digital radiation monitoring systems. For the identified functions, the inspector followed up by obtaining the needed documentation and assessing the Maintenance Rule Program performance related to:

- Program adjustments made in response to unbalanced reliability and availability

- Cause determination of degraded performance or failure to meet performance criteria

- Adequacy of corrective action and goal setting

- Monitoring of established goals for functions placed in Category (a)(1)

- Program revisions to scoping and risk significance

- Creation of new risk-significant functions to improve performance monitoring

- Assessment of plant level performance

In order to validate that the licensee was identifying programmatic issues from outside of the Maintenance Rule Program, the inspector also reviewed the reports for the self-assessment of the Maintenance Rule Program that are referenced in the attachment.

b. Findings

No findings of significance were identified.

.3 Identification and Resolution of Problems

a. Inspection Scope

The inspector evaluated the use of the corrective action system within the Maintenance Rule Program. This evaluation was accomplished by reviewing the CRs and a sample of the control room logs listed in the attachment. The purpose of this review was to establish that the corrective action program was entered at the appropriate threshold for the purposes of:

- Starting the evaluation and determination of the corrective action process when performance criteria was exceeded

- Correction of performance-related issues or conditions identified during the periodic evaluation

- Correction of generic issues or conditions identified during programmatic surveillances, audits, or assessments

The inspector verified that the identification and implementation of corrective action was acceptable.

b. Findings

No findings of significance were identified.

.4 Review of Maintenance Rule Determinations

a. Inspection Scope

The inspectors selected the following performance problems and evaluated the effectiveness of the licensee's corrective actions and Maintenance Rule determinations:

- CR 2000-1294, "NHS-LDC2C Feeder Breaker for Cooling Tower C Tripped"
- CR 2000-0831, "Breaker 3A on EHS MCC-2A Tripped During Hydrogen Ignitor Testing"
- CR 2000-0188, "Electrical Terminations in Distribution Transformer to E22-S002PNL Made With Improper Material"
- CR 2000-0026, "High Pressure Core Spray Pump Experienced 5-10 Seconds of Cavitation"
- CR 1999-1671, "Stator Cooling Water Pump A Breaker was Found With Springs Uncharged"
- CR 1999-0630, "Division III Feeder Breaker to E22-S004 Tripped"

b. Findings

No findings of significance were identified.

.5 (Closed) Unresolved Item 50-458/0014-04: safety-significance determination for the failure of station blackout valve to open automatically. The issue involved the failure of station blackout Valve SWP-AOV599 to automatically open during testing on March 10, 2000.

The inspectors identified a noncited violation which involved the failure to monitor the performance of station blackout Valve SWP-AOV599 against established Maintenance Rules goals. The issue was determined to be of very low safety significance (Green).

Station blackout Valve SWP-AOV599 is a nonsafety-related valve which actuates automatically during a station blackout event. With Valve SWP-AOV599 open, a cooling water return flow path to the standby service water cooling towers would be enabled. Therefore, the Division III emergency diesel generator and high pressure core spray system would be available during a station blackout event. Valve SWP-AOV599 has a standby service water system Maintenance Rule function (F-256-002) to supply cooling water to the Division III emergency diesel generator during a station blackout event. The licensee's Probabilistic Safety Assessment Model, Revision 3, implemented on January 11, 2001, indicated that the significance of the station blackout valve being out of service was a low-to-moderate increase in the incremental risk of 3.9 E-6/yr.

Between August 1992 and March 1995, Valve SWP-AOV599 was tested as part of the inservice testing program on a quarterly basis. Testing consisted of stroking the valve from the main control room and did not include testing of the entire actuating mechanism. Specifically, the solenoid valves which would be required to actuate during a station blackout event were not tested. On March 2, 1995, testing of Valve SWP-AOV599 was deleted from the inservice testing program because the air supply was not safety related. The licensee's initial safety and environmental evaluation specified that Valve SWP-AOV599 was to be tested for its importance to safety, but not in the inservice testing program. However, no action was taken by the licensee to ensure the valve was added to an appropriate testing program.

On December 5, 1995, the licensee initiated Maintenance Action Item (MAI) 220828 to perform a diagnostic test on Valve SWP-AOV599. The MAI was canceled with a note that specified that the shop did not require it.

On March 4, 1999, engineering personnel initiated CR 1999-0263 to evaluate the need for testing of Valve SWP-AOV599.

On March 10, 2000, the licensee tested Valve SWP-AOV599 in accordance with MAI 330213. Valve SWP-AOV599 failed to open automatically due to blown fuses on the station blackout valve air supply line control solenoid Valve SWP-SOV602C. The licensee initiated CR 2000-0531 to document the failure.

