UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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ARLINGTON, TEXAS 76011-8064

July 24, 2002

Paul D. Hinnenkamp, 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 50-458/02-02

Dear Mr. Hinnenkamp:

On June 29, 2002, the NRC completed an inspection at your River Bend Station. The enclosed report documents the inspection findings which were discussed on July 3, 2002, with you and other members of your staff.

This 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. Within these areas, the inspection consisted of selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, the NRC has identified three issues that were evaluated under the risk significance determination process as having very low safety significance (Green).

The NRC has increased security requirements at River Bend Station in response to terrorist acts on September 11, 2001. Although the NRC is not aware of any specific threat against nuclear facilities, the NRC issued an Order and several threat advisories to commercial power reactors to strengthen licensees' capabilities and readiness to respond to a potential attack. The NRC continues to monitor overall security controls and will issue temporary instructions in the near future to verify by inspection the licensee's compliance with the Order and current security regulations.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response will be made 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/reading-rm/adams.html (the Public Electronic Reading Room).

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

Sincerely,

/RA/

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

Docket: 50-458
License: NPF-47

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NRC Inspection Report
50-458/02-02

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-458
License:	NPF-47
Report:	50-458/02-02
Licensee:	Entergy Operations, Inc.
Facility:	River Bend Station
Location:	5485 U.S. Highway 61 St. Francisville, Louisiana
Dates:	March 31 through June 29, 2002
Inspectors:	P. J. Alter, Senior Resident Inspector M. O. Miller, Resident Inspector M. E. Murphy, Senior Reactor Engineer, Operations Examiner P. C. Gage, Senior Reactor Engineer, Operations Examiner R. E. Lantz, Senior Emergency Preparedness Inspector G. F. Larkin, Resident Inspector, Waterford
Approved By:	D. N. Graves, Chief, Project Branch B
ATTACHMENT:	Supplemental Information

SUMMARY OF FINDINGS

River Bend Station
NRC Inspection Report 50-458/02-02

IR 05000458-02-02; on 03/31/2002-06/29/2002; Entergy Operations, Inc; River Bend Station. Integrated Resident & Regional Report. Maintenance Risk and Control of Emergent Work, Operability Evaluations, Refueling and Other Outage Activities. Three Green Findings.

The inspections were conducted by the resident inspectors, regional operations examiners, and a regional emergency preparedness inspector. The inspectors identified three Green findings. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, "Significance Determination Process". Findings for which the Significance Determination Process does not apply 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 station blackout diesel generator was found to be inoperable by the licensee because its starting battery had been allowed to completely discharge. The station blackout diesel generator had been moved from its normal storage location as a contingency for a planned maintenance outage on several Division I safety-related systems. The inspectors determined that the Division I maintenance outage contingency plan and the weekly work schedule did not plan for the return of the station blackout diesel generator to its normal storage location to re-energize its battery charger. The licensee determined that this is a repeat of a similar event of April 4, 1998, documented in Condition Report CR-RBS-1998-0384.

  The failure to maintain its starting battery charged caused the risk significant station blackout diesel generator to be inoperable and unavailable. The inspectors, using the significance determination process, determined that the safety significance of the unavailability of the station blackout diesel generator was very low because the length of time the diesel generator was unavailable was less than 24 hours and all other electrical systems were available during that time. This human performance error was documented is the licensee's corrective action program as Condition Report CR-RBS-2002-0664 (Section 1R13).

- Green. Following maintenance performed on May 9, 2002, to determine the source of a leak from the Division 1 emergency diesel generator jacket cooling water system, the leak rate more than doubled. The licensee's attempt to correct the problem on May 30, 2002, resulted in another increase in the leak rate to the point that makeup to the jacket cooling water system would be required within approximately 2 hours of Division I emergency diesel generator operation during a loss of offsite power. Although, the cause for the increased jacket water leak was repaired on June 4, 2002, the diesel generator remained degraded, but operable. The licensee planned to repair the original leak during the next extended diesel generator maintenance outage.

The inspectors determined that the increased leak rate was beyond the licensee's evaluation that concluded that the Division 1 emergency diesel generator was degraded but operable. If left uncorrected, the jacket cooling water leak could have caused the emergency diesel generator to become inoperable and unavailable. The normal source of makeup water would not have been available during a loss of offsite power and the licensee did not develop a written procedure for use of an alternate makeup source until May 30, 2002. Using the significance determination process, the risk significance of the finding was determined to be very low because the emergency diesel generator remained operable, although degraded. This maintenance induced problem was documented in the licensee's corrective action program as Condition Report CR-RBS-2002-0672 (Section 1R15).

- Green. Following a planned reactor scram during a plant shutdown, operators failed to take manual control of the feedwater level control system in time to stop an unexpected rise in reactor water level until after the running reactor feed pump tripped on high reactor water level. The licensee determined that the reduction of the reactor pressure control setpoint and subsequent opening of the main turbine bypass valves caused a "swell" in reactor water level which contributed to the higher than expected reactor water level transient. The inspectors determined that the operators did not manually close and isolate one of the two automatic feedwater regulating valves in time to eliminate leakage past the feedwater regulating valve and failed to reject water from the reactor through the reactor water cleanup system in time to stop the rise in reactor water level to the high level trip of the reactor feed pump.

