UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-4005

June 16, 2004

Paul D. Hinnenkamp
Vice President - Operations
Entergy Operations, Inc.
River Bend Station
5485 US Highway 61N
St. Francisville, LA 70775

SUBJECT: RIVER BEND STATION - NRC SAFETY SYSTEM DESIGN AND PERFORMANCE CAPABILITY INSPECTION REPORT 05000458/2004-008

Dear Mr. Hinnenkamp:

On May 21, 2004, the U.S. Nuclear Regulatory Commission (NRC) completed the onsite portion of an inspection at your River Bend Station. The enclosed report documents the inspection findings, which were discussed 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, no findings of significance were identified.

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/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Jeff Clark, Chief
Engineering Branch
Division of Reactor Safety

Docket: 50-458
License: NPF-47

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

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

General Manager
Plant Operations
Entergy Operations, Inc.
River Bend Station
5485 US Highway 61N
St. Francisville, LA 70775

Director - Nuclear Safety
Entergy Operations, Inc.
River Bend Station
5485 US Highway 61N
St. Francisville, LA 70775

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

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

Manager - Licensing
Entergy Operations, Inc.
River Bend Station
5485 US Highway 61N
St. Francisville, LA 70775

The Honorable Charles C. Foti, Jr.
Attorney General
Department of Justice
State of Louisiana
P.O. Box 94005
Baton Rouge, LA 70804-9005

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

Burt Babers, President
West Feliciana Parish Police Jury
P.O. Box 1921
St. Francisville, LA 70775

Michael E. Henry, State Liaison Officer
Department of Environmental Quality
Permits Division
P.O. Box 4313
Baton Rouge, LA 70821-4313

Brian Almon
Public Utility Commission
William B. Travis Building
P.O. Box 13326
1701 North Congress Avenue
Austin, TX 78711-3326

Electronic distribution by RIV:
Regional Administrator **(BSM1)**
DRP Director **(ATH)**
DRS Director **(DDC)**
Senior Resident Inspector **(PJA)**
Branch Chief, DRP/B **(DNG)**
Senior Project Engineer, DRP/B **(RAK1)**
Staff Chief, DRP/TSS **(PHH)**
RITS Coordinator **(KEG)**
Rebecca Tadesse, OEDO
RIV Coordinator **(RXT)**
RBS Site Secretary **(LGD)**

ADAMS: ☒ Yes ☐ No Initials: _JAC_____
☒ Publicly Available ☐ Non-Publicly Available ☐ Sensitive ☒ Non-Sensitive

SRI:EB	RI:EB	RI:EB	RI:EB	RI:EB	C:EB	C:PBB	C:EB
LEEllershaw/lmb	JPAdams	JMMateychick	WMMcNeill	BWHenderson	CSMarschall	DNGraves	JClark
/RA/	/RA/	/RA/	/RA/	/RA/	/RA/	/RA/	/RA/
06/08/04	06/09/04	06/09/04	06/09/04	06/10/04	06/10/04	06/10/04	06/16/04

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

ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket Nos:	50-458
License Nos:	NPF-47
Report No:	05000458/2004-008
Licensee:	Entergy Operations, Inc.
Facility:	River Bend Station
Location:	5485 U.S. Highway 61 St. Francisville, Louisiana
Dates:	May 3-21, 2004
Team Leader:	L. E. Ellershaw, Senior Reactor Inspector, Engineering Branch
Inspectors:	J. P. Adams, Reactor Inspector, Engineering Branch J. M. Mateychick, Reactor Inspector, Engineering Branch W. M. McNeill, Reactor Inspector, Engineering Branch B. W. Henderson, Reactor Inspector, Engineering Branch
Approved by:	Jeff Clark, Chief Engineering Branch Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000458/2004008; 05/03/2004 through 05/21/2004, River Bend Station; Evaluation of Changes, Tests, or Experiments, and Safety System Design and Performance Capability

The NRC conducted an inspection with five regional inspectors. No findings of significance were identified. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

<u>NRC-Identified Findings and Self-Revealing Findings</u>

No findings of significance were identified.