The licensee's investigation determined that the most probable failure date of Valve SWP-SOV602C was the first operation of the valve following the September 25, 1997, postmodification test supporting a fire protection modification. The licensee was unable to identify a specific cause for the failure of Valve SWP-SOV602C. Additionally, engineering and maintenance personnel did not save the solenoid valve for failure analysis when it was replaced following the failed test in March 2000.

In January 2001, engineering personnel informed the inspectors that, with the exception of the inservice testing described above, no routine preventive maintenance or testing had been completed on Valve SWP-AOV599 and its associated solenoid valves since initial installation of the valve and the failure of the valve to actuate during testing in March 2000. The inspectors determined that the failure of the licensee to identify the need for periodic maintenance or testing of Valve SWP-AOV599 and its associated solenoid valves before March 2000 was of concern given the individual contribution to reducing core damage frequency (CDF).

The inspectors evaluated the issue using the SDP because the failure to conduct testing had a credible impact on safety and the issue affected the availability and reliability of the service water system during a station blackout event. The Phase One SDP evaluation identified the need to perform a Phase Two evaluation because the finding represented an actual loss of safety function of a non-Technical Specification component which was designated as risk significant per the Maintenance Rule for greater than 24 hours. Since the Phase 2 worksheets for River Bend Station were still in draft form, a Phase II evaluation was not performed. Consequently, a Phase Three analysis was conducted by an NRC senior reactor analyst to evaluate the risk significance of the issue.

The licensee performed an assessment of this condition as documented in an interoffice memorandum dated September 21, 2000. At the time the issue was identified, the licensee's Probabilistic Safety Assessment, Revision 2D, was in effect. Based on the valve being out of service for an entire year, the incremental risk was determined to be 9.8E-7/yr.

On January 11, 2001, the licensee implemented Probabilistic Safety Assessment, Revision 3. The Revision 3 baseline CDF (including maintenance unavailability) increased to 9.44E-6/yr. The increase in CDF from 3.16E-6/yr (Revision 2D) to 9.44E-6/yr (Revision 3) was due primarily to an increase in probability of nonrecovery of offsite power during loss of offsite power events and an earlier containment failure on loss of all decay heat removal (from 26 hours to 14 hours).

Using Probabilistic Safety Assessment, Revision 3, the licensee determined that the instantaneous risk associated with the failure of Valve SWP-AOV599 was 1.33E-5/yr. Based on the valve being out of service for an entire year, the incremental risk was determined to be:

1.33E-5/yr - 9.44E-6/yr = 3.9E-6/yr

This value did not consider that operators would recognize that Valve SWP-AOV599 was not open and either take actions to open the valve or secure the Division III emergency diesel generator within 5 minutes.

The licensee's letter to the NRC dated February 8, 2001, provided a qualitative description of how the operators would recognize and correct the failure of Valve SWP-AOV599 to open. The licensee stated that, in addition to the operators training and testing on station blackout events, Procedure AOP-0050, "Station Blackout," contained a step directing operators to verify that Valve SWP-AOV599 was open. When the operators recognized that the valve had not opened automatically, they would open the valve manually using a switch in the main control room. The licensee concluded that there was a high probability that operators would recognize the failure of Valve SWP-AOV599 in time to mitigate the consequences and maintain the Division III emergency diesel generator operating.

An NRC senior reactor analyst, in conjunction with the Office of Nuclear Reactor Regulation, performed a review of the licensee's assessment of the Valve SWP-AOV599 failure and determined the following:

- Based on discussions with the inspectors who completed simulator observations and a review of Procedure AOP-0050, there was a high probability that operators would recognize the failure of Valve SWP-AOV599 in time (less than 5 minutes) to mitigate the consequences and maintain the Division III emergency diesel generator operating.

- Although a quantitative assessment was not conducted, the nonrecovery probability associated with these actions was approximately 0.2 or less. A nonrecovery probability of 0.2 or less would result in a conditional core damage probability (delta core damage frequency) of less than 1E-6yr.

  3.9E-6/yr x 0.2 = 7.8E-7/yr

  As a result, when operator recovery actions were considered, the failure of

Valve SWP-AOV599 was determined to be of very low safety significance (Green).

10 CFR 50.65(a)(1) requires, in part, that the licensee monitor the performance or condition of structures, systems, or components, against established goals, in a manner sufficient to provide reasonable assurance that such structures, systems, and components are capable of fulfilling their intended safety functions. Between July 1996 (implementation date of the Maintenance Rule) and March 2000, the licensee did not monitor the performance of station blackout Valve SWP-AOV599 against established goals in a manner sufficient to assure the valve was capable of performing its intended Maintenance Rule function. Specifically, in March 1995, the licensee removed the quarterly performance test of Valve SWP-AOV599 from the inservice testing program and did not place the valve in another testing program. The failure to conduct periodic testing of Valve SWP-AOV599 was of concern because the lack of testing prevented the licensee from being able to identify an adverse condition which existed between September 1997 and March 2000. The inspectors determined that the failure to monitor the performance of Valve SWP-AOV599 was a violation of 10 CFR 50.65(a)(1) (NCV 50-458/0016-04) (EA-01-090). This violation is associated with an inspection finding that is characterized by the SDP as having very low safety 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 1999-0263.

