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

January 4, 2001

EA-00-267

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

SUBJECT: RIVER BEND STATION--NRC INTEGRATED INSPECTION REPORT NO. 50-458/00-15

Dear Mr. Edington:

On December 23, 2000, the NRC completed inspections at your River Bend Station facility. The enclosed report presents the results of these inspections which were discussed with you and other members of your staff on December 21, 2000.

These inspections examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of the inspection, the inspectors identified one finding which involved a violation of NRC requirements. However, because of the very low safety significance and because the finding was entered into your corrective action program, the NRC is treating the finding as a noncited violation (EA-00-267), in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny the noncited violation, you should provide a response with the basis of your denial, within 30 days of the date of this inspection report to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the River Bend Station facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, 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/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.
50-458/00-15

cc w/enclosure:
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RAKopriva for	***/RA/***	***/RA/***	***CEJohnson for***	***RAKopriva for***
1/4/01	12/29/00	12/27/00	12/29/00	1/4/01

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-458
License No.:	NPF-47
Report No.:	50-458/00-15
Licensee:	Entergy Operations, Inc.
Facility:	River Bend Station
Location:	5485 U.S. Highway 61 St. Francisville, Louisiana
Dates:	November 12 through December 23, 2000
Inspectors:	T. W. Pruett, Senior Resident Inspector S. M. Schneider, Resident Inspector R. A. Kopriva, Senior Project Engineer R. V. Azua, Project Engineer M. F. Runyan, Senior Reactor 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 Station
NRC Inspection Report No. 50-458/00-15

IR 05000458-00-15; on 11/12-12/23/2000; Entergy Operations, Inc.; River Bend Station; Integrated Report. Changes to License Conditions and Safety Analysis Report.

This inspection was conducted by resident inspectors, regional project engineers, and a regional reactor inspector. The inspection identified one no color issue which was a noncited violation.

Cornerstone: Mitigating Systems

- The licensee made a change to the fuel handling accident dose reported in the Updated Safety Analysis Report for the exclusion area boundary thyroid that represented an increase in consequences without obtaining prior Commission approval as required by 10 CFR 50.59. This violation of 10 CFR 50.59(b)(1) identified above is categorized at Severity Level IV and is being treated as a noncited violation, consistent with Section VI.A1 of the NRC Enforcement Policy. This violation (50-458/0015-01) (EA-00-267-1) was entered into the licensee's corrective action program as Condition Report 2000-2050 (Section 1R02.b).

  This finding was of very low safety significance because previous and subsequent doses for the fuel handling accident exclusion area boundary thyroid were greater than the value implemented by this change.

Report Details

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

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

1R01 Adverse Weather (7111101)

a. Inspection Scope

The inspectors evaluated the design features and the implementation of the licensee's procedures to protect mitigating systems from adverse weather effects. The inspectors also toured heat trace protected systems, checked the alignment of heat protection power supplies, and reviewed operator cold weather logs. The following procedures and documents were reviewed during the assessment:

- OSP-0043, "Freeze Protection and Temperature Maintenance"
- Main Control Room and Auxiliary Control Room Cold Weather Logs

b. Findings

No findings of significance were identified.

1R02 Changes to License Conditions and Safety Analysis Report (7111102)

a. Inspection Scope

The inspectors reviewed a selected sample of safety evaluations to verify that the licensee had appropriately considered the conditions under which the licensee may make changes to the facility or procedures or conduct tests or experiments without prior NRC approval.

The inspectors reviewed a selected sample of safety evaluation screenings, in which the licensee determined that safety evaluations were not required, to ensure that the licensee's exclusion of a full evaluation was consistent with the requirements of 10 CFR 50.59.

The inspectors reviewed condition reports initiated by the licensee that addressed problems or deficiencies associated with 10 CFR 50.59 to ensure that appropriate corrective actions were being taken.

b. Findings

The inspectors reviewed License Change Notice (LCN) 15.7-004, which revised the Updated Safety Analysis Report (USAR) to reflect changes to data, assumptions, and doses reported for a fuel handling accident. This revision, dated May 19, 1999, was necessitated by Revision 2 to Calculation G13.18.9.5*045, "Design Basis Fuel Handling

Accident Dose Assessment," Case B (fuel handling accident in containment). The licensee made the USAR change under 10 CFR 50.59 after it determined that the change did not constitute an unreviewed safety question.