Report Details

1. REACTOR SAFETY

Introduction

The NRC conducted an inspection to verify that licensee personnel adequately preserved the facility safety system design and performance capability and that licensee personnel preserved the initial design in subsequent modifications of the systems selected for review. The scope of the review also included any necessary nonsafety-related structures, systems, and components that provided functions to support safety functions. This inspection also reviewed the licensee's programs and methods for monitoring the capability of the selected systems to perform the current design basis functions. This inspection verified aspects of the initiating events, mitigating systems, and barrier cornerstones.

The licensee personnel based the probabilistic risk assessment model for the River Bend Station on the capability of the as-built safety systems to perform their intended safety functions successfully. The team determined the area and scope of the inspection by reviewing the licensee's probabilistic risk analysis models to identify the most risk significant systems, structures, and components. The team established this according to their ranking and potential contribution to dominant accident sequences and/or initiators. The team also used a deterministic approach in the selection process by considering recent inspection history, recent problem area history, and all modifications developed and implemented.

The minimum sample size for this procedure is one risk-significant system for mitigating an accident or maintaining barrier integrity. The team completed the required sample size by reviewing the containment structures. The primary review prompted parallel review and examination of support systems, such as, containment atmosphere control, standby gas treatment, residual heat removal, and related structures and components.

The team assessed the adequacy of calculations, analyses, engineering processes, and engineering and operating practices that licensee personnel used for the selected safety system and the necessary support systems during normal, abnormal, and accident conditions. Acceptance criteria used by the team included NRC regulations, the technical specifications, applicable sections of the Updated Final Safety Analysis Report, applicable industry codes and standards, and industry initiatives implemented by the licensee's programs.

1R02 Evaluations of Changes, Tests, or Experiments (71111.02)

a. Inspection Scope

The minimum sample size for this procedure is 5 evaluations and 10 screenings. The team reviewed 7 licensee-performed 10 CFR 50.59 evaluations to verify that licensee personnel 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. These evaluations had been performed since the last NRC inspection of 10 CFR 50.59 activities.

The team reviewed 10 licensee-performed 10 CFR 50.59 screenings, in which licensee personnel determined that evaluations were not required to ensure that exclusion of a full evaluation was consistent with the requirements of 10 CFR 50.59. Additionally, the team reviewed 10 licensee-performed applicability determinations, in which licensee personnel determined that neither screenings nor evaluations were required, to ensure consistency with the requirements of 10 CFR 50.59 in the licensee's exclusion of screenings and evaluations.

The team reviewed and evaluated the most recent licensee 10 CFR 50.59 program self assessment and a sample of 10 corrective action documents written since the last NRC 10 CFR 50.59 inspection to determine whether licensee personnel conducted sufficient in-depth analyses of their program to allow for the identification and subsequent resolution of problems or deficiencies.

b. Findings

No findings of significance were identified.

1R21 Safety System Design and Performance Capability (71111.21)

.1 System Requirements

a. Inspection Scope

The team inspected the following attributes of the reactor containment structures: (1) process medium (water, steam, and air), (2) energy sources, (3) control systems, and (4) equipment protection. The team examined the procedural instructions to verify instructions are consistent with actions required to meet, prevent, and/or mitigate design basis accidents. The team also considered requirements and commitments identified in the Updated Final Safety Analysis Report, technical specifications, design basis documents, and plant drawings. In conjunction with the primary review of the reactor containment structures, a parallel review and examination of support systems, such as, containment atmospheric control, standby gas treatment, residual heat removal (shutdown cooling mode), penetrations, and related structures and components was also conducted.

b. Findings

No findings of significance were identified.

.2 System Condition and Capability

a. Inspection Scope

The team reviewed the periodic testing procedures for the containment and support systems to verify that the capabilities of the systems were verified periodically. The team also reviewed the systems' operations by conducting system walkdowns; reviewing normal, abnormal, and emergency operating procedures; and reviewing the Updated Final Safety Analysis Reports, technical specifications, design calculations, drawings, and procedures.

b. Findings

No findings of significance were identified.