  The failure of the operators to manually control reactor water level resulted in the unavailability of a risk-significant reactor feed pump. The inspectors, using the significance determination process, determined that the safety significance of the high reactor water level trip of the running reactor feed pump following a planned reactor scram was very low because the reactor feed pump was restarted from the main control room as soon as reactor water level was lowered, the high reactor water level trip signal was cleared, and other reactor water makeup sources remained available. This human performance error was documented in the licensee's corrective action program as Condition Report CR-RBS-2002-0688 (Section 1R20).

Report Details

Summary of Plant Status: The reactor was operated at 100 percent power from the beginning of the inspection period until shutdown on May 12, 2002, for Planned Outage 02-01, to investigate and repair leaking valves in the drywell. On May 15, 2002, the reactor was started and attained 100 percent power on May 19, 2002. The plant operated at 100 percent power throughout the remainder of the inspection period.

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

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

During the week of April 8, 2002, the inspectors reviewed the licensee's implementation of plant procedures to protect mitigating systems from thunderstorms and heavy rain conditions. Specifically the inspectors: (1) verified that selected systems and components would remain functional when challenged by thunderstorm weather conditions; (2) verified that thunderstorm weather conditions such as river level and access to intake structure are monitored; (3) verified that plant features for operation of the ultimate heat sink during thunderstorm weather conditions are appropriate; and (4) evaluated implementation of the thunderstorm weather preparation procedures and compensatory measures for affected systems or components before the onset of and during thunderstorm weather conditions. The inspectors reviewed abnormal operating Procedure AOP-0046, "Severe Weather Operation," Revision 14.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

.1 Standby Service Water System Walkdown

On May 6, 2002, the inspectors performed a partial system walkdown of the standby service water system, as a backup for the fire protection water system, while that system was out of service for planned maintenance. The inspectors reviewed system operating Procedure SOP-0037, "Fire Protection Water System," Section 5.3, "Fire Protection Water Supply via Standby Service Water," Revision 19, to determine the correct system lineup. The inspectors walked down critical portions of the system to identify any discrepancies between the existing equipment lineup and the correct lineup.

.2 Fire Protection Water System Walkdown

On May 7, 2002, the inspectors performed a partial system walkdown of the fire protection water system following restoration of system Train A, following planned maintenance, while Train B remained out of service. The inspectors reviewed system operating Procedure SOP-0037, “Fire Protection Water System,” Revision 19, to determine the correct system lineup. The inspectors walked down critical portions of the system to identify any discrepancies between the existing equipment lineup and the correct lineup.

.3 Residual Heat Removal System Walkdown

On May 15, 2002, the inspectors performed a partial system walkdown of residual heat removal Train A after it was secured from shutdown cooling mode and returned to standby mode for low pressure coolant injection. The inspectors reviewed system operating Procedure SOP-0031, “Residual Heat removal System,” Revision 38, to determine the correct system lineup. Then the inspectors walked down critical portions of the system to identify any discrepancies between the existing equipment lineup and the correct lineup.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

.1 Fire Protection Area Walkdowns

The inspectors toured six plant areas important to reactor safety to observe conditions related to: (1) licensee control of transient combustibles and ignition sources; (2) the material condition, operational lineup, and operational effectiveness of fire protection systems, equipment and features; and (3) the material condition and operational status of fire barriers used to prevent fire damage or fire propagation. In addition, the inspectors walked down the areas covered by continuous fire watches during the planned maintenance outage of the fire protection water system on May 6, 2002.

- Control building 116 foot elevation, Fire Zone C-24, April 3, 2002
- Division I engineered safety features switchgear room, Fire Zone C-15, April 12, 2002
- Division III engineered safety features switchgear room, Fire Zone C-22, April 12, 2002
- Control building 70 foot elevation, Fire Zone C-1A, May 7, 2002
- Diesel-driven fire Pump A room, Fire Zone FP-1, May 7, 2002

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

- Pre-Fire Strategy Book
- Updated Safety Analysis Report (USAR), Section 9A.2, “Fire Hazards Analysis”
- River Bend postfire safe shutdown analysis

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

On April 8, 2002, the inspectors conducted an external flooding assessment to verify that the licensee’s flooding mitigation plans and equipment were consistent with design requirements and risk analysis assumptions. The inspectors conducted a walkdown of the owner controlled area outside the protected area. Specifically, the inspectors examined: (1) routing and capacity of drainage trench and pipe systems during heavy rains with the Mississippi River at or above flood levels, (2) capability of the drainage trench and pipe systems to direct flood waters away from plant structures and mitigating systems, (3) ability of the licensee to correctly assess flooding situations, and (4) ability of the licensee to maintain access to the intake structure during flooding. The inspectors reviewed the following documents during the inspection:

- River Bend individual plant examination of external events
- USAR Section 3.4.1, “Flood Protection”
- Abnormal Operating Procedure, AOP-0029, "Severe Weather Operation," Revision 14

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors reviewed the method and results of residual heat removal heat Exchanger A testing performed on March 22, 2002. The inspectors reviewed the performance testing results in performance engineering Procedure PEP-0239, “Performance Monitoring Program for the Residual Heat Removal Heat Exchangers E12-EB001A and E12-EB001C,” Division I, Revision 2, Attachment 12, “Residual Heat Removal Heat Exchangers E12-EB001A and E12-EB001C Heat Transfer Capacity