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

a. Inspection Scope

The inspectors evaluated the effectiveness of risk assessments performed by the licensee for the work weeks beginning December 31, 2000, January 28, and February 11, 2001. The following procedures were reviewed during the assessment:

- Maintenance Planning guideline
- On-line maintenance guidelines
- Weekly maintenance schedules

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors observed personnel performance following the licensee's entry into Procedure AOP-0007, "Loss of Feedwater Heating," on December 31, 2000, due to the failure of the first point heater normal level control valve. In addition, the inspectors observed personnel performance following the licensee's entry into Procedure AOP-0007 on January 4, 2001, due to the failure of the first point heater high level dump valve. The following documents were reviewed during the assessment:

- Plant operating logs
- CR 2001-0001, “Feedwater Heater String A Low Water Level”
- CR 2001-0002, “Entered AOP-0007 Loss of Feedwater Heating”
- CR 2001-0020, “Extreme High Level Transient Occurred in First Point Heater A”
- CR 2001-0022, “Entered AOP-0007 Loss of Feedwater Heating”
- Procedure AOP-0007, “Loss of Feedwater Heating”

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following documents to ensure that operability was properly justified, the components remained available, and there was not a significant increase in risk:

- CR 2001-0025, “Discrepancies With Inverter Heat Load Calculation”

- Engineering Report R-IC-00-0002, “Engineering Report for Rosemont Transmitter Saturation”

- CR 2001-249, “CRD Cooling Water Flow Reading Lower Than Required Value”

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds

a. Inspection Scope (71111.16)

The inspectors assessed the cumulative effect of operator workarounds, main control room deficiencies, and annunciator deficiencies. Additionally, the inspectors assessed the licensee’s processes for identifying, tracking, and resolving operator workarounds, main control room deficiencies, and annunciator deficiencies. The following documents were reviewed during the assessment:

- Procedure OSP-0015, “Problem Annunciator Resolution Program”
- Operator workaround main control room deficiency program
- Operator workaround tracking list
- Main control room deficiency tracking list
- Main control room annunciator deficiency list
- Auxiliary control room annunciator deficiency list

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

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

- MAI 334105, “Functional Test of Division II Emergency Diesel Generator Vibration Monitoring Equipment”
- MAI 336546, “Calibrate Division II Emergency Diesel Generator Jacket Water Pump Discharge Pressure Indicator”
- MAI 332928, “Install Time Delay Dropout Relay to Prevent Sudden Motor Reversal and Subsequent Breaker Trip”
- MAI 337739, “Troubleshoot/Rework LPRM 14-15A to correct drifting problem”

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

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

- STP-508-4202, “Drywell Pressure-High Channel Calibration and Logic System Functional Test”
- STP-209-6310, “RCIC Quarterly Pump and Valve Operability Test”
- STP-109-6801, “Main Steam Cold Shutdown Valve Operability Test”
- STP-205-6301, “LPCS Quarterly Pump and Valve Operability Test”
- STP-309-0202, “Division II Emergency Diesel Generator Operability Test”

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

.1 Loose Parts Monitor

a. Inspection Scope

The inspectors reviewed the installation of a spare loose parts sensor (LPM-NBE8) onto the feedwater piping in the drywell which is being used intermittently to locate the source of a loose part indication. The inspectors reviewed the temporary modification and associated 10 CFR 50.59 screening against the licensee's design basis documentation. This included the licensee's USAR and Technical Specifications. The inspectors verified that the temporary modification had not affected system operability or availability and that it had been installed in accordance with the licensee's procedures.

b. Findings

No findings of significance were identified.

.2 Disabled Annunciators

a. Inspection Scope

The inspectors reviewed the process associated with disabling annunciators in the auxiliary control room to ensure plant procedures for removing circuit cards or lifting leads were followed. The following documents were reviewed during the assessment:

- Procedure ADM-0031, “Temporary Alterations”
- Procedure OSP-0015, “Problem Annunciator Resolution Program”
- Main Control Room Annunciator Deficiency Report
- Auxiliary Control Room Annunciator Deficiency Report

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

.1 Simulator Observation

a. Inspection Scope

The inspectors observed simulator training on January 29, 2001, in order to assess the ability of operations personnel to complete classifications and notifications. The following procedures and documents were reviewed during the assessment:

- NRC Emergency Preparedness Position 2, “Emergency Preparedness Position on Timeliness of Classification of Emergency Conditions”

- EIP-2-001, “Classifications of Emergencies”

- EIP-2-002, “Classification Actions”

- EIP-2-006, “Notifications”

b. Findings

No findings of significance were identified.