.3 Identification and Resolution of Problems

a. Inspection Scope

The team reviewed a sample of problems associated with containment structures and support systems that were identified by licensee personnel in the corrective action program to evaluate the effectiveness of corrective actions related to design issues. The sample included open and closed condition reports for the past 3 years and are listed in the attachment to this report. Inspection Procedure 71152, "Identification and Resolution of Problems," was used as guidance to perform this part of the inspection. Older condition reports that were identified while performing other areas of the inspection were also reviewed.

b. Findings

No findings of significance were identified.

.4 System Walkdowns

a. Inspection Scope

The team performed walkdowns of the accessible portions of the containment structures and support systems. The team focused on the installation and configuration of switchgear, motor control centers, manual transfer switches, field cabling, raceways, piping, components, and instruments. During the walkdowns, the team assessed:

- The placement of protective barriers and systems,
- The susceptibility to flooding, fire, or environmental conditions,
- The physical separation of trains and the provisions for seismic concerns,
- Accessibility and lighting for any required operator action,
- The material conditions and preservation of systems and equipment, and
- The conformance of the currently-installed system configurations to the design and licensing bases.

b. Findings

No findings of significance were identified.

.5 Design Review

a. Inspection Scope

The team reviewed the current as-built instrument and control, electrical, and mechanical design of the containment structures and support systems. These reviews included an examination of design assumptions, calculations, environmental qualifications, required system thermal-hydraulic performance, electrical power system performance, control logic, and instrument setpoints and uncertainties. The team assessed the adequacy of calculations, analyses, test procedures, and operating procedures that licensee personnel used during normal and accident conditions.

The team also reviewed the adequacy of the combustible gas control system's original design to control hydrogen concentrations in the drywell and containment during post-accident conditions, including maintaining the capability of the selected support systems to perform their design basis functions. The support systems reviewed in detail were the drywell and containment hydrogen analyzer system, hydrogen igniter system, hydrogen recombiner system and drywell purge system.

b. Findings

No findings of significance were identified.

6. Safety System Inspection and Testing

a. Inspection Scope

The team reviewed the program and procedures for testing and inspecting selected components for the containment structures and support systems. The review included the results of surveillance tests required by the technical specifications and selective review of inservice tests.

b. Findings

No findings of significance were identified.

4OA3 Event Followup

(Closed) Licensee Event Report 05000458/2003003, Primary Containment Airlock Breach Due to Door Interlock Malfunction

On March 10, 2003, the upper airlock door interlocks at the 171-foot elevation of the primary containment did not properly function, such that both the inner and outer doors were unsealed for a period of approximately 36 minutes. At the time the reactor was operating at 87 percent power in end-of-cycle coastdown. This event was reported in accordance with 10 CFR 50.73(a)(2)(v)(c) as a condition that could have prevented the primary containment from performing its safety function. The licensee repaired the 171-foot upper airlock door interlocks to restore proper function. A root-cause analysis was performed following the event and resulted in the licensee changing maintenance practices and replacing the cables which operate the interlocks with a new and improved design. Identical actions were taken on the 113-foot elevation lower airlock door interlocks to prevent similar problems. The team reviewed the licensee event report and the root-cause analysis, and no findings of significance were identified. The licensee documented this condition in Condition Report CR-RBS-2003-0882. This licensee event report is closed.

4OA6 Management Meetings

Exit Meeting Summary

The inspection findings were acknowledged during an exit meeting presented by the team leader on May 21, 2004, to Mr. T. E. Trepanier, and other members of licensee management staff. The team leader confirmed that proprietary information, while reviewed, had not been retained by the team.

ATTACHMENT

PARTIAL LIST OF PERSONS CONTACTED

Licensee:

M. Ballard, Supervisor, Quality Audits
T. Burnett, Superintendent, Chemistry
C. Forpahl, Manager, Corrective Action and Assessments
R. Hebert, Manager, Materials, Procurement, and Contracts
R. King, Director, Nuclear Safety Assurance
J. Malara, Director, Engineering
W. Mashburn, Manager, Programs and Components
T. Trepanier, General Manager, Plant Operations
J. Clark, Assistant Manager, Operations
A. Roshto, Superintendent, Electrical Maintenance
B. Fountain, Licensing Specialist