Verification." As part of the inspection, the inspectors verified: (1) selected test methodology consistent with Electric Power Research Institute NP 7552, "Heat Exchanger Performance Monitoring Guidelines," December 1991, (2) test acceptance criteria and results appropriately considered differences between testing and design conditions, (3) frequency of testing and inspection modified appropriately to detect further degradation prior to loss of heat removal capabilities below design basis values, (4) test conditions consistent with the selected methodology and procedural requirements, (5) test acceptance criteria and results consistent with the design basis values, (6) test results considered test instrument inaccuracies and differences, and (7) acceptance criteria developed for bio-fouling controls.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

1. Biennial Licensed Operator Requalification Evaluation

a. Inspection Scope

During the week of May 13, 2002, operator performance since the last requalification program evaluation was assessed to determine if performance deficiencies have been addressed through the requalification training program. The inspection evaluated licensed operator performance in mitigating the consequences of events, since poor licensed operator performance results in increased risk through increased operator recovery rates and licensed personnel-induced common-cause error rates assumed in the licensee's individual plant examinations. This inspection effort of the licensed operator requalification program included the following major areas: (1) facility operating history, (2) requalification written examinations and operating tests, (3) training feedback system, (4) licensee remedial training program, and (5) conformance with operator license conditions.

Examination security measures and procedures were evaluated for compliance with 10 CFR 55.49. The licensee's sample plan for the written examinations was evaluated for compliance with 10 CFR 55.59 and NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 8, as referenced in the facility requalification program procedures. In addition, the inspectors: (1) reviewed the number of applicants and pass/fail results of the written examinations, individual operating tests, and simulator operating tests; (2) interviewed personnel regarding the policies and practices for administering examinations; (3) observed the administration of three dynamic simulator scenarios to one requalification crew by facility evaluators; (4) observed a facility evaluator administer two in-plant job performance measures to five licensed operators; and (5) observed a facility evaluator administer four simulator job performance measures in the control room simulator in a dynamic mode to seven licensed operators.

The inspectors reviewed the licensee's process for revising and maintaining an up-to-date licensed operator continuing training program, including the use of feedback from plant events and industry experience information.

The inspectors verified the adequacy and effectiveness of the remedial training conducted since the last requalification examinations and the training planned for the current examination cycle to ensure that identified licensed operator or crew performance weaknesses during training and plant operations were addressed. Remedial training and examinations for examination failures were reviewed for compliance with facility procedures and responsiveness to address areas failed. The inspectors also reviewed the remediation documented for seven individuals, which involved two written examination failures, five job performance measure failures, and one simulator examination failure by one crew.

Maintenance of license conditions was evaluated for compliance with 10 CFR 55.53 by review of facility records, procedures, and tracking systems for licensed operator training, qualification, and watchstanding.

b. Findings

No findings of significance were identified.

2. Quarterly Licensed Operator Requalification Program Inspection

a. Inspection Scope

On June 19, 2002, the inspectors observed simulator training of an operating crew, as part of the operator requalification training program, to assess licensed operator performance and the training evaluator's critique. Emphasis was placed on observing evaluation of high risk licensed operator actions, operator activities associated with the emergency plan, and lessons learned from industry and plant experiences. The inspectors reviewed simulator training Scenario RBS-1-SIM-SMS-00619.01, "Loss of High Pressure Feed, ATWS, Fuel Failure, Leak in the Drywell, Radioactive Release," dated February 23, 2002. In addition, the inspectors compared simulator control panel configurations with the actual control room panels for consistency, including recent modifications implemented in the plant.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

The inspectors reviewed five structure, system, or component (SSC) performance problems to assess the effectiveness of the licensee's maintenance efforts for SSCs scoped under the licensee's maintenance rule program. The inspectors verified the licensee's implementation of the maintenance rule, 10 CFR 50.65, for the performance problems reviewed by answering the following questions: (1) was the SSC scoped for monitoring in accordance with 10 CFR 50.65; (2) was the SSC assigned the proper safety significance; (3) were the problems characterized properly; (4) as a result of the problems, was the SSC assigned the proper classification under 10 CFR 50.65; and (5) were the appropriate performance criteria established for the SSC or, when necessary, were appropriate goals set and corrective actions taken to restore the SSC status under the maintenance rule? The following performance problems were evaluated:

- Condition Report (CR) CR-RBS-2002-0437, Failure of the low pressure core spray minimum flow valve to close, reviewed April 8, 2002
- CR-RBS-2002-00667, Division II emergency diesel generator inoperable due to lube oil keep warm heater breaker trip, reviewed June 19, 2002
- CR-RBS-2002-0684, Station blackout diesel generator batteries found discharged and inoperable, reviewed June 20, 2002
- CR-RBS-2002-0688, High level trip of reactor feed pump following manual scram from 26 percent reactor power, reviewed June 18, 2002
- CR-RBS-2002-00787, Division I emergency diesel generator jacket water leaks, reviewed June 19, 2002

The following documents were reviewed as part of this assessment:

- NUMARC 93-01, "Nuclear Energy Institute Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 2
- River Bend maintenance rule function list
- River Bend maintenance rule performance criteria list

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

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

.1 Risk Assessment and Management of Risk

The inspectors verified performance of risk assessments, in accordance with administrative Procedure ADM-096, "Risk Management Program Implementation and On-Line Maintenance Risk Assessment," Revision 01, for planned maintenance activities and emergent work involving SSCs within the scope of the Maintenance Rule. Specific work activities evaluated included planned and emergent work for the week of June 10, 2002; station fire water system outage on May 5 and 6, 2002; and Division I standby service water, low pressure emergency core cooling, emergency diesel generator outage on May 7, 8 and 9, 2002.