The USAR dose for the exclusion area boundary thyroid was revised by LCN 15.7-004 from 35.06 rem to 44.6 rem. The regulatory limit is 75 rem. The change was made to correct an error made in Revision 1 to Calculation G13.18.9.5*045 and to remove some conservativisms in the previous model. The USAR dose for exclusion area boundary thyroid had previously been 46.63 rem, prior to being changed to 35.06 rem by LCN 15.07-002, dated May 28, 1998 (which inadvertently incorporated the error made in Calculation G13.18.9.5*045, Revision 1). The changes to the model were performed in a successful effort to keep the dose below the previous value of 46.63 rem. The licensee used the previous higher dose as justification for not considering the increased dose under LCN 15.7-004 to constitute an unreviewed safety question, citing guidance published by the Nuclear Energy Institute (NEI). However, this approach was not valid because each change to the USAR under 10 CFR 50.59 should be evaluated in isolation and, as noted above, the May 1999 change did not solely correct the error made in May 1998.

The inspector noted that the NRC had not accepted the NEI guidance in place at the time of this LCN (May 1999) and that the subsequently published NEI guidance (NEI 96-07, "Guidelines for 10 CFR 50.59 Evaluations," Revision 1, endorsed by NRC in draft Regulatory Guide 1.187) would not specifically exempt this situation from requiring a license amendment under 10 CFR 50.90. The new rule, 10 CFR 50.59, which is not yet in effect, allows a licensee to make changes without NRC approval as long as the consequences of the change are not more than minimal. NEI 96-07, Revision 1, states in Section 4.3.3 that "an increase in consequences from a proposed activity is defined to be no more than minimal if the increase is less than 10 percent of difference between the current calculated dose value and the regulatory guideline value (10 CFR 100 or GDC 19, as applicable)." The dose change administered by LCN 15.7-004 from 35.06 rem to 44.6 rem, given the regulatory limit of 75 rem, did not meet this definition for a minimal change in consequences. Accordingly, this issue would have been a violation under the new 10 CFR 50.59 rule.

The inspector determined that, by changing the USAR dose for the exclusion area boundary thyroid from 35.06 rem to 44.6 rem, the licensee made changes to the facility as described in the USAR, without Commission approval, that involved an unreviewed safety question. Specifically, the unreviewed safety question was that the consequences of an accident were increased. This is a violation of 10 CFR 50.59.

This finding was of very low safety significance because previous and subsequent USAR doses for the fuel handling accident exclusion area boundary thyroid were greater than the value implemented by LCN 15.7-004. Normally, a license amendment would be required to correct the violation. However, in the time since the violation occurred, the NRC approved USAR Amendment 100, which increased the exclusion area thyroid dose for a fuel handling accident to 67 rem. Therefore, the inspector noted that a

license submittal specific to LCN 15.7-004 (as corrective action for this violation) may not be necessary. The licensee will evaluate this matter within its corrective action program.

Based on the very low safety significance of the analytical change to the USAR, the violation of 10 CFR 50.59(b)(1) identified above is categorized at Severity Level IV and is being treated as a noncited violation consistent with Section VI.A1 of the NRC Enforcement Policy. This violation (50-458/0015-01) (EA 00-267-1) was entered into the licensee's corrective action program as Condition Report (CR) 2000-2050.

1R04 Equipment Alignment (7111104)

.1 Verification of the 4160 and 480 Volt ac System

a. Inspection Scope

The inspectors performed an equipment alignment check of the 4160 and 480 volt ac system to verify that the system was properly configured and to identify any discrepancies that might impact the function of the system and thereby potentially increase risk. The inspectors reviewed documents to determine the correct system lineup and performed a walkdown to identify any discrepancies between the existing system lineup and the correct lineup. The inspectors also reviewed outstanding maintenance work requests and deficiencies which would preclude the system from performing its function and reviewed outstanding design issues and items tracked by the licensee to ensure equipment alignment problems had been properly identified and resolved. The inspectors also sampled the condition reporting system to verify equipment alignment problems were being identified at an appropriate threshold and properly evaluated for resolution. The following procedures and documents were reviewed during the assessment:

- SOP-0046, “4.16 KV System”
- SOP-0047, “480 VAC System”
- USAR
- Technical Specifications

b. Findings

No findings of significance were identified.