NRC

M. Miller, Resident Inspector

LIST OF ITEMS CLOSED

Closed

50-458/2003-003	LER	Airlock door interlocks of the primary containment did not properly function such that both the inner and outer doors were unsealed and could have prevented the primary containment from performing its safety function (Section 4OA3)

DOCUMENTS REVIEWED

Calculations

G13.18.2.3*147, “G.L. 89-10 Design Basis Review for E12-MOVF003A/B,” Revision 1

G13.18.2.3*148, “G.L. 89-10 Design Basis Review for E12-MOVF004A/B,” Revision 4

G13.18.2.3*155, “G.L. 89-10 Design Basis Review for E12-MOVF024A/B,” Revision 5

G13.18.2.3*160, “G.L. 89-10 Design Basis Review for E12-MOVF042A/B, Revision 3

G13.18.15.2*052, “Cat. 1 Maximum Thrust Force for Valve 1E12*MOVF064A, 1E12*MOVF064B, & 1E12*MOVF064C,” Revision 0

G13.18.15.2*053, “Maximum Thrust Force for Valves: 1E12*MOVF042A, 042B, & 042C and 1E21*MOVBF005,” Revision 1

G13.18.15.2*107, “Maximum Thrust Force for Valves: 1E12*MOVF048A & F048B,” Revision 0

ES-170-6, “Addendum C (ES-170-6C), Accident Environmental Conditions In The Containment, Auxiliary Building, and Steam Tunnel,” Revision 0

ES-170-6, “Addendum D (ES-170-6D), Accident Environmental Conditions In The Containment, Auxiliary Building, and Steam Tunnel,” Revision 0

ES-170-6, “Addendum E (ES-170-6E), Accident Environmental Conditions In The Containment, Auxiliary Building, and Steam Tunnel,” Revision 0

ES-225, “Post DBA Hydrogen Concentration in Drywell and Containment for FSAR Section 6.2.5,” Revision 0

PB-326, “Effect of LOCA on Containment Unit Coolers,” Revision 0

PN-268, “RHR System Pumps TDH and NPSHA Except LPCI (Mode A-2) Operation,” Revision 5

7221-438-312-002C, “Weak Link Analysis for 14" ANSI Class 300 PermaSeat Valve,” Revision D

BV-45.20-1, “Fan External Total Pressure 1GTS*FN1A 7 FN1B Containment/Drywell Purge Exhaust Normal Operation Mode III,” Revision 1

BV-45.21, “Fan External Total Pressure 1GTS*FN1A and FN2B Decay Heat Exhaust Fans Normal Operation Mode II,” Revision 2

BV-45.22-1, “Fan External Total Pressure Fans 1GTS*FN1A & FN1B Annulus and Auxiliary Building Exhaust Accident Mode,” Revision 0

ES-073, “Wind Velocity to Offset Building Vacuum,” Revision 2

ES-194, “Auxiliary Building Pressure Following Loss of Coolant Accident for Updated Safety Analysis Report Sections 6.2.3,” Revision 4

ES-0205, “Technical Specification Secondary Containment Integrity Drawdown Times During Normal Operation,” Revision 1

G13.18.2.1*079, “Evaluation of Standby Gas Treatment System Drawdown Data,” Revision 0

G13.18.2.7*023, “Shield Building Annulus Pressure Following Loss of Coolant Accident for Updated Safety Analysis Report Section 6.2.3,” Revision 2

G13.18.14.0*01, “Standby Gas Treatment System and Fuel Building Charcoal Filter Decay Heat,” Revision 0

G13.18.9.5*051, “Loss of Coolant Accident Doses for Updated Safety Analysis Report Chapter 15,” Revision 2

G.13.18.9.5*061, “Alternate Source Term Loss of Coolant Accident Off-Site and Control Room Dose Analysis,” Revision 0

PB-206, “Sizing of Emergency Charcoal Filters 1GTS*FLT 1A and 1B,” Revision 1

PB-361, “Charcoal Weight and Iodine Loading Filtration Units, 1HVF*FLT2A-2B and 1GTS*FLT1A-1B,” Revision 0

PR-C-087, “Halogen Loading on the Standby Gas Treatment System and Fuel Building Charcoal Filters due to a Loss of Coolant Accident,” Revision 0