.2 Emergent Work Control

During emergent work, the inspectors verified that the licensee took actions to minimize the probability of initiating events, maintained the functional capability of mitigating systems, and maintained barrier integrity. The inspectors also reviewed the emergent work activities to ensure the plant was not placed in an unacceptable configuration. Specific emergent work activities evaluated included:

- Diesel-driven instrument air compressor Dryer IAS-DRY4 maintenance during the week of April 8, 2002
- Station blackout diesel generator restoration on May 11, 2002
- Division I emergency diesel generator jacket cooling water leak reduction work performed on May 30, 2002

b. Findings

On May 11, 2002, as part of his normal watchstanding duties, an operator discovered that the station blackout diesel generator was inoperable due to its starting battery being discharged. The diesel generator had been moved from its normal storage location on May 6, 2002, as part of a contingency plan for a major maintenance outage on the Division I engineered safety features systems. The same operator checks were performed on the previous day, so the diesel generator was determined to have been inoperable for less than 24 hours. The station blackout diesel generator was risk-significant for a loss of offsite power and a failure of the emergency diesel generators, i.e., station blackout. The inspectors determined that the finding had very low risk significance (Green).

Following the discovery of the discharged batteries, the station blackout diesel generator was returned to its normal storage location, the batteries were replaced, and the battery charger was re-energized. The licensee's root cause analysis determined that the battery charger was left de-energized for over 4 days due to lack of specificity in system operating Procedure, SOP-0054, “Station Blackout Diesel Generator,” Revision 3, and that there was no specific procedure step to ensure the battery charger was re-energized when the diesel generator was moved away from its normal storage location. In addition, the inspectors determined that work planning for the Division I maintenance outage and the operations department’s contingency plan provided no guidance to return the diesel generator to its normal storage location following the maintenance outage work.

The licensee determined that this event was a repeat of another event when the station blackout diesel generator had to be declared inoperable on April 4, 1998. At that time, the diesel generator was moved to support planned maintenance on other plant equipment and the battery charger was not re-energized, causing the starting battery to become discharged. The corrective actions for CR-RBS-1998-0384 were to add a precaution and procedure steps to the station blackout diesel generator system procedure to ensure power was restored to the battery charger whenever the diesel engine was moved from its normal storage location. The licensee determined that these corrective actions did not prevent the recurrence of the similar event on May 11, 2002.

The inspectors determined that the finding was more than minor in that it effected the operability and availability of the risk-significant station blackout diesel generator. The inspectors evaluated the finding using inspection manual Chapter 0609, “Significance Determination Process,” Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations.” The inspectors determined that the failure to maintain the station blackout diesel generator operable was of very low safety significance (Green) because of the short time (less than 24 hours) that it was inoperable and the availability of all other electrical systems. This human performance error was entered into the licensee’s corrective action program as CR-RBS-2002-0664.

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

a. Inspection Scope

High Level Trip of Reactor Feed Pump during a Manual Scram Recovery

The inspectors observed and reviewed personnel performance during the manual reactor scram from 26 percent power at the end of the plant shutdown for Planned Outage 02-01 on May12, 2002. During the scram recovery, reactor water level unexpectedly rose high enough to trip the running reactor feed pump. The inspectors reviewed the procedures used by the operators during the event and evaluated the root cause analysis and human performance error review of the event as documented in CR-RBS-2002-0688. In addition, the inspectors reviewed operator logs, plant computer data, and strip charts to determine what occurred and that operators responded in accordance with plant procedures and training. For more details, see Section 1R20.

b. Findings

See report Section 1R20.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed eight operability evaluations performed by the licensee for risk significant systems to determine that operability was justified, such that availability was assured, and no unrecognized increase in risk had occurred. Specific areas evaluated included: (1) the technical adequacy of the evaluation; (2) whether other existing degraded conditions were considered; and (3) if operability was based on compensatory measures, were these measures in place and would they work. The inspectors also reviewed Nuclear Procedure RBNP-078, "Operability Determinations," Revision 6.

- CR-RBS-2002-0437, Low pressure core spray minimum flow valve failure to close
- CR-RBS-2002-0526, Division I emergency diesel generator governor oil sightglass cracked
- CR-RBS-2002-0611, Station blackout diesel generator standby switch alignment
- CR-RBS-2002-0620, Technical Specification safety related degraded voltage relay maximum voltage and time delay values
- CR-RBS-2002-0643, Evaluation of validity of reactor core isolation cooling system turbine speed test acceptance criteria
- CR-RBS-2002-0645, reactor core isolation cooling system line fill pump discharge check Valve E51-VF061 failed functional testing
- CR-RBS-2002-0667, Loss of Division II emergency diesel generator standby lube oil keep warm heater
- CR-RBS-2002-0672, Division I emergency diesel generator jacket cooling water leak

b. Findings

On May 9, 2002 the licensee performed maintenance on the Division I emergency diesel generator jacket cooling water system. Upon reassembly of the system, a pre-existing leak in the jacket cooling water system was made worse. From May 10-29, 2002, the licensee did not monitor the change in the leak rate from the jacket cooling water system or have in place a written contingency plan to make up to the jacket cooling water system during a loss of offsite power, the design basis event for the emergency diesel generators. On June 4, 2002, the licensee repaired the cause of the excessive leakage. The inspectors determined that the operable, but degraded, condition of the Division I emergency diesel generator for a period of 19 days was of very low safety significance (Green).

