Entergy

**Entergy Operations, Inc.**
River Bend Station
5485 U.S. Highway 61N
St. Francisville, LA 70775
Tel 225 381 4157
Fax 225 635 5068
dlorfin@entergy.com

**David N. Lorfing**
Manager-Licensing

December 18, 2006

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: Licensee Event Report 50-458 / 06-007-00
River Bend Station – Unit 1
Docket No. 50-458
License No. NPF-47

File Nos. G9.5, G9.25.1.3

RBG-46638
RBF1-06-0189

Ladies and Gentlemen:

In accordance with 10CFR50.73, enclosed is the subject Licensee Event Report. This document contains no commitments.

Sincerely,

David N. Lorfing
Manager – Licensing

DNL/dhw
Enclosure

IE22

cc: U. S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011

NRC Sr. Resident Inspector
P. O. Box 1050
St. Francisville, LA 70775

INPO Records Center
E-Mail

Mr. Jim Calloway
Public Utility Commission of Texas
1701 N. Congress Ave.
Austin, TX 78711-3326

Mr. Jeff Meyers
Louisiana Department of Environmental Quality
Office of Environmental Compliance
P.O. Box 4312
Baton Rouge, LA 70821-4312

NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 06/30/2007

Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

# LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

| 1. FACILITY NAME | 2. DOCKET NUMBER | 3. PAGE |
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**4. TITLE**
Automatic Reactor Scram Due to Inadvertent Isolation of Main Feedwater Headers

| 5. EVENT DATE | | | 6. LER NUMBER | | | 7. REPORT DATE | | | 8. OTHER FACILITIES INVOLVED | |
|---|---|---|---|---|---|---|---|---|---|---|
| MONTH | DAY | YEAR | YEAR | SEQUENTIAL NUMBER | REV NO. | MONTH | DAY | YEAR | FACILITY NAME | DOCKET NUMBER 05000 |
| 10 | 19 | 2006 | 2006 | - 007 - | 00 | 12 | 18 | 2006 | FACILITY NAME | DOCKET NUMBER 05000 |

**9. OPERATING MODE**: 1

**10. POWER LEVEL**: 100

**11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§:** *(Check all that apply)*

| | | | |
|---|---|---|---|
| ☐ 20.2201(b) | ☐ 20.2203(a)(3)(i) | ☐ 50.73(a)(2)(i)(C) | ☐ 50.73(a)(2)(vii) |
| ☐ 20.2201(d) | ☐ 20.2203(a)(3)(ii) | ☐ 50.73(a)(2)(ii)(A) | ☐ 50.73(a)(2)(viii)(A) |
| ☐ 20.2203(a)(1) | ☐ 20.2203(a)(4) | ☐ 50.73(a)(2)(ii)(B) | ☐ 50.73(a)(2)(viii)(B) |
| ☐ 20.2203(a)(2)(i) | ☐ 50.36(c)(1)(i)(A) | ☐ 50.73(a)(2)(iii) | ☐ 50.73(a)(2)(ix)(A) |
| ☐ 20.2203(a)(2)(ii) | ☐ 50.36(c)(1)(ii)(A) | ☒ 50.73(a)(2)(iv)(A) | ☐ 50.73(a)(2)(x) |
| ☐ 20.2203(a)(2)(iii) | ☐ 50.36(c)(2) | ☐ 50.73(a)(2)(v)(A) | ☐ 73.71(a)(4) |
| ☐ 20.2203(a)(2)(iv) | ☐ 50.46(a)(3)(ii) | ☐ 50.73(a)(2)(v)(B) | ☐ 73.71(a)(5) |
| ☐ 20.2203(a)(2)(v) | ☐ 50.73(a)(2)(i)(A) | ☐ 50.73(a)(2)(v)(C) | ☐ OTHER |
| ☐ 20.2203(a)(2)(vi) | ☐ 50.73(a)(2)(i)(B) | ☐ 50.73(a)(2)(v)(D) | Specify in Abstract below or in NRC Form 366A |