PR-C-344, “Peak Radiological Decay Heat Rate for Standby Gas Treatment System Filters during Loss of Coolant,” Revision 1A

Condition Reports (CRs)

CR-RBS-2003-03431	CR-RBS-2004-00172	CR-RBS-2004-01328	CR-RBS-2002-00353
CR-RBS-2002-01177	CR-RBS-2002-01404	CR-RBS-2002-02057	CR-RBS-2002-00131
CR-RBS-1993-00403	CR-RBS-2004-01374	CR-RBS-2002-00751	CR-RBS-2002-01343
CR-RBS-2002-00751	CR-RBS-2003-03341	CR-RBS-2004-01320	CR-RBS-2003-00364
CR-RBS-1998-00437	CR-RBS-2003-02661	CR-RBS-2002-01404	CR-RBS-2003-02028
CR-RBS-1997-00703	CR-RBS-2004-01346	CR-RBS-2002-00738	CR-RBS-2004-01045
CR-RBS-2002-00738	CR-RBS-2003-00961	CR-RBS-2004-01305	CR-RBS-2003-00882
CR-RBS-1998-00430	CR-RBS-2004-00172	CR-RBS-2004-00194	CR-RBS-2002-00443
CR-RBS-1997-02154	CR-RBS-2004-01346	CR-RBS-1997-00526	CR-RBS-2004-00122
CR-RBS-2004-01325	CR-RBS-2003-00605	CR-RBS-2002-01177	CR-RBS-2004-00004
CR-RBS-2003-00009	CR-RBS-2002-02057	CR-RBS-2004-01086	CR-RBS-2004-01478
CR-RBS-2004-00945	CR-RBS-2004-01331	CR-RBS-2002-00443	CR-RBS-2004-01479
CR-RBS-2004-01493	CR-RBS-2003-00009	CR-RBS-2004-00197	CR-RBS-2004-01480
CR-RBS-2003-01407	CR-RBS-2003-00605		

Design Basis Documents

CSD-27-13, Control System Description for Engineered Safety Features Hydrogen Recombiner System Diagram 27-13, April 6, 1984

CSD-27-21, Control System Description for Engineered Safety Features Containment Hydrogen Purge Diagram 27-21, July 26, 1985

CSD-27-24, Control System Description for Engineered Safety Features Hydrogen Mixing Diagram 27-24, September 17, 1985

SDC-204, Residual Heat Removal System Design Criteria System Number 204, Revision 3

SDC-403, 404, & 409, Reactor Plant Ventilation System Design Criteria System Numbers 403, 404, & 409, Revision 4

SDRD-E2, System Design Requirements Document - Hydrogen Control, Revision 0

SDRD-P12, System Design Requirements Document - Containment Hydrogen Control, Revision 0

Drawings

ESK-06HVR09, Elementary Diagram 480V SWGR Containment Unit Cooler*UC1A, Revision 22

ESK-06HVR10, Elementary Diagram 480V SWGR Containment Unit Cooler*UC1B, Revision 20

PID-27-21A, Engineering P & I Diagram System 254 Hydrogen Mixing Purge & Recombiner, Revision 5

PID-33-02A, Engineering P & I Diagram System 552 Containment Atmosphere and Leakage Monitoring, Revision 16

PID-33-02B, Engineering P & I Diagram System 552 Containment Atmosphere and Leakage Monitoring, Revision 1B

PID-33-02C, Engineering P & I Diagram System 552 Containment Atmosphere and Leakage Monitoring, Revision 7

LSK 22-1.6B, “Logic Diagram Reactor Plant Ventilation Annulus Mixing,” Revision 12

LSK 22-1.6C, “Logic Diagram Reactor Plant Ventilation Annulus Mixing,” Revision 12

LSK 27-15A, “Logic Diagram Standby Gas Treatment System,” Revision 121

LSK 27-15D, “Logic Diagram Standby Gas Treatment System,” Revision 12

LSK 27-15F, “Logic Diagram Standby Gas Treatment System,” Revision 11

PID-22-01B, “Engineering Process and Instrument Diagram System 403 Heating, Ventilation and Air Conditioning - Containment Building,” Revision 16