On May 9, 2002, the licensee disassembled the Division I emergency diesel generator exhaust shroud to find the location of a possible leak in the jacket cooling water system from the exhaust shroud. The licensee determined that the leak was sufficiently small to declare the Division I emergency diesel generator “operable but degraded” in accordance with guidance provided in NRC Generic Letter 91-18, “Information to licensees regarding NRC Inspection Manual Section on Resolution of Degraded and Non-Conforming Conditions,” Revision 1. USAR Section 9.5.5.1, “Diesel Generator Cooling Water System,” states in part that “no makeup needs are anticipated for 7 days of continuous operation at rated power.” On May 10, 2002, engineering analysis indicated that the leak would require makeup to the jacket cooling water system within 16.5 hours of fully loaded operation during a loss of offsite power. During a loss of offsite power, the normal source of makeup water to the emergency diesel generators would not be available and operators would have to provide makeup water to the jacket cooling water system from another system. At the time, the operations manager determined that, because of the short time needed and personnel that would be available (station emergency response organization) during a prolonged loss of offsite power, there was no need to provide written procedural guidance for makeup to the emergency diesel generator jacket cooling water system during a loss of offsite power.

During the period of time from May 10-29, 2002, the operators were logging each addition of water to the Division I jacket cooling water system but were not trending this addition rate compared to the addition rate prior to May 9, 2002. On May 29, 2002, the inspectors questioned the increased addition rate to the jacket cooling water system over the past week. An engineering evaluation of the leak rate, which had more than doubled, indicated that make up to the jacket cooling water system would be required within 4.63 hours. Operators again determined that this condition maintained the Division I emergency diesel generator operable but degraded but no specific procedural guidance was required to make up water to the jacket cooling water system during a loss of offsite power. On May 30, 2002, the licensee attempted to readjust the torque on bolts used to hold the jacket cooling water cylinder return header piping to the top of the exhaust shroud. As a result, the leak rate from the exhaust shroud increased to the point where makeup would be required after approximately 2 hours of full load operation during a loss of offsite power. At that time, operators raised the normal operating level in the Division I emergency diesel generator jacket cooling water standpipe and initiated a procedure change to alarm response Procedure ARP-EGS*PNL3A/D-3, “Jacket Cooling Water Level Low,” Revision 14, that described in detail a method of providing makeup water to the jacket cooling water system from the standby service water system during a loss of off site power. On June 4, 2002, the licensee rewelded the bolting pads for the Division I emergency diesel generator jacket cooling water cylinder return header to the exhaust shroud. Operators evaluated the leak rate after that maintenance and determined that it was lower than prior to May 9, 2002. The remaining Division I emergency diesel generator jacket cooling water system leak from the exhaust shroud was scheduled to be repaired during the next extended maintenance outage for the Division I emergency diesel generator.

The inspectors determined that the increased leak rate from the jacket cooling water system caused the Division I emergency diesel generator to be further degraded, but still operable, beyond the existing licensee’s analysis that concluded that the diesel generator was operable from May 10-29, 2002. The increased leak rate and lack of a proceduralized method for makeup to the jacket water system increased the risk for a failure of the Division I emergency diesel generator during a loss of offsite power. The inspectors determined that the finding was more than minor in that, if the condition was left uncorrected, it could deteriorate and effect the availability and operability of the Division I emergency diesel generator. The inspectors evaluated the finding using inspection manual Chapter 0609, "Significance Determination Process," Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." The Division I emergency diesel generator was in a degraded condition because of the leak in the jacket cooling water system, but the diesel generator remained operable. Therefore, the inspectors determined that the maintenance-induced increase in the jacket water system leakage was of very low safety significance (Green). This maintenance induced problem was entered into the licensee's corrective action program as CR-RBS-2002-0672.

1R16 Operator Workarounds (IP 71111.16)

a. Inspection Scope

An operator workaround is defined as a degraded or nonconforming condition that complicates the operation of plant equipment and is compensated for by operator action. On June 19, 2002, the inspectors reviewed the required operator actions necessary to control reactor water level following a reactor scram from low power levels to determine if the functional capability of any mitigating system or human reliability in responding to an initiating event, a reactor scram, was affected. Specifically the inspectors evaluated the effect of these required actions on the operator's ability to control reactor water level following a scram. For more details, see Section 1R20.

b. Findings

See report Section 1R20.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

The inspectors reviewed the permanent modification made to station blackout Valve SWP-AOV599 solenoid-operated control valves. The inspectors verified that the design basis, licensing basis, and performance capabilities of the station blackout valve and the standby service water system had not been degraded. The inspectors also verified that the performance of the modification during at-power operations did not place the plant in an unsafe condition. The inspectors reviewed ER-RB-2001-0470-000, "Upgrade of SWP-SOV602A, B & C and SOV601," February 4, 2002. Specifically the inspectors: (1) evaluated the design adequacy of the modification; (2) verified that the modification preparation, installation, and testing did not interfere with safe operation of the plant; (3) verified that the postmodification testing verified the operability of the plant component and system; and (4) verified that plant design documents and affected plant procedures were updated.

