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

July 5, 2000

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 INSPECTION REPORT NO. 50-458/00-10

Dear Mr. Edington:

On June 24, 2000, the NRC completed inspections at your River Bend Station facility. The enclosed report presents the results of these inspections. The results of the inspections were discussed with you and other members of your staff.

Based on the results of this inspection, three issues of very low safety significance (green) were identified. Two of these issues were determined to involve violations of NRC requirements. However, the violations were not cited due to their very low safety significance and because they have been entered into your corrective action program. If you contest these noncited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U. S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U. S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011, 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 No.: 50-458
License No.: NPF-47

Enclosure:
NRC Inspection Report No.
50-458/00-10

cc w/enclosure:
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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)**
Wayne Scott **(WES)**

R:_RB\RB2000-10RP-TWP.wpd

RIV:RI:DRP/B	SRI:DRP/B	PE:DRP/B	SPE:DRP/B	SEPI:DRS/PSB
SMSchneider	TWPruett	RVAzua	RAKopriva	WAMaier
E-WDJohnson	***T-WDJohnson***	***/RA/***	***/RA/***	***/RA/***
7/5/00	7/5/00	6/30/00	6/29/00	7/5/00

EPI:DRS/PSB	C:DRS/PSB	C:DRP/B		
PJElkmann	GMGood	WDJohnson		
WAMaier for	***/RA/***	***/RA/***		
7/5/00	7/5/00	7/5/00		

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-458
License No.:	NPF-47
Report No.:	50-458/00-10
Licensee:	Entergy Operations, Inc.
Facility:	River Bend Station
Location:	5485 U.S. Highway 61 St. Francisville, Louisiana
Dates:	May 7 through June 24, 2000
Inspectors:	T. W. Pruett, Senior Resident Inspector S. M. Schneider, Resident Inspector R. A. Kopriva, Senior Project Engineer R. V. Azua, Project Engineer W. A. Maier, Senior Emergency Preparedness Inspector P. J. Elkmann, Emergency Preparedness Inspector
Approved By:	William D. Johnson, Chief, Project Branch B Division of Reactor Projects

ATTACHMENTS:
1. Supplemental Information
2. NRC's Revised Reactor Oversight Process

SUMMARY OF FINDINGS

River Bend Nuclear Station
NRC Inspection Report 50-458/00-10

The report covers a 6-week period of resident inspection and an announced inspection by two regional emergency preparedness inspectors. 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 determined that engineering personnel provided inaccurate information to operations personnel on the functional capability of the residual heat removal heat exchanger bypass valve following the inspectors' discovery that the antirotation device had fallen off. Consequently, operations personnel took conservative action to disable the suppression pool cooling function of residual heat removal Train A for approximately 36 hours.

  Disabling the residual heat removal Train A suppression pool cooling function had a small impact on safety and affected the safety function of a train of a mitigating system. This issue was of very low risk significance because redundant methods of suppression pool cooling remained operable and unavailability time was less than that allowed by the Technical Specifications (Section 1R15.1).

- Green. The inspectors determined that planning personnel failed to identify required postmaintenance testing requirements in four maintenance packages. The failure to identify the appropriate postmaintenace testing requirements as required by planning procedures was considered a violation of Technical Specification 5.4.1.a. This issue was entered into the licensee's corrective action program as Condition Reports 2000-1010 and 2000-1199.

  The risk significance of this issue was very low because in-process maintenance activities provided assurance that the affected components were functionally capable (Section 1R19.1).

- Green. The failure to perform functional testing of standby service water supply Valve SWP-MOV502B following breaker maintenance resulted in the Division II primary containment unit cooler being inoperable while the facility was in MODE 1 between February 9 and March 4, 2000. The failure to restore the Division II containment unit cooler within 7 days with the facility in MODE 1 was considered a violation of Technical Specification 3.6.1.7. The circumstances involving this issue were discussed in Licensee Event Report 50-458/00-05. This issue was entered into the licensee's corrective action program as Condition Report 2000-0736.

  The inspectors and a senior reactor analyst used the significance determination process to evaluate the risk significance of this issue. The most limiting initiating event was an anticipated transient without scram. The risk significance for this event was very low because one containment unit cooler and two residual heat removal trains in the suppression pool cooling mode were available for mitigation (Section 1R19.2).

Report Details

Summary of Plant Status: The plant operated essentially at 100 percent power throughout the inspection period.