PID-22-01C, “Engineering Process and Instrument Diagram System 403 Heating, Ventilation and Air Conditioning - Containment Building,” Revision 13

PID-27-15A, “Engineering Process and Instrument Diagram System 257 Standby Gas Treatment,” Revision 15

TLD HVR-041, “Annulus Pressure Control System Suction Flow,” Sheet 1, Revision 0

TLD HVR-041, “Annulus Pressure Control System Suction Flow,” Sheet 2, Revision 0

828E535AA, “Elementary Diagram Low Pressure Core Spray System,” Sheet 6, Revision 22

828E535AA, “Elementary Diagram Low Pressure Core Spray System,” Sheet 10, Revision 234

Root-Cause Analysis Reports

“171 ft. Containment Airlock Inadvertent Breach Event”

Miscellaneous Documents

NRC Letter J. F. Harold to R. K. Edington, Subject: River Bend, Unit 1 - Issuance of Amendment Re: Increase in Maximum Allowable Thermal Power to 3039 Megawatts Thermal (TAC MA6185), dated October 6, 2000 and attached Safety Evaluation.

Modification Request 91-0101, To Override LOCA Signal for Manual Initiation of the System, September 23, 1991

Modification Request 95-0016, RHR Test Return Lines Vibration Reduction, August 24, 1995

Documentation of Telecom with NRC Regarding Fuel Reloads for GGNS&RBS From Ron Byrd, April 9, 2001. (Conversation between Adrienne Smith, Ron Byrd (both Entergy) and Pat Sekerak (NRC-NRR) dated April 3, 2001)

Technical Specification 3.3.6.3, Containment Unit Cooler System Instrumentation, Amendment 81

Technical Specification 3.6.3.1, Primary Containment Hydrogen Recombiners, Amendment 81

Technical Specification 3.6.3.2, Primary Containment and Drywell Hydrogen Igniters, Amendment 81

Technical Specification 3.6.3.3, Primary Containment/Drywell Hydrogen Mixing System, Amendment 89

Technical Specification 5.5.13, Primary Containment Leakage Rate Testing Program, Amendment 132

Technical Specification Bases B 3.6.3.1, Primary Containment Hydrogen Recombiners, Revision 4-3

Technical Specification Bases B 3.6.3.2, Primary Containment and Drywell Hydrogen Igniters, Revision 3-3

Technical Specification Bases B 3.6.3.3, Primary Containment/Drywell Hydrogen Mixing System, Revision 4-3

Entergy Licensing Position, “Evaluation and Resolution of Degraded and Nonconforming Conditions,” Revision 1

Loop Calibration Report No. 1.ILGTS.012, “Standby Gas Treatment Filter Train A Electric Heater Temperature Loop,” Revision 5

LO-RLO-204-00113, “Assessment Report: Evaluation of River Bend Station 10 CFR 50.59 Changes, Tests, or Experiments Program,” dated April 16, 2004

NFTA-NTR423, Field Test Reports, dated October 15, 1985

SDRD-P50, “System Design Requirements Document,” Revision 0

Safety Evaluation Report, “Increase in Maximum Allowable Thermal Power to 3039 Megawatts Thermal,” dated October 6, 2000

Specification 216.160, “Shop Fabrication of Ventilation and Air-Conditioning Systems Safety-Related Areas,“ Revision 2

Specification 225.220, “Standby Gas Treatment Units,” dated January 7, 1976
Stone & Webster Letter C-RBS-04444, dated June 25, 1986

NEI 94-01, “Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J,” dated July 26, 1995

Procedures

ADM-0050, “Primary Containment Leakage Rate Testing Program,” Revision 14

ADM-0085, “Repetitive Task Program, “ Revision 5

SEP-APJ-001, “Primary Containment Leakage Rate Testing Program (App. J),” Revision 0

CEP-IST-1, “Inservice Testing Bases Document, RBS Appendix,” Revision 3

CEP-IST-2, “Inservice Testing Plan” (including RBS Appendix, Valve Summary Listing and Pump Summary Listing), Revision 3

CEP-IST-3, “EN-S IST Cross-Reference Document” (including listing of procedures by pump and a listing of procedures by valve), Revision 1