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 four maintenance action items to ensure that testing activities were adequate to verify system operability and functional capability.

- MAI 356961, Rework and adjust valve position switches for reactor sample Valve SSR-SOV130
- MAI 357540, Refurbish control room air conditioning unit Fan HVC-ACU1B
- MAI 357836, Replace reactor protection system Relay C71A-K67
- MAI 356948, 356949, 356950, control rod drive hydraulic control unit refurbishment

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors observed licensee outage planning and execution activities for Planned Outage 02-01. The inspectors' review included scheduling, training, outage configuration management, and reactivity controls. Specific activities monitored included:

- Plant shutdown and planned reactor scram on May 12, 2002
- Drywell inspection and closeout on May 14, 2002
- Plant safety review committee meeting to approve startup on May 14, 2002
- Portions of the reactor startup on May 15, 16 and 17, 2002

b. Findings

On May 12, 2002, the operating crew shut down the reactor for Planned Outage 02-01. The operators manually scrammed the reactor as planned at 26 percent reactor power. Shortly after the scram, reactor water level rose more than expected and the operating reactor feed pump tripped due to high reactor water level. The inspectors were present in the control room at the time and observed the operating crew’s response to the scram and their actions to control reactor water level. After interviewing the operating crew members and control room supervisor, the inspectors determined that the reason for the high level trip of the reactor feed pump was failure of the operators to manually control reactor water level following a scram. The inspectors determined that the high level trip of the running reactor feed pump was a finding of very low risk significance (Green).

On March 25, 2002, the operating crew attended simulator training on reactor level response and control following a reactor scram. The inspectors reviewed simulator training Scenario RBS-1-SIM-STG-40208.00, “Critical Parameter Control during Emergency Conditions,” dated March 7, 2002. This training incorporated procedural guidance from several plant procedures to control reactor water level following a scram by the addition of water to the reactor and the use of the reactor water cleanup system to remove water from the reactor. During the event of May 12, 2002, although the operators took manual control of one of the feedwater regulating valves, closed the valve, and closed the feedwater regulating valve blocking valve, as required by system operating Procedure SOP-009, “Reactor Feedwater System,” Revision 25, the steps were not performed in time to prevent the feedwater pump trip on high reactor water level. In addition, although reactor water cleanup was lined up to reject water from the reactor in accordance with system operating Procedure SOP-090, “Reactor Water Cleanup System,” Revision 28, reject flow was not established until just before the reactor feed pump tripped.

During their root cause analysis, the licensee determined that the time delay relay that controls the level setpoint setdown control circuit for the feedwater level control system may have caused the feedwater regulating valves to remain open longer than expected, adding extra water to the reactor vessel. During subsequent testing of the time delay relay, the licensee determined that the time delay of the relay varied considerably. Technicians replaced the relay on July 2, 2002.

The inspectors determined that the procedural guidance to take manual control of the feedwater regulating valve and close it and its blocking valve indicated that there was excessive leakage past the feedwater regulating valves. The feedwater regulating valve leakage contributed to the high reactor water level condition. This extra operator burden during the response to a planned reactor scram contributed to the reactor water level rising more than expected and the unplanned trip of the running reactor feed pump.

On May 2, 2002, the operating crew attended “just-in-time” training for the planned reactor shutdown. This training concentrated on the reactor scram and establishing an acceptable cooldown rate following the scram. At that time, the control room supervisor recommended that the operators reduce the reactor pressure control setpoint to follow the normal reactor pressure decrease pressure following a scram. The objective was to control reactor pressure at approximately 800 psig immediately after the scram to reduce the time needed to cool down the reactor. During the event of May 12, 2002, this action taken by the operating crew caused the main turbine bypass valves to open when pressure setpoint was lowered below actual reactor pressure. The resultant 6-inch “swell” in reactor water level occurred at 40 inches and did not contribute to the high level trip of the reactor feed pump at 51 inches, since water level returned to below 45 inches before continuing to rise at its previously established rate. Prior to the high level trip of the reactor feed pump, the turbine bypass valves opened automatically at the reduced pressure setpoint. The inspectors determined that this bypass valve opening did not contribute to the trip of the reactor feed pump. However, the inspectors determined that the reduction of the reactor pressure control setpoint was contrary to the intent of emergency operating Procedure EOP-1, “RPV Control,” Revision 16, step RP-3, to “Stabilize pressure . . .” before reducing pressure (step RP-4) or commencing a cooldown (step RP-5). The basis for EOP-1, step RP-3, was “RPV pressure is stabilized to facilitate control of RPV water level and reactor power.”