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

1R04 Equipment Alignment

a. Inspection Scope

The inspectors performed a partial equipment alignment check on the Division II emergency diesel generator (EDG) and the high pressure core spray system to ensure that the systems were in the correct configuration for Mode 1 operations.

b. Issues and Findings

There were no findings identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors toured the Divisions I, II, and III EDG rooms 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.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors selected the following three performance problems associated with EDG systems and evaluated the effectiveness of the licensee's corrective actions and maintenance rule determinations.

- Condition Report 2000-0192, Fuel oil level transmitter out of tolerance
- Condition Report 2000-0232, Heater control switch found in off position
- Condition Report 2000-0875, Diesel generator lube oil cooler leak

b. Issues and Findings

There were no findings identified.

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 May 15, May 29, and June 3, 2000.

b. Issues and Findings

There were no findings identified

1R15 Operability Evaluations

.1 Residual Heat Removal Heat Exchanger Bypass Valve

a. Inspection Scope

The inspectors reviewed the licensee's response to the inspectors' identification that the antirotation device was missing from residual heat removal (RHR) heat exchanger bypass Valve E12-MOVF048A.

b. Issues and Findings

On May 16, 2000, the inspectors identified that the antirotation device had fallen off RHR heat exchanger bypass Valve E12-MOVFO48A. The antirotation device prevents stem rotation during the opening or closing stroke of the valve. Valve E12-MOVF048A provided two safety functions. In the closed position, the valve provided suppression pool cooling. In the open position, the valve provided low pressure coolant injection.

Engineering personnel initially informed operations personnel that the operation of Valve E12-MOVF048A would be impacted by the missing antirotation device. Consequently, operations personnel tagged Valve E12-MOVF048A in the open position to maintain low pressure coolant injection capability. Tagging the valve in the open position disabled the suppression pool cooling function of the valve. Disabling the suppression pool suction function resulted in an entry into Technical Specification 3.6.2.3, "RHR Suppression Pool Cooling."

During a subsequent discussion with the inspectors, one system engineer stated that the valve would be impacted by the missing antirotation device while a second engineer stated that the valve would not be affected. The inspectors were concerned that operations personnel may have unnecessarily disabled the suppression pool cooling function of Valve E12MOV-F048A. Engineering personnel subsequently performed an additional review and determined that the valve would have performed its safety function without the antirotation device. Engineering personnel then provided a justification to operations personnel which supported the degraded but operable condition associated with Valve E12MOV-F048A. The results of the evaluation were not used since corrective maintenance was in progress to replace the antirotation device.

The inspectors determined that the initial information provided to operations personnel regarding the antirotation device and the impact on valve operability resulted in operations personnel unnecessarily disabling the suppression pool cooling function of the RHR system for approximately 36 hours. Disabling the suppression pool cooling function was of very low risk significance in that the unavailability time was less than that allowed by the Technical Specifications. Additionally, redundant methods of suppression pool cooling remained operable. Therefore, this issue did not meet the initial significance determination process screening and is considered to be green.

.2 Auxiliary Building Door AB098-04, Agastat/ETR Relay, and Leakage from the Division 2 Standby Service Water to Normal service Water System Operability Concerns

a. Inspection Scope

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

- Auxiliary Building Door AB098-04 was found with some sections of the door seal cracked.
- Nine Agastat Model EGPI004 and ETR relays with a code date of 9629 and 9746, with potential soldered connection problems as described in a GE Part 21 report, were discovered to be installed in plant equipment. Equipment affected included upper and lower containment personnel air-lock supply air isolation valves, main steam isolation valve position indication relays, and the instrument air supply control relay for the service water line vacuum release accumulator.
- Division 2 Standby Service Water system was found to be leaking into the Normal Service Water system. This was identified during a Division 2 Emergency Core Cooling System test.

The inspectors discussed the issues with licensee personnel and reviewed the following documents:

- Condition Report 2000-0849, “Door AB098-04 Seal Was Found Cracked.”
- Condition Report 2000-0280, “Technical Evaluation of Reasonable Assurance of Operability of 10 CFR 21 notification on Agastat and ETR relays.”
- Condition Report 2000-0865, "Leakage identified from Division 2 Standby Service Water system to the Normal Service Water system".

b. Issues and Findings

There were no findings identified.