CEP-IST-4, “Entergy Nuclear South Standard on Inservice Testing, “Revision 0

LI-102, “Corrective Action Process,” Revision 4

OE-100, “Operating Experience Program,” Revision 1

EOP-0005, “Emergency Operating and Severe Accident Procedures Enclosures,” Revision 15

EDG-AA-014, Technical Evaluations for Operability Determinations, Revision 0

ENG-3-028, Processing of System Design Criteria (SDC) Documents, Revision 6

EOP-0002, Emergency Operating Procedure - Primary Containment Control, Revision 13

EOP-0003, Emergency Operating Procedure - Secondary Containment and Radioactive Release Control, Revision 13

ENS-DC-126, Engineering Calculation Process, Revision 3

ENS-DC-141, Design Inputs, Revision 2

G12.1.22, Preparation, Review, Approval and Control of Revisions to the River Bend Station Environmental Design Criteria Database (RBS-EDC), Revision 06

PEP-0240, Performance Monitoring Program for the Residual Heat Removal Heat Exchangers E12-EB001B and E12-EB001D (Div II), Revision 03

SAP-0001, Severe Accident Procedure - RPV and Primary Containment Control, Revision 04

SAP-0002, Severe Accident Procedure - Containment and Radioactive Release Control, Revision 02

SOP-0031, Residual Heat Removal (Sys #204), Revision 41

SOP-0040, Hydrogen Mixing, Purge, Recombiners, and Ignitors, Revision 10A

SOP-0084, Containment Atmospheric Monitoring System (Sys #552), Revision 11

STP-051-4279, Containment Unit Cooler System Instrumentation, Unit Cooler A - Containment to Annulus Differential Pressure High Channel Calibration and Logic System Functional Test (HVR-ESZ60A, HVR-ESY60A, HVR-ESX60A, HVR-PDT60A), Revision 9

STP-254-0601, Containment/Drywell H2 Mixing System Flow Test, Revision 8

STP-254-1401, Division 1 Hydrogen Igniter Train Current and Voltage Check, Revision 4

STP-254-1603, Division 1 and 2 Hydrogen Igniter Current, Voltage and Temperature Check, Revision 4

STP-254-4203, Post Accident Monitoring - Drywell and Containment Hydrogen Analyzer Channel Calibration CMS-PNL10A, CMS-PNL12A, CMS-AR25A (Point 1), Revision 22A

STP-403-0301, Containment Unit Cooler HVR-UC1A Flow Rate Verification, Revision 10

STP-403-0303, Containment Unit Cooler HVR-UC1B Flow Rate Verification, Revision 0A

STP-403-1200, HVR-UC1A System A Timer Channel Functional Test, Revision 10

STP-403-1201, HVR-UC1B System A Timer Channel Functional Test, Revision 4

TSP-0016, Loop A RHR System Leak Test, Revision 3B

TSP-0017, Loop B RHR System Leak Test, Revision 4

AOP-0003, “Automatic Isolations,” Revision 19

ARP-RMS-DSPL230, “DRMS RM-11 CRT (RMS-DSPL230) Alarm Response,” Revision 3

ARP-863-73, “Division I Standby Gas Treatment Heat Removal System Inoperative,” Revision 5

DG-LI-101, “10 CFR 50.59 Review Program Guidelines,” Revision 5

EOP-00003, “Emergency Operating Procedure-Secondary Containment and Radioactive Release Control,” Revision 13

LI-101, “10 CFR 50.59 Review Program,” Revision 3

LI-113, “Licensing Basis Document (LBD) Control Program,” Revision 3

SOP-0043, “Standby Gas Treatment System (System Number 257),” Revision 10

SOP-0059, “Containment Heat Ventilation and Air Conditioning System (System 403),” Revision 22

STP-000-0001, “Daily Operating Logs”, Revision 45

STP-257-3602, “Inservice Testing of Division II Standby Gas Treatment Filtration System,” Revision 00B

STP-257-0601, “Standby Gas Treatment System Train A Drawdown Test,” Revision 14

STP-257-0602, “Standby Gas Treatment System Train B Drawdown Test,” Revision 5

STP-257-8601, “Standby Gas Treatment System Laboratory Carbon Filter Analysis,” Revision 11