During the operations department human performance review and root cause analysis of the event, the licensee determined that time/schedule pressure was not a contributor to the high level trip of the reactor feed pump. Based on observation of the crew for the 2 hours leading up to the manual scram, the inspectors determined that there was schedule pressure on the crew prior to the scram. This observation was based on the following: (1) the control room supervisor participated directly in evolutions performed by the reactor operators, such as peer checking and system procedure place keeping; (2) the control room supervisor’s stated purpose for the reduction in reactor pressure control setpoint was to cut one half hour from the time required for the reactor cooldown; (3) the control room supervisor had, and used as a guide, a shutdown sequence document generated specifically for this plant shutdown with significant plant evolutions to be performed at various plant conditions compared to expected time of completion and a graph of reactor power verses expected time; and (4) two phone calls came into the control room from the outage control center with the message that they were one half hour behind the expected time line for the shutdown.

The inspectors determined that the finding was more than minor in that it effected the operability and availability of the feedwater and condensate systems. The inspectors evaluated the finding using inspection manual Chapter 0609, “Significance Determination Process,” Appendix A “Significance Determination of Reactor Inspection Findings for At-Power Situations.” The inspectors determined that the operator’s failure to operate the feedwater level control system promptly in accordance with station procedures resulted in the high reactor water level trip of the running reactor feed pump and was of very low safety significance (Green) because the pump was immediately available for restart when level was reduced and all other reactor makeup systems remained functional. This human performance error was entered into the licensee’s corrective action program as CR-RBS-2002-0688.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors verified, by observing and reviewing test data, that four selected risk significant systems and component surveillance tests met Technical Specification, USAR, and procedure requirements. The inspectors ensured that the surveillance tests demonstrated that the systems were capable of performing their intended safety functions and provided operational readiness. The inspectors specifically evaluated surveillance tests for preconditioning, clear acceptance criteria, range, accuracy, and current calibration of test equipment and verified that equipment was properly restored at the completion of the testing. The inspectors reviewed and/or observed the following surveillance tests and documents:

- STP-610-3827, "Reactor Plant Sampling Penetration, KJB-Z601B, Leak Rate Test," Revision 10A, of reactor recirculation system sample inboard isolation Valve SSR-SOV130, performed on April 7, 2002
- Repetitive Task 228, Periodic load test of station blackout diesel generator, performed on April 26, 2002
- STP-052-3701, "Control Rod Scram [Time] Testing," performed on May 12, 2002
- MCP-4303, "Functional Test of Standby Cooling Tower Station Blackout Division I Standby Service Water Return Valve and Valve Logic (SWP-AOV-599)," Revision 0, as-found test performed June 12, 2002

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

On April 23, 2002, the inspectors reviewed the temporary modification made to the main turbine control system to disable the turbine backup speed sensor circuit to allow for troubleshooting and repairs. Specifically the inspectors: (1) reviewed the temporary modification and its associated 10 CFR 50.59 screening against the system design basis documentation, including the USAR and Technical Specifications; (2) verified that the installation of the temporary modification was consistent with the modification documents; and (3) reviewed the postinstallation test results to confirm the actual impact of the temporary modification on the affected system had been adequately verified.

b. Findings

No findings of significance were identified.

Emergency Preparedness

1EP1 Exercise Evaluation (71114.01)

a. Inspection Scope

During the week of June 10, 2002, the inspectors reviewed the objectives and scenario for the 2002 Biennial Emergency Preparedness Exercise to determine if the exercise would acceptably test major elements of the emergency plan. The scenario included reactor protection system problems, equipment and electrical power failures, an unisolable steam leak and containment breach, fuel damage, and a radiological release to demonstrate the licensee's capabilities to implement the 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 simulator control room and the following emergency response facilities:

- 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, protection of emergency workers, emergency repair capabilities, and the overall implementation of the emergency plan to verify compliance with the requirements of 10 CFR 50.47(b), 10 CFR 50.54(q), and Appendix E to 10 CFR Part 50.

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

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

During the week of June 10, 2002, the inspectors reviewed Revision 25 to the River Bend Station Emergency Plan to determine if the revision decreased the effectiveness of the emergency plan.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors observed the emergency preparedness simulator training exercise conducted on June 19, 2002, to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation development activities. The inspectors also evaluated the licensee assessment of classification, notification, and protective action recommendation development during the exercise in accordance with plant procedures and NRC guidelines. The following procedures and documents were reviewed during the assessment:

- EIP-2-001, "Classification of Emergencies," Revision 11
- EIP-2-006, "Notifications," Revision 29
- EIP-2-007, "Protective Action Guidelines Recommendations," Revision 18
- RBS-1-SIM-SMS-0526.01, "Radiological Release," simulator training scenario, May 23, 2001.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

The inspectors verified the accuracy and completeness of the data used to calculate and report performance indicator data for the last quarter of 2000 and the first quarter of 2001. The inspectors used Nuclear Energy Institute 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 2, as guidance and interviewed licensee personnel responsible for compiling the information. The following performance indicators were reviewed:

- Safety System Unavailability - Residual Heat Removal System
- Unplanned Scrams per 7,000 Critical Hours
- Scrams with a Loss of Normal Heat Removal
- Reactor Coolant System Specific Activity
- Emergency Preparedness Drill and Exercise Performance
- Emergency Response Organization Drill Participation
- Alert and Notification System Reliability

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

a. Inspection Scope

The inspectors examined a sample of the licensee corrective action issues to provide an indication of the overall problem identification and resolution performance. Specifically, the inspectors examined CR-RBS-2001-1435, loss of normal power to Division 2 engineered safety feature 4160 VAC bus and start of Division II emergency diesel generator, to assess the licensee's identification of root and contributing causes.