1R19 Postmaintenance Testing (PMT)

.1 Determination of PMT Requirements

a. Inspection Scope

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

- Maintenance Action Item (MAI) 326198, Replace Diesel Generator B shutdown actuation solenoid Valve EGS-SOVY22B
- MAI 318876, Replace leakage control system Valve LSV-V110CB
- MAI 318877, Replace leakage control system Valve LSV-V110EB
- MAI 333443, Retorque of RHR Pump A suction flange

b. Issues and Findings

The inspectors identified a noncited violation for not specifying PMT requirements in maintenance packages. Following additional review, the licensee determined that in-process maintenance activities provided reasonable assurance that components were functionally capable.

Maintenance Action Item (MAI) 326198 included PMT requirements for a functional test and operational leak test. The standard PMT requirements for replacing solenoid valves, as specified in the work management system, were: functional test, operational leak check, verification that hardware is secure, and verification that wiring was restored. The inspectors determined that the PMT requirements for MAI 326198 did not include a verification that the hardware was secure and that the wiring was restored. The licensee stated that, even though not all of the PMTs were specified, the functional test of the solenoid valve provided reasonable assurance of operability.

MAIs 318876 and 318877 included PMT requirements for a functional test, an operational leak test, and verification that the hardware was secure. The standard PMT requirements for replacement of a check valve, as specified in the work management system, were: functional test, operational leak check, verification that hardware was secure, verification of valve orientation, verification of reconnected piping, and verification of welds by nondestructive testing. The inspectors determined that the PMT requirements for MAIs 318876 and 318877 did not include verification of valve orientation, reconnected piping, and nondestructive testing of welds. The licensee stated that, even though not all of the PMTs were specified, the quality control inspector involvement during welding and the functional test of the penetration valve leakage control system compressor provided a reasonable assurance of operability.

During a review of PMT requirements in April 2000, the inspectors identified that not all PMT requirements had been specified for MAI 323585, "Replace Division I EDG Fuel

Booster Pump." Specifically, MAI 323585 did not specify the operational test requirements. The issue was determined to be an isolated example and was dispositioned as a minor violation. Because of the three additional examples of PMT issues identified during this inspection, the inspectors determined that not specifying all of the applicable PMT requirements in maintenance packages was more than a minor violation of NRC requirements.

The inspectors determined that, even though not all of the required PMTs were identified and completed, reasonable assurance existed to demonstrate the continued reliability and operability of the above components. 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. Section 4.4 of Procedure ADM-0080, "Postmaintenance Testing," specified that planning personnel were responsible for identifying the required PMT and placing the necessary documentation into the corrective maintenance work package. The failure of planning personnel to identify the required PMT in maintenance documents is a violation of Technical Specification 5.4.1.a and is being treated as a noncited violation (50-458/0010-01). This violation is in the licensee's corrective action program as Condition Reports 2000-0911 and 2000-1010.

.2 <u>(Closed) Licensee Event Report (LER) 50-458/0005-00 and LER 50-458/0005-01</u>: Incorrectly connected motor leads for primary containment unit cooler service water supply valve. One noncited violation was identified for the failure to implement PMT requirements which verified the functional capability and operability of a primary containment unit cooler.

On March 21, 2000, with the plant in MODE 5, the licensee identified that the motor operator breaker for containment unit cooler service water supply Valve SWP-MOV502B was incorrectly wired. The wiring error caused Valve SWP-MOV502B to move in the closed direction upon receipt of an open signal. The licensee subsequently determined that Valve SWP-MOV502B had been inoperable since the performance of its associated breaker overload test on February 9, 2000.

Section 7.1.4 of Procedure STP-303-1601, "120 and 480 VAC Breaker Overload Functional Test," required a functional test for the connected load per the applicable procedure. Additionally, Section 5.9.5 of Procedure ADM-0022 required, in part, that, following maintenance which required the breaker to be racked out, a test of the breaker must be performed in the connect position, thus requiring the operation of the equipment supplied by the breaker. The PMT completed by maintenance personnel involved a verification that the breaker phase leads were properly relanded. However, maintenance technicians reversed Phases B and C during the installation of the breaker and the independent verification failed to identify the discrepancy. No functional test of the component was performed.