.2 Alignment Check of the Division II Emergency Diesel Generator

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

b. Findings

No findings of significance were identified.

.3 Alignment Check of the Reactor Core Isolation Cooling System

a. Inspection Scope

The inspectors completed a partial walkdown of the reactor core isolation cooling system while the high pressure core spray system was out of service to verify that the system was properly configured and to identify any discrepancies that might impact the function of the system and thereby potentially increase risk. The inspectors reviewed Procedure SOP-0035, “Reactor Core Isolation Cooling System,” during the assessment.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (7111105)

a. Inspection Scope

The inspectors toured the Division I and II switchgear areas and Cable Chases I, II, III, and IV to assess the control of transient combustible material, operational effectiveness of fire protection equipment, and the material condition of fire barriers. The following procedures were reviewed during the assessment:

- FPP-0030, "Storage of Combustibles"
- FPP-0050, “Handling of Flammable Liquids and Gases"
- FPP-0040, “Control of Transient Combustibles”

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

The inspectors verified that the licensee’s flooding mitigation plans and equipment were consistent with the licensee’s design requirements and the risk analysis assumptions. The area inspected was Pipe Tunnel C. CR 2000-2033 was reviewed during the assessment.

b. Findings

No findings of significance were identified.

1R07 Heat Exchangers (7111107)

a. Inspection Scope

No risk significant heat exchangers were tested during the inspection period. Consequently, this inspection was not completed.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (7111112)

a. Inspection Scope

The inspectors selected the following three performance problems associated with the ac electrical distribution system and evaluated the effectiveness of the licensee's corrective actions and maintenance rule determinations.

- CR 1999-0826, "Inverter Supply Breaker Tripped"
- CR 1999-1756, "Feeder Breaker for B21-MOVF067D Tripped Unexpectedly"
- CR 2000-0707, "Overcurrent Induction Disk Failed to Travel During 10 Amp Test"

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (7111113)

a. Inspection Scope

The inspectors evaluated the effectiveness of risk assessments performed by the licensee for the work weeks beginning November 26, December 3, and December 10, 2000. The following procedures were reviewed during the assessment:

- Maintenance Planning Guideline
- On-line Maintenance Guidelines
- Weekly Maintenance Schedules

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (7111115)

a. Inspection Scope

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

- CR 2000-2014, "Incomplete Testing of Remote Shutdown Circuitry"
- CR 2000-1977, "Current SSW Inventory Loss Through Boundary Valves Encroaching on Allowable Loss"
- CR 2000-1914, "Coaxial Cables Susceptible to Thermally Induced Currents"

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (7111119)

a. Inspection Scope

The inspectors reviewed the postmaintenance testing requirements specified for the maintenance action items (MAIs) listed below to ensure that the postmaintenance test procedures and test activities were adequate to verify system operability and functional capability:

- MAI 328777, "Service Water Return Header Isolation Valve SWP-MOV74A"
- MAI 328778, "Service Water Return Header Isolation Valve SWP-MOV74B"
- MAI 337784, "Replace Leaking Outboard Mechanical Seal On CRD Pump B"
- MAI 337361, "Troubleshoot and Repair LPRM 5C-38-23 (Input to APRM "A")"

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (7111122)

a. Inspection Scope

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

- STP-302-1201, "ENS-SWG1A Loss of Voltage Channel Functional Test"
- STP-302-1202, "ENS-SWG1A Degraded Voltage Channel Functional Test"

- STP-309-0301, "Division I Diesel Generator Operability Test"
- STP-309-6304, "Division I EDG Forward Bank Air Start System Quarterly Valve Operability Test"

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (7111123)

a. Inspection Scope

No risk significant temporary modifications were implemented by the facility since the last review of the area. Consequently, this inspection was not completed.

b. Findings

No findings of significance were identified.