In addition as part of other inspection activities, the inspectors reviewed the root cause analyses for:

- CR-RBS-2002-0664, Station blackout diesel generator inoperable due to discharged starting battery. For more details, see Section 1R13.
- CR-RBS-2002-0688, High reactor water level trip of reactor feed pump following scram during shutdown for planned outage. For more details, see Section 1R20.

c. Findings or Issues

No findings of significance were identified.

4OA6 Management Meetings

Exit Meetings

The inspectors presented the inspection results to Mr. Paul Hinnenkamp, Vice President-Operations, and other members of licensee management at the conclusion of various parts of the inspection on May 16, June 13, and July 3, 2002.

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

ATTACHMENT

SUPPLEMENTARY INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

B. Allen, Manager, Emergency Preparedness
M. Bakarich, Superintendent, Security
W. Brian, Director, Engineering
C. Bush, Assistant Operations Manager - Plant
J. Clark, Assistant Operations Manager - Staff
P. Felker, Operator Requalification Program Lead Instructor
J. Fowler, Manager, Quality Assurance
J. Heckenberger, Manager, Planning and Scheduling
P. Hinnenkamp, Vice President, Operations
R. King, Director, Nuclear Safety Assurance
J. Leavines, Manager, Nuclear Safety and Regulatory Affairs
T. Lynch, Manager, Operations
W. Mashburn, Manager, Engineering Programs
J. McGhee, Manager, Maintenance
D. Mims, General Manager, Plant Operations
W. Trudell, Manager, Corrective Action and Assessment

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:

Emergency Plan Implementing Procedures

EIP-2-001	Classification of Emergencies	Revision 11
EIP-2-002	Classification Actions	Revision 22
EIP-2-006	Notifications	Revision 29
EIP-2-007	PAR Guidelines	Revision 18
EIP-2-016	Operation Support Center	Revision 20
EIP-2-018	Technical Support Center	Revision 24
EIP-2-020	Emergency Operations Facility	Revision 25
EIP-2-023	Joint Information Center	Revision 12

EIP-2-024	Offsite Dose Calculations	Revision 19
EIP-2-026	Evacuation, Personnel Accountability, and Search and Rescue	Revision 12
EPP-2-703	Performance Indicators	Revision 1
LI-107	NRC Performance Indicator Process	Revision 1

Plant Administrative Procedures

RBNP-058	Licensed Operator Medical Certification Program	Revision 3
ADM-0022	Conduct of Operations	Revision 30
R-DAD-TQ-011	Simulator Training	Revision 2
TPP-7-011	Licensed Operator Requalification Training Program	Revision 13
98-02-00	Examination Security	9/7/98

Job Performance Measures

200-06	RSS Transfer following Control Room Evacuation
800-04	Bypass RWCU RPV Level 2 and SLC Isolation Interlocks
800-10	De-energize Scram Solenoids (EOP 0001)
800-11	Vent the Scram Air Header per EOP-0005
800-13	Operate Individual Scram Test Switches
800-29	Operate the Containment and Drywell H2 Igniters
05204.01	Alternate Control Rod Drive Pumps
10701.01	Startup Reactor Feedwater Pump “C”
10902.01	Return Isolated Main Steam Line “A” to Service
20003.01	Place Standby Service Water System in Service from the Remote Shutdown Panel

Simulator Dynamic Scenarios

00800.01	Loss of All Feedwater / DBA LOCA
00810.03	Single Rod Scram / RCIC Steam Leak / Loss of Offsite Power
00820.01	ATWS / Loss of HP Injection / Emergency Depressurization

Written Operator Requalification Examinations

LORJ-0001	LORQ MOD 10 SRO1BIEN.EXM
LORJ-0002	LORQ MOD 10 SRO2BIEN.EXM
LORJ-0003	LORQ MOD 10 RO1BIEN.EXM
LORJ-0003	LORQ MOD 10 RO2BIEN.EXM
LRS3-0209	BIENNIAL SRO EXAM 3
LRR3-0209	BIENNIAL RO EXAM 3

Operations Training Evaluation Reports

July - August, 2000
August - October, 2000
October - December, 2000
January - February, 2001
February - April, 2001
April - May, 2001
June - July, 2001
July - August, 2001
November - December, 2001
January - February, 2002
March - April, 2002

Condition Reports

CR-RBS-2002-0574
CR-RBS-2002-0592
CR-RBS-2002-0762

Other Documents

- Operations Department Standards and Expectations, Revision 13
- River Bend Station Emergency Plan, Revision 25
- EP Drill and Exercise Reports and Offsite Siren Test Results from January 2002 through March 2002
- EP Lesson Plans, ETT-032-9 and ETT-031-9, "Emergency Identification and Classification"
- EP Pager Test Results, November 2001 through May 2002.
- Biennial Requalification Training Program Two-Year Plan (January 2001 - December 2002)
- Licensed Operator Requalification Biennial Training Matrix

LIST OF ACRONYMS AND INITIALISMS USED

CFR	Code of Federal Regulations
CR	condition report
NRC	U.S. Nuclear Regulatory Commission
SSC	structure, system, or component
USAR	Updated Safety Analysis Report