Technical Specification 3.6.1.7 required that two primary containment unit coolers be operable in MODES 1, 2, and 3. With one primary containment unit cooler inoperable, the licensee must restore the primary containment unit cooler to an operable status within 7 days, or be in MODE 3 in the next 12 hours and MODE 4 in the next 36 hours. The inspectors determined that the failure to perform functional testing of Valve SWP-MOV502B following breaker maintenance resulted in the Division II primary containment unit cooler being inoperable while the facility was in MODE 1 between February 9 and March 4, 2000. The failure to restore the Division II primary containment unit cooler within 7 days with the facility in MODE 1 is a violation of Technical Specification 3.6.1.7 and is being treated as a noncited violation (NCV 50-458/0010-02). The issue was entered into the licensee's corrective action system as Condition Report 2000-0736.

Risk Significance

Valve SWP-MOV502B was designed to open automatically on a high drywell pressure, low reactor vessel level, and high containment to annulus differential pressure. Because the valve would have closed upon receipt of an open signal, the Division II containment unit cooler would not have been available for containment temperature and pressure control during emergency conditions.

Updated Safety Analysis Report Section 6.2.2 specified that the primary containment unit coolers are not required to mitigate the consequences of a loss of coolant accident except in the case of drywell steam bypass. The licensee evaluated the magnitude of steam bypass leakage necessary to exceed containment temperature and pressure limits with no containment unit coolers in operation. The evaluation concluded that the actual drywell steam bypass leakage rate was approximately 11 percent of the steam bypass leakage rate necessary to exceed containment temperature and pressure limits with no containment unit coolers in service. Therefore, plant operation with one containment unit cooler out of service would not have resulted in containment temperature and pressure limits being exceeded during a loss of coolant accident.

The inspectors and a senior reactor analyst used the significance determination process to evaluate the risk significance of this event. During the Phase 1 screening the, inspectors determined that a Phase 2 screening was required for transients, small break loss of coolant accident (LOCA), and an anticipated transient without scram (ATWS). Using Significance Determination Process Table 1 event types and an exposure time of 3-30 days, the inspectors determined that the "Estimated Likelihood Rating" was B for a transient, C for a small break LOCA, and F for ATWS. The inspectors and the senior reactor analyst reviewed the issues and determined that mitigation would have been achieved for a transient, small break LOCA, and ATWS.

The inspectors and senior reactor analyst determined that the most limiting event was an ATWS. The "Estimated Likelihood Rating" for an ATWS was F, based on an "Exposure Time for Degraded Conditions" of 3-30 days and a frequency of 1 event per 10,000 to 100,000 years. Referring to Significance Determination Process Table 2, the inspectors determined that, even though one containment unit cooler was unavailable, two RHR systems for the suppression pool cooling mode and one containment unit cooler were available. The available mitigation capability for an estimated likelihood of F indicated that the issue was of very low risk significance and within the licensee's response band (green).

Corrective Actions

The licensee implemented several corrective actions which involved, in part, an inspection of the breaker, restoration of the breaker to an operable condition, an evaluation of the actual motor operated valve stresses, and a review of other three-phase breakers to ensure each connected component had been tested following breaker maintenance.

1R22 Surveillance Testing

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 Updated Safety Analysis Report, and procedures were met:

- STP-209-6310, "RCIC Quarterly Pump and Valve Operability Test"
- STP-204-1300, "LPCI Pump “A” Start Time Delay Channel Calibration and Channel Functional Test"
- STP-204-6303, “DIV I RHR Quarterly Valve Operability Test.”

b. Issues and Findings

There were no findings identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed Temporary Alteration 00-0011, “Condensate O2 Temporary Injection,” to ensure that the modification did not affect the functions of the condensate system. The temporary modification included the installation of a temporary oxygen hose from an oxygen panel to the inlet of Condensate Pump C. This temporary modification was being used to evaluate the capability of the Condensate Pump C oxygen diffuser to keep oxygen in solution and enter the discharge of the condensate pumps without being removed by the normal suction barrel vent.

b. Issues and Findings

There were no findings identified.

1EP1 Exercise Evaluation (71114.01)

a. Inspection Scope

The inspectors reviewed the objectives and scenario for the 2000 exercise to determine if the exercise would acceptably test major elements of the emergency plan. The scenario included equipment and electrical power failures, a loss of reactor coolant, core damage, a radiological release, and a meteorological change to support demonstration of the licensee's capabilities to implement its emergency plan.

The inspectors evaluated exercise performance by focusing on the risk-significant activities of classification, notification, protective action recommendations, and assessment of offsite dose consequences in the following emergency response facilities:

- Simulator Control Room
- Technical Support Center
- Operations Support Center
- Emergency Operations Facility

The inspectors also assessed personnel recognition of abnormal plant conditions, the transfer of emergency responsibilities between facilities, communications, and the overall implementation of the emergency plan.