STP-309-0601, “Division I 18 Month Emergency Core Cooling System Test,.” Revision 20

STP-403-0603, “Division I Standby Gas Treatment System and Annulus Mixing System Functional Test,” Revision 2

STP-403-0604, “Division II Standby Gas Treatment System and Annulus Mixing System Functional Test,” Revision 2

STP-601-6801, “Reactor Water Clean Up Cold Shutdown Valve Operability Test,” Revision 3

UFSAR, Revision 14

Section 6.2, Containment Systems

Section 7.3, Engineered Safety Feature Systems

Section 15.6, Decrease In Reactor Coolant Inventory

10 CFR 50.59 Evaluations

SEN 2003-013
SEN 2003-007
SEN 2001-024
SEN 2003-010
SEN 2002-024
SEN 2003-014
SEN 2002-016

10 CFR 50.59 Screenings

Procedure AOP-0020, Revision 2
Procedure AOP 0031, Revision 19
Procedure AOP 0052, Revision 12
Procedure EOP 0001, Revision 19
Procedure TSP 0010, Revision NA
Procedure TSP 0019, Revision NA
Procedure FHP 0001, Revision 25
Procedure FHP 0008, Revision 3
ER-RB-2001-0801-000, Revision 0
ER03-0510-000, Revision 0

10 CFR 50.50 Evaluation Exemptions

LAR 2004-05
ER-00-0330, Revision 0 & LCN 09.03-259
ER-RB-2002-0333-000 & LBDC-07.03-194
LBDC N.03.04-001
LCN 10.04-191
LBDC 09.02-346
LBDC 05.02-032
LBDC 06.02-107
LBDC 09.04-175
LBDC 06.02-108

Set Point Data Sheet Numbers

12210-IA-GTS*FS24
12210-IA-GTS-PDS6
12210-IA-GTS-PDS7
12210-IA-GTS-PDS16
12210-IA-GTS-PDS17
12210-IA-GTS-PDS26

Special Test Report
TP-98-0002 (both tests performed on April 25 and April 18, 1998)

System Design Criteria
SDC-257, “Standby Gas Treatment System Number 257,” Revision 1

Maintenance Action Items and Associated Test Reports:

346761	351498	369418	369423	316759	369419
350231	368735	349620	369424	316760	349620

Training Manuals

R-STM-257.01, “Standby Gas Treatment System,” Revision 1

Work Orders

50685430	50870370	50970230	50973996
50686758	50967705	50971425	50975102
50690017	50968944	50972175	50373327-01
50691392			

Work Requests

WO 18895
WO 18896

Engineering Requests

98-0166	98-0342	99-0885
98-0215	97-0291	2001-0115
98-0323		

Inservice Test Reports for the following Residual Heat Removal Pumps and Valves (last four quarterly tests):

E12-PC002A	E12-MOVF011B	E12-MOVF048A
E12-PC002B	E12-MOVF024A	E12-MOVF053B
E12-PC002C	E12-MOVF027A	E12-MOVF064A
E12-MOVF003A	E12-MOVF042A	E12-MOVF094
E12-MOVF004A	E12-MOVF047A	E12-MOVF096

Surveillance Test Reports

Test Reports for Containment Purge Valves in Penetrations KJB-Z31 and KJB-Z33: HVR-AOV123, -AOV128, -AOV165, -AOV166, CPP-MOV104, -MOV105, and -SOV140, dated 6/4-5/03, 8/27-28/03, 11/18-19/03, 2/9-10/04, and 5/3-4/04, respectively

Test Reports for Primary Containment Upper Airlock Inner and Outer Doors, and Lower Airlock Inner and Doors dated 1/6/04, 2/5/04, 3/2/04, 3/31/04, and 4/27/04

Section XI Safety and Relief Valve Testing for FPW-RV40 for period from 9-15-1997 to present (seven reports)

Surveillance Test Procedures

STP-057-0401, Revision 15
STP-403-7301, Revision 3
STP-057-7705, Revision 7
STP-057-3900, Revision 9