.2 Emergency Preparedness Drill

a. Inspection Scope

The inspectors observed the licensee’s February 20, 2001, emergency preparedness drill in order to evaluate the adequacy of the drill and critique. The following procedures and documents were reviewed during the assessment:

- NRC Emergency Preparedness Position 2, “Emergency Preparedness Position on Timeliness of Classification of Emergency Conditions”

- EIP-2-001, “Classifications of Emergencies”

- EIP-2-002, “Classification Actions”

- EIP-2-006, “Notifications”

- EP-04, “Scenario 04 Site Drill Manual”

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY
Cornerstones: Occupational Radiation Safety and Public Radiation Safety

2OS2 As Low As Is Reasonably Achievable (ALARA) Planning and Controls (71121.02)

a. Inspection Scope

The inspector interviewed radiation workers and radiation protection personnel involved in high dose rate and high exposure jobs in the controlled access areas during normal operations. Independent radiation surveys of selected work areas within the controlled access areas were conducted. The following items were reviewed and compared with regulatory requirements to determine whether the licensee had an adequate program to maintain occupational exposure ALARA:

- ALARA program procedures

- Processes used to estimate and track exposures

- Plant collective exposure history for the past 3 years, current exposure trends, and 3-year rolling average dose information

- Eleven radiation work permit packages (2000-1081, -1201, -1800, -1904, 2001-1023, -1024, -1028, -1029, -1030, -1048, and -1056) for work activities that had resulted in the highest personnel collective exposures during the inspection period

- One job in a posted high radiation area (radiography) was observed, and tours were conducted in various areas of the plant

- Use of engineering controls to achieve dose reductions

- Exposures of selected work groups (radiation protection, operations, radwaste/decon, instrument and controls, and mechanical maintenance)

- Hot spot tracking and reduction program

- Plant-related source term data, including source term control strategy

- Radiological work planning

- Two quality surveillances (QS-2000-RBS-19 and -026) and one self-assessment (RP-2000-08-RBS)

- Selected corrective action documents involving the ALARA program and radiation worker practice deficiencies (CRs 2000-0656, -0778, -1337, -1338, -1347, -1436, -1467, -1820, 2001-0027 and -0242)

- ALARA Committee meeting minutes (00-18, 00-19, 00-20, 00-21, 00-22, 00-23, 00-24, 00-25, 00-26, 00-27, and 00-28)

- Declared pregnant worker dose monitoring controls

b. Findings

No findings of significance were identified.

2PS1 Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems (71122.01)

a. Inspection Scope

The inspector interviewed cognizant personnel and walked down the major components of the gaseous and liquid release systems to observe ongoing activities, equipment material condition, and the system configuration, as compared to the description in the Updated Final Safety Analysis Report. The following items were reviewed and compared with regulatory requirements:

- 1999 Radiological Effluent Release Report

- Changes to the offsite dose calculation manual (ODCM) and to the radioactive waste system design and operation

- Anomalous results, if any, reported in the radiological effluent release report

- Effluent radiological occurrence performance indicator incidents

- Sample collection and analysis of the radwaste building vent gaseous effluent release point

- Selected radioactive effluent release permits and associated projected doses to members of the public (2000027, 2000080, 2000086, 2000089, 2000091, 2000096, 2000097, 2000107, 2000146, 2000157, 2001001, 2001002, 2001006, and 2001010)

- Compensatory sampling and radiological analyses conducted when effluent monitors were declared out of service

Monthly, quarterly, and annual dose calculations:

- Engineered safety feature air cleaning system surveillance test results (STP-257-3601(A&B), STP-402-3601(A&B), STP-406-3601(A&B), STP-257-1600, STP-257-1601, STP-402-1600, STP-402-1601, STP-406-1600, and STP-406-1601)

- Records of instrument calibrations performed since the last inspection for each point of discharge effluent radiation monitor and flow measurement device

- Effluent radiation monitor alarm setpoint values

- Calibration and quality control records of counting room instrumentation associated with effluent monitoring and release activities

- Audit (QA-6-2000-RBS-1), quality surveillance (QS-2001-RBS-005), and self-assessment (RP-2001-03-RBS) related to the radioactive effluent treatment and monitoring program

- Selected condition reports related to the radioactive effluent treatment and monitoring program (1999-2010, -0271, -0273, -0403, -0409, -0934, -0982, -1001, -1218, -1230, -1418, -1460, -1567, -1731, -1893, -2033, -2072; 2001-0015, -0224, and -0310).

b. Findings

No findings of significance were identified.