The inspectors attended the postexercise critiques in each of the above facilities to evaluate the initial licensee self-assessment of exercise performance. The inspectors also attended a subsequent presentation of critique items to plant management.

b. Issues and Findings

There were no findings identified.

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04)

a. Inspection Scope

The inspectors reviewed Revision 21 to the River Bend Station Emergency Plan, transmitted by the licensee on April 14, 2000, to determine if the revision decreased the effectiveness of the emergency plan.

b. Issues and Findings

There were no findings identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

a. Inspection Scope

The inspectors completed NRC Inspection Manual Procedure 71151, “Performance Indicator Verification,” to verify the accuracy and completeness of performance indicators involving transients per 7000 critical hours, emergency response organization drill/exercise performance, emergency response organization readiness, and alert and notification system reliability.

b. Issues and Findings

There were no findings identified.

OA5 Other

.1 (Closed) LER 50-458/9915-01: RHR Train C and reactor core isolation cooling operability affected by an unsealed wall penetration between their rooms in the auxiliary building. This event was discussed in NRC Inspection Report 50-458/99-15. No new issues were revealed by the LER.

.2 (Closed) LER 50-458/9916: Thermally induced accelerated corrosion of boiling water reactor fuel. This event was discussed in NRC Inspection Report 50-458/99-07. No new issues were revealed by the LER.

.3 (Closed) LER 50-458/0002: Inoperability of high pressure core spray diesel generator due to closure of Division I service water isolation valve. This event was discussed in NRC Inspection Report 50-458/00-02. No new issues were revealed by the LER.

.4 (Closed) LER 50-458/0003: Incorrectly assembled battery terminal. The inspectors reviewed this LER and determined that the issue is minor and warrants no additional inspection.

.5 (Closed) LER 50-458/0004: Automatic standby gas treatment system actuation due to annulus exhaust radiation monitor trip. This event was discussed in NRC Inspection Report 50-458/00-01. No new issues were revealed by the LER.

.6 (Closed) LER 50-458/0006: Noncompliance with Technical Specifications during control rod scram time testing due to procedure implementation error. This event was discussed in NRC Inspection Report 50-458/00-09. No new issues were revealed by the LER.

.7 (Closed) LER 50-458/0007: Unplanned automatic isolation of the reactor core isolation cooling system during surveillance testing. This event was discussed in NRC Inspection Report 50-458/00-09. No new issues were revealed by the LER.

4OA6 Exit Meeting Summary

The inspectors presented the inspection results of the emergency plan review by telephone to Mr. M. Bakarich, Emergency Planning Manager, and other members of licensee management at the conclusion of the inspection on May 4 and 22, 2000. The licensee acknowledged the findings presented.

The inspectors presented the emergency exercise inspection results to Mr. R. Edington, Vice President-Operations, River Bend Station, and other members of licensee management at the conclusion of the inspection on June 9, 2000. The inspectors discussed the recharacterization of one inspection issue with members of licensee management in telephone conversations on June 15 and 16, 2000. The licensee acknowledged the findings presented.

The inspectors presented the resident inspection results to Mr. D. Mims, General Manager, Plant Operations, and other members of licensee management at the conclusion of the inspection on June 26, 2000. The licensee acknowledged the findings presented.

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

ATTACHMENT 1

PARTIAL LIST OF PERSONS CONTACTED

Licensee

R. Azzarello, Manager, Training and Emergency Planning
M. Bakarich, Manager, Emergency Planning
R. Biggs, Coordinator, Licensing
E. Bush, Superintendent Operations
D. Dormady, Manager, Performance and System Engineering
R. Edington, Vice President-Operations
J. Fowler, Manager, Quality Assurance
H. Goodman, Superintendent, Reactor Engineering
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W. Mashburn, Manager, Engineering Programs and Components
J. McGhee, Manager, Operations
C. Miller, Superintendent, Composite Team
D. Mims, General Manager, Plant Operations
D. Myers, Senior Licensing Specialist, Nuclear Safety Assurance
A. Shahkarami, Manager System Engineering
P. Sicard, Manager, Safety Analysis
D. Wells, Superintendent, Radiation Protection
M. Wyatt, Manager, Planning and Scheduling/Outage

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

50-458/0010-01	NCV	Failure to specify postmaintenance requirements in maintenance packages (Section 1R19.1)
50-458/0010-02	NCV	Failure to perform functional test of containment unit cooler supply valve (Section 1R19.2)

Closed

50-458/9915-01	LER	Residual heat removal Train C and reactor core isolation cooling operability affected by an unsealed wall penetration (Section OA5.1).
50-458/9916	LER	Thermally induced accelerated corrosion of boiling water reactor fuel (Section OA5.2).