4OA6 Management Meetings

Exit Meeting Summary

The reactor inspector presented his inspection results to Mr. R. Edington and other members of licensee management at the conclusion of the inspection on November 16, 2000. A supplemental exit was conducted on November 30, 2000, by telephone.

The resident inspectors presented the results of their inspection to Mr. Dwight Mims, General Manager Plant Operations, and other members of licensee management at the conclusion of the inspection on December 21, 2000.

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

ATTACHMENT 1

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

R. Biggs, Coordinator, Licensing
W. Brian, Director, Engineering
B. Burmeister, 10 CFR 50.59 Coordinator
E. Bush, Superintendent, Operations
R. Edington, Vice President-Operations
J. Fowler, Manager, Quality Assurance
T. Hildebrandt, Manager, Maintenance
J. Holmes, Manager, Technical Support
R. King, Director, Nuclear Safety Assurance
D. Lorfing, Coordinator, Licensing
J. Leavines, Manager, Licensing
J. McGhee, Manager, Operations
D. Mims, General Manager
D. Myers, Senior Specialist, Licensing
A. Shahkarami, Manager, System Engineering
D. Williamson, Licensing Specialist
M. Wyatt, Manager, Planning and Scheduling/Outage

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

50-458/0015-01	NCV	Failure to obtain Commission approval for a change to the USAR (Section 1R02)

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:

Safety Evaluations

98-0006, Evaluation of ER 98-0059 and LCN 09.03-238, Revision 0

98-0039, Evaluation of ER 97-0296 and LCN 11.05-003, Revision 0

98-0056, Evaluation of MR 90-0120 and LCN 3.2-013, Revision 0

98-0057, Evaluation of changes to Procedures EOP-0001, EOP-0002, EOP-0003, EOP-0004, EOP-0005, SAP-0001, and SAP-0002; Revision 0

99-0016, Evaluation of LCN 15.7-004, Calculation G13.18.9.5*045, “Design Basis Fuel Handling Accident Dose Assessment,” Revision 2; Revision 0

99-0020, Evaluation of LCN 10.04-178, “Change to USAR 10.4.6.5 Condensate Demineralizer Instrumentation,” Revision 0; Revision 0

99-0025, Evaluation of ER 99-0121, Reision 0

99-0026, Evaluation of changes to Procedure SOP-0031, “Residual Heat Removal,” Revision 0

99-0030, Evaluation of ER 99-0320, Revision 0

99-0031, Evaluation of LCN 6.2-08, Revision 0

99-0040, Evaluation of ER 98-0958, Revision 0

(No number assigned) Evaluation of MR 90-0146

10 CFR 50.59 Screenings

LCN 03.05-005, Revision 0 (ps)
LCN 05.02-028, Revision 0 (ps)
LCN 06.02-077, Revision 0 (ps)
LCN 06.02-082, Revision 0 (ps)
LCN 6.5-11, Revision 0 (ps)
LCN 09.01-060, Revision 0 (ps)
LCN 12.03-016, Revision 0 (ps)
ER 96-0695, Revision 0
ER 96-0607, Revision 0
MR 94-0173, Revision 0
ER 98-0733, Revision 1
ER 96-0712, Revision 0
ER 97-0529, Revision 0
ER 99-0679, Revision 0
ER 99-0404, Revision 0

Condition Reports

CR-RBS-1998-0795
CR-RBS-1999-1838
CR-RBS-2000-0190
CR-RBS-2000-0191
CR-RBS-2000-0292

Calculations

Calculation G13.18.9.5*045, "Design Basis Fuel Handling Accident Dose Assessment," Revision 2

LIST OF ACRONYMS AND INITIALISMS USED

CFR	Code of Federal Regulations
CR	condition report
LCN	license change notice
MAI	maintenance action item
NCV	noncited violation
NEI	Nuclear Energy Institute
NRC	U. S. Nuclear Regulatory Commission
USAR	Updated Safety Analysis Report

ATTACHMENT 2

NRC'S REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) 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 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	Radiation Safety	Safeguards
•Initiating Events	•Occupational	•Physical Protection
•Mitigating Systems	•Public	
•Barrier Integrity		
•Emergency Preparedness		

To monitor these seven cornerstones of safety, the NRC used 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, and 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. 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.