2PS3 Radiological Environmental Monitoring Program and Radioactive Material Control Program (71122.03)

a. Inspection Scope

The inspector interviewed members of the licensee's staff responsible for implementing the radiological environmental, meteorological monitoring, and radioactive material control programs. The inspector observed the following activities and equipment:

- Preparation of airborne particulate and charcoal samples for analysis
- Meteorological instrument data displays at the primary and secondary meteorological towers and in the control room
- Two environmental air sampler stations (AGC and AQS-2) and five thermoluminescent dosimetry stations (TA-1, TG-S, TQ-1, TQS-2, and TR-1)

No material was released from the controlled access area; therefore, this aspect of the above procedure was not performed.

The following items were reviewed and compared with regulatory requirements to determine whether the licensee had an adequate program to verify the impact of radioactive effluent releases to the environment and to ensure that the licensee's surveys and controls were adequate to prevent the inadvertent release of licensed materials into the public domain:

- Implementing procedures for the radiological environmental monitoring program
- Number and location descriptions of the environmental sampling stations as specified in the offsite dose calculation manual
- Environmental sample analytical results
- Calibration and maintenance records for environmental air sampling equipment and radiation measurement instrumentation
- Changes to the radiological environmental monitoring program
- 1999 Annual Radiological Environmental Operating Report (Files G9.5, G9.25.1.5, and G10.6)
- The environmental laboratory's performance in the interlaboratory comparison program
- Implementing procedures for the meteorological monitoring program
- Meteorological instrument operability, reliability, and annual meteorological data recovery
- Procedures, methods, and instruments used to survey, control, and release materials from the controlled access area
- Calibration procedures and records for instruments used to perform radiological surveys prior to material release

- Detection sensitivities of radiation survey instruments used for the release of potentially contaminated materials from the controlled access area

- Criteria used for the unrestricted release of potentially contaminated material from the controlled access area

- Audit QA-6-2000-RBS-1 and Surveillance QS-2001-RBS-007

- A summary of meteorological, environmental, and release of licensed radioactive material related corrective action reports written since February 1, 2000 (10 of these reports were reviewed in detail: CR2000-0241, -1218, -1268, -1525, -1669, -1697, and -0131; CR2001-0241, -0329, and -0334)

b. Findings

No findings of significance were identified.

3. SAFEGUARDS
Cornerstone: Physical Protection

3PP1 Access Authorization (71130.01)

a. Inspection Scope

The inspector performed the following inspection activities:

- Reviewed LERs and safeguards event logs to identify problems in the access authorization program

- Reviewed procedures, audits, and self-assessments of the following programs/areas: behavior observation, access authorization, fitness-for-duty, supervisor and escort training, and requalification training

- Interviewed five supervisors/managers and five individuals who were authorized to escort visitors into the protected and/or vital areas to determine their knowledge and understanding of their responsibilities in the behavior observation program

- Reviewed CRs, LERs, safeguards event logs, audits, selected security event reports, and self-assessments for the licensee's access authorization program to determine the licensee's ability to identify and resolve problems

b. Findings

No findings of significance were identified.

3PP2 Access Control (71130.02)

a. Inspection Scope

The inspector performed the following inspection activities:

- Reviewed LERs and safeguards event logs to identify problems with access control equipment

- Reviewed procedures and audits for testing and maintenance of access control equipment and for granting and revoking unescorted access to protected and vital areas

- Interviewed security personnel concerning the proper operation of the explosive and metal detectors, X-ray devices, and key card readers

- Observed licensee testing of access control equipment and the ability of security personnel to control personnel, packages, and vehicles entering the protected area

- Reviewed procedures to verify that a program was in place for controlling and accounting for hard keys to vital areas

- Reviewed the licensee’s process for granting access to vital equipment and vital areas to authorized personnel having an identified need for that access

- Reviewed CRs, LERs, safeguards event logs, audits, selected security event reports, and self-assessments for the licensee's access control program in order to assess the licensee's ability to identify and resolve problems with the access control program

- Interviewed key security department and plant support personnel to determine their knowledge and use of the corrective action reports and resolution of problems regarding repair of security equipment

b. Findings

No findings of significance were identified.

3PP3 Security Plan Changes (71130.04)

a. Inspection Scope

The inspector completed the following inspection activities:

- Reviewed the Physical Security Plan, Revisions 19 and 19A; Safeguards Contingency Plan, Revision 8; and Training and Qualifications Plan, Revision 13A to determine if requirements of 10 CFR 50.54(p) had been met.