50-458/0002	LER	Inoperability of high pressure core spray diesel generator due to closure of Division I service water isolation valve (Section OA5.3).
50-458/0003	LER	Incorrectly assembled battery terminal (Section OA5.4).
50-458/0004	LER	Automatic standby gas treatment system actuation due to annulus exhaust radiation monitor trip (Section OA5.5).
50-458/0005	LER	Incorrectly connected motor leads for primary containment unit cooler service water supply valve (Section 1R19.2).
50-458/0005-01	LER	Incorrectly connected motor leads for primary containment unit cooler service water supply valve (Section 1R19.2).
50-458/0006	LER	Noncompliance with Technical Specifications during control rod scram time testing due to procedure implementation error (Section OA5.6).
50-458/0007	LER	Unplanned automatic isolation of the reactor core isolation cooling system during surveillance testing (Section OA5.7).

LIST OF ACRONYMS AND INITIALISMS USED

ATWS	anticipated transient without scram
CFR	Code of Federal Regulations
EDG	emergency diesel generator
LER	licensee event report
LOCA	loss of coolant accident
MAI	maintenance action item
PMT	postmaintenance testing
NCV	noncited violation
RHR	residual heat removal

LIST OF DOCUMENTS REVIEWED

Calculations

12210-IA-E22*1, Verification of Setpoint for Relief Valve 1E22*RVF014
12210-IA-E22*2, Verification of Setpoint for Relief Valve 1E22*RVF035

Condition Reports

CR 1999-1514	Diesel generator system classified as 10 CFR 50.65(a)(1)
CR 2000-0192	Fuel oil level transmitter out of tolerance
CR 2000-0232	Diesel generator heater control switch found in off position
CR 2000-0437	Additional washer installed on battery post
CR 2000-0736	Service water valve breaker incorrectly wired
CR 2000-0739	Standby Service Water system cross-divisional leakage
CR 2000-0875	Diesel generator lube oil cooler leak
CR 2000-0911	PMT concerns
CR 2000-1010	PMT process concerns

Condition Reports initiated between May 7 and June 24, 2000

Maintenance Rule

Maintenance rule database for emergency diesel generator systems

Plant Procedures

ADM-0031	Temporary Alterations, Revision 9
ADM-0076	Verification Program, Revision 3
ADM-0092	Foreign Material Exclusion, Revision 1B
COP-1050	Post-Accident Estimation of Fuel Core Damage, Revision 5
EIP-2-001	Classification of Emergencies, Revision 10
EIP-2-002	Classification Actions, Revision 20
EIP-2-006	Notifications, Revision 27
EIP-2-007	Protective Action Recommendations, Revision 17
EIP-2-012	Radiation Exposure Controls, Revision 13
EIP-2-016	Operations Support Center, Revision 16
EIP-2-018	Technical Support Center, Revision 20
EIP-2-020	Emergency Operations Facility, Revision 22
EIP-2-501	Emergency Facilities and Equipment Readiness, Revision 12
EIP-2-502	Emergency Communications Equipment Testing, Revision 18
EIP-2-701	Prompt Notification Monthly System Testing, Revision 12
EIP-2-703	Performance Indicators, Revision 0
EP - 13.3	River Bend Station Emergency Plan, Revision 20
EP - 13.3	River Bend Station Emergency Plan, Revision 21
ENG-3-041	ASME Section XI Inservice Testing Program
GOP-0001	Plant Startup, Revision 31
NDE-10.02	VT-2 Inspections, Revision 0
OSP-0042	ASME Section XI Inservice Testing Implementation, Revision 3
SOP-0049	125 VDC System
SOP-0053	Standby Diesel Generator and Auxiliaries

Fire Hazards Analysis

Maintenance Planning Guideline

Weekly Maintenance Schedules

River Bend Online Maintenance Guidelines

Flooded Pool Foreign Material Exclusion Guidelines

ATTACHMENT 2

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the significance determination process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, or RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.