- Reviewed the previous year’s safeguards event logs and interviewed security personnel to determine their knowledge and use of the corrective action program and resolution of problems as it relates to making changes to the licensing documents.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Reactor Core Isolation Cooling and Residual Heat Removal Performance Indicators

a. Inspection Scope

The inspectors used NRC Inspection Manual Procedure 71151, Performance Indicator Verification, to verify the accuracy and completeness of data associated with the safety system unavailability performance indicators. Systems reviewed during the assessment included reactor core isolation cooling and residual heat removal. The following procedures and documents were reviewed during the verification:

- Performance indicator data summary report for the first quarter of 2000
- Performance indicator data summary report for the second quarter of 2000
- Performance indicator data summary report for the third quarter of 2000
- Performance indicator data summary report for the fourth quarter of 2000

b. Findings

No findings of significance were identified.

.2 Physical Security Performance Indicators

a. Inspection Scope

The inspector reviewed the program for collection and submittal of performance indicator data. Specifically, a random sampling of security event logs and corrective action reports were reviewed for the following program performance areas:

- Protected area security equipment
- Personnel screening program performance
- Fitness-for-duty/personnel reliability program performance

b. Findings

No findings of significance were identified.

.3 (Closed) Unresolved Item 50-458/0011-07: review of process to revise performance indicator data based on an error in historical data. The issue involved an error in the licensee’s submitted historical data due to incorrect World Association of Nuclear

Operators information. The inspectors reviewed the NRC's response to Frequently Asked Question 170 to Nuclear Energy Institute 99-02, "Regulatory Assessment Performance Indicator Guideline," and determined that the licensee was not required to revise the historical data. Therefore, the licensee's historical data submission was acceptable.

4OA3 Event Followup

.1 (Closed) LER 50-458/0011-00: secondary containment inoperable due to failure of auxiliary building door opening device. On August 10, 2000, while the plant was at 100 percent power, the licensee identified that a secondary containment door leading into the auxiliary building had been blocked open for a short period of time when the door opening device did not reset appropriately. Due to the low safety significance of this issue, the inspectors verified that the licensee had entered this issue in their corrective action program (CR 2000-1470) and determined that no additional inspection was warranted.

.2 (Closed) LER 50-458/0013-00: SLC system inoperable due to foreign material in the storage tank. See Section 1R04.

.3 (Closed) LER 50-458/0014-00: postmaintenance document review discovered 9-hour period of concurrent inoperability of two emergency diesel generators. This item was withdrawn by the licensee.

.4 (Closed) LER 50-458/0015-00: inadequate surveillance test procedure results in failure to fully perform required surveillance. See Section 4OA7.

.5 (Closed) LER 50-458/0016-00: automatic actuation of primary containment isolation valve in suppression pool cooling system due to test switch failure. The inspectors determined that the issue is minor and warrants no additional inspection.

.6 (Closed) Unresolved Item 50-458/0011-06: review of support system unavailability hours for one of four standby service water pumps out of service. The inspectors reviewed the issue with personnel from the Office of Nuclear Reactor Regulation and determined that the licensee did not need to assume a single failure of equipment when removing a support system from service. Therefore, the licensee's determination that only the residual heat removal system should be considered unavailable when one of four standby service water pumps was removed from service was appropriate.

4OA6 Management Meetings

Exit Meeting

The inspectors presented the inspection results to Mr. R. Edington, Vice-President, Operations, and other members of licensee management at the conclusion of various parts of the inspection on February 16, March 8, and March 30, 2001. The inspectors presented the inspection results to Mr. Dwight Mims, General Manager, Plant Operations, and other members of licensee management on March 22, 2001.

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

4OA7 Licensee Identified Violations

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

.1 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. The licensee determined that they failed to implement effective corrective actions to correct a condition adverse to quality involving the performance of loss of offsite power logic system functional testing. Consequently, during Refueling Outage 8, the licensee did not adequately perform Technical Specification Surveillance Requirement 3.3.8.1.4, which required that a logic system functional test of the loss of offsite power instrumentation be completed every 18 months. The issue is described in the licensee's corrective action program reference CR 2000-1813 and LER 50-458/0015. This is a noncited violation (NCV 50-458/0016-05).

.2 10 CFR 20.1501(a) states, in part, each licensee shall make or cause to be made, surveys that are reasonable under the circumstances to evaluate radiation levels, concentrations or quantities of radioactive material, and potential radiological hazards. On October 9, 2000, the licensee identified three examples of detectable licensed radioactivity that was unconditionally released from the controlled access area, as described in the licensee's corrective action program, reference CR 2000-1788. This is a noncited violation (NCV 50-458/0016-06).

.3 Technical Specification 5.5.1.b requires that the ODCM contain radioactive effluent controls. ODCM 7.2.2.1 states, in part, that release rates shall be administratively controlled to maintain the fraction of 10 CFR Part 20 limits less than or equal to 0.3. Station Procedure SOP-0113, step 5.6.20, requires that the LWS-FIC197 (liquid effluent discharge flow) setpoint is adjusted to the desired flow rate not to exceed the value specified on the liquid radwaste discharge permit. On February 26, 2000 (Permit 2000027) and April 15, 2000 (Permit 2000080), liquid discharges were made which exceeded the maximum allowable release rate, as described in the licensee's corrective action program (CRs 2000-0403 and -0982). This is a noncited violation (NCV 50-458/0016-07).

ATTACHMENT

SUPPLEMENTARY INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

R. Biggs, Coordinator, Licensing
W. Brian, Director, Engineering
E. Bush, Superintendent, Operations
R. Edington, Vice President-Operations
J. Fowler, Manager, Quality Assurance
R. Frayer, Supervisor, System Engineering
H. Goodman, Superintendent, Reactor Engineering
D. Heath, Supervisor, Radiation Protection
T. Hildebrandt, Manager, Maintenance
H. Holmes, Coordinator, ALARA
J. Holmes, Manager, Technical Support
G. Javaherian, Senior Engineer
C. Kelly, Director, Security - EOI
R. Kerar, Fire Protection Engineer
R. King, Director, Nuclear Safety Assurance
J. Leavines, Manager, Licensing
F. Lenox, Technical Specialist IV, Maintenance Rule Coordinator
D. Lorfing, Coordinator, Licensing
J. McGhee, Manager, Operations
D. Mims, General Manager
D. Myers, Senior Specialist, Licensing
C. Parker, Superintendent, Plant Security
J. Roberts, Senior Lead Engineer
A. Shahkarami, Manager, System Engineering
D. Stewart, System Engineer
M. Walton, Licensing
D. Wells, Superintendent, Radiation Protection
D. Williamson, Licensing Specialist
M. Wyatt, Manager, Planning and Scheduling/Outage

ITEMS OPENED AND CLOSED

Opened and Closed

50-458/0016-01	NCV	Failure to maintain the required inventory of alternate standby liquid control chemicals (Section 1R04.7)
50-458/0016-02	NCV	Failure to complete annual walkdowns of emergency operating procedure enclosures (Section 1R04.7)
50-458/0016-03	NCV	Failure to maintain a fire barrier between two fire areas which contain redundant safe shutdown equipment (Section 1R05.1)

50-458/0016-04	NCV	Failure to monitor the performance of a standby service water component against established goals to ensure it was capable of performing its Maintenance Rule function (Section 1R12.5)
50-458/0016-05	NCV	Licensee identified failure to implement corrective actions for a condition adverse to quality (Section 4OA7)
50-458/0016-06	NCV	Failure to survey licensed radioactive materials (Section 4OA7)
50-458/0016-07	NCV	Failure to control liquid effluent release rates below the value specified on two discharge permits (Section 4OA7)

Closed

50-458/9915-02	VIO	Failure of operating personnel to be aware of plant indications (Section 1R04.6)
50-458/0011-03	VIO	Failure to complete monthly inspections of portable fire extinguishers (Section 1R05.3)
50-458/0011-07	URI	Review of process to revise performance indicator data based on an error in historical data (Section 4OA1.2)
50-458/0014-01	URI	Safety-significance determination for unavailability of the normal and alternate standby liquid control systems (Section 1R04.7)
50-458/0014-04	URI	Significance determination for the failure of station blackout valve to open (Section 1R12.5)
50-458/0011-00	LER	Secondary containment inoperable due to failure of auxiliary building door opening device (Section 4OA3.1)
50-458/0013-00	LER	Standby liquid control system inoperable due to foreign material in the storage tank (Section 4OA3.2)
50-458/0014-00	LER	Postmaintenance document review discovered 9-hour period of concurrent inoperability of two emergency diesel generators (Section 4OA3.3)
50-458/0015-00	LER	Inadequate surveillance test procedure results in failure to perform required surveillance (Section 4OA3.4)
50-458/0016-00	LER	Automatic actuation of primary containment isolation valve in suppression pool cooling system due to test switch failure (Section 4OA3.5)
50-458/0011-06	URI	Review of support system unavailability hours for one of four standby service water pumps out of service (Section 4OA3)

DOCUMENTS REVIEWED

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

Condition Reports

1997-0891	1999-1790	2000-1104	2000-1746
1998-0794	1999-1898	2000-1302	2000-2185
1998-1501	2000-0075	2000-1410	2000-2207
1999-0957	2000-0132	2000-1261	2001-0001
1999-1306	2000-0225	2000-1287	2001-0232
1999-1438	2000-0312	2000-1298	
1999-1614	2000-0934		
		2000-1705	

Maintenance Action Items

310025	325749	336673	336696	336717
314783	334562	336695	336716	336996

Engineering Reports

DESCRIPTION	REVISION
Ultimate Heat Sink Inventory Evaluation Feasibility Study	03/30/1997
Service Water System Design Criteria	01

Procedures

NUMBER	DESCRIPTION	REVISION
AOP-0004	Loss Of Offsite Power	21
AOP-0050	Station Blackout	13
ARP-870-55	Annunciator Response, Alarm No. 1066	7
*EDG-PR-001	*Reliability Monitoring Program	*0, 02
*EDG-PR-001	*Maintenance Rule Program	03, 04
*EDG-PR-001	*Maintenance Rule and Unavailability Monitoring Programs	05
OM-110	Continual Behavioral Observation Program	01
*PEP-0219	*Reliability Monitoring Program	*07, *08
PSP-4-300	Access Control	21
SPI-09	Vehicle Search/Warehouse Officer	26

Procedures

NUMBER	DESCRIPTION	REVISION
STP-256-6303	Standby Service Water A Loop Quarterly Pump and Valve Operability Test	14
STP-256-6304	Standby Service Water B Loop Quarterly Pump and Valve Operability Test	13

Calculations

NUMBER	DESCRIPTION	REVISION
PM-194	Standby Cooling Tower Performance and Evaporation Losses	5
G13.18.12.3*042	River Bend Specific Complimentary Cumulative Density Function (CCDF) for the Recovery of LOSP as a Function of Time for the Level 1 IPE	1
G13.18.13.2*088	Temperature and Inventory Effects of Maximum Safeguards Operation on the Ultimate Heat Sink	0
G13.18.13.2*088	Temperature and Inventory Effects of Maximum Safeguards Operation on the Ultimate Heat Sink	0A

Drawings

NUMBER	DESCRIPTION	REVISION
NCT-678-T2H	Cooling Tower General Arrangement	5
NCT-678-T12H	Cooling Tower General Arrangement	3
84449	18" Expansion Joint Assembly	D
88130-131	Heat Exchanger added Tolerances	1
P&ID-9-10E	P&ID Standby Service Water	18
P&ID-9-10H	P&ID Normal Service Water	21
P&ID-9-11A	P&ID Service Water Cooling	10
P&ID-9-11B	P&ID Service Water Cooling	5

Miscellaneous Documents

00-OTN-068, M&TE Out-Of-Tolerance for HDA-043A, March 25, 2000

00-OTN-024, M&TE Out-Of-Tolerance for HDA-043A, February 16, 2001

1997 Maintenance Rule Periodic Assessment Report, March 30, 1998

1998 Maintenance Rule Periodic Assessment Report, January 6, 2001

1999 Maintenance Rule Periodic Assessment Report, February 9, 2001

Access Authorization and Fitness for Duty Audit RBF5-00-0015, dated June 12, 2000

Entergy General Employee Training Handbook

Fitness for Duty Semi-annual Reports for year 2000

General Electric Access Authorization Audit, VA 00059, dated November 30, 2000

Individual Plant Examination dated January 15, 1993

LER 458/99-04

LER 458/99-13

Maintenance Rule: Performance Criteria vs. Actual Performance, February 12, 2001

Maintenance Rule: Functional Failures (Past 01-Year Period), February 9, 2001

Maintenance Rule Expert Panel Meeting Minutes Number 086, February 16, 1999

Maintenance Rule Expert Panel Meeting Minutes Number 087, March 2, 1999

Maintenance Rule Expert Panel Meeting Minutes Number 088, August 10, 1999

Maintenance Rule Expert Panel Meeting Minutes Number 089, October 6, 1999

Maintenance Rule Expert Panel Meeting Minutes Number 090 July 21, 2000

Maintenance Rule Expert Panel Meeting Minutes Number 091, December 5, 2000

Probabilistic Safety Assessment Revision 2D Update

Probabilistic Safety Assessment Revision 3

Risk Impact of SWP-MOV077A Out of Service dated February 10, 2000

River Bend letter to NRC dated February 8, 2001

Safeguards Event Logs - First through fourth quarters 2000 and first quarter 2001

Self-Assessment Maintenance Rule - Functional Failure Determinations, October 4, 2000

Self-Assessment Maintenance Rule - Second Phase, January 6, 2001

Service Water Fault Trees pages 37, 38, and 41

Summary of Cycle 9 EOOS Changes dated July 19, 1999

Westinghouse Access Authorization and Fitness for Duty Audit, QAA-00-027, dated November 30, 2000

LIST OF ACRONYMS AND INITIALISMS USED

ALARA	as low as is reasonably achievable
ATWS	anticipated transient without scram
CDF	core damage frequency
CFR	Code of Federal Regulations
CR	condition report
LER	licensee event report
MAI	maintenance action item
NCV	noncited violation
NRC	U.S. Nuclear Regulatory Commission
ODCM	offsite dose calculation manual
SLC	standby liquid control
SDP	significance determination process
SRA	senior reactor analyst
URI	unresolved item
VIO	violation