**12. LICENSEE CONTACT FOR THIS LER**

| FACILITY NAME | TELEPHONE NUMBER (Include Area Code) |
|---|---|
| David N. Lorfing, Manager – Licensing | 225-381-4157 |

**13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT**

| CAUSE | SYSTEM | COMPONENT | MANU-FACTURER | REPORTABLE TO EPIX | CAUSE | SYSTEM | COMPONENT | MANU-FACTURER | REPORTABLE TO EPIX |
|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | |

| 14. SUPPLEMENTAL REPORT EXPECTED | | 15. EXPECTED SUBMISSION DATE | MONTH | DAY | YEAR |
|---|---|---|---|---|---|
| ☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* | ☒ NO | | | | |


ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On October 19, 2006, at approximately 5:57 p.m. CDT, an automatic reactor scram occurred in response to a low water level signal (Level 3) in the reactor vessel. This condition was the result of the inadvertent closure of the motor-operated isolation valves in the main feedwater headers supplying the reactor. These valves closed when part of a chart recorder was accidentally dropped on their control switches. The high pressure core spray system automatically actuated as designed when its reactor water level (Level 2) initiation setpoint was reached. The reactor core isolation cooling system was out of service for planned maintenance. Reactor steam pressure began to decrease as expected, and when pressure reached 849 psig approximately three minutes after the scram, the main steam isolation valves (MSIVs) automatically closed. The MSIVs closed because the reactor mode switch was not promptly re-positioned as required by scram response procedures. This event is being reported in accordance with 10CFR50.73(a)(2)(iv) as an automatic actuation of the reactor protection system and the high pressure core system (including the Division 3 diesel generator). Also, primary containment isolation signals actuated as a result of the Level 2 condition and the low reactor steam pressure signal to the MSIVs.

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REPORTED CONDITION

On October 19, 2006, at approximately 5:57 p.m. CDT, an automatic reactor scram occurred in response to a low water level signal (Level 3) in the reactor vessel. This condition was the result of the inadvertent closure of the motor-operated isolation valves (**ISV**) in the main feedwater (**SJ**) headers supplying the reactor. The high pressure core spray (HPCS) (**BJ**) system automatically actuated as designed when its reactor water level (Level 2) initiation setpoint was reached. The reactor core isolation cooling (RCIC) (**BN**) system was out of service for planned maintenance.

Reactor steam pressure began to decrease as expected, and when pressure reached 849 psig approximately three minutes after the scram, the main steam (**SB**) isolation valves (MSIVs) automatically closed.

This event is being reported in accordance with 10CFR50.73(a)(2)(iv) as an automatic actuation of the reactor protection system (**JC**) and the high pressure core system (including the Division 3 diesel generator). Also, primary containment isolation signals actuated as a result of the Level 2 condition and the low reactor steam pressure signal to the MSIVs.

When the immediate report of this event was made to NRC (Event Notification No. 42921), one of the reporting criteria cited was the loss of safety function of the HPCS system. The rationale for that report was, (1) HPCS is a single-train safety system, and (2) when the injection valve is manually closed following an actuation, it is then incapable of responding to any further automatic signals to open until it is reset by the operator. However, during this event the HPCS system actuated as designed and restored reactor water level, completing its safety function. As reactor water level was rising toward the normal operating range, the operator closed the injection shutoff valve at approximately 6:00 p.m. This action is consistent with transient response procedures, and was taken as part of the restoration of normal feedwater flow. Additionally, the description of the HPCS system in the River Bend Updated Safety Analysis Report (USAR) states that, "The HPCS pump motor and injection valve are provided with manual override controls. These controls permit the reactor operator to control the system manually following automatic initiation." As such, the original report of a loss of safety function of the HPCS system was conservative and is hereby withdrawn.

SEQUENCE OF EVENTS

Prior to the event, the plant was operating at 100 percent power. The on-coming main control room crew was present and performing shift turnover. The reactor operator had taken the watch, and was performing a control panel walkdown. He identified a chart recorder (**LR**) that required adjustment in the paper roller mechanism. The

operator pulled the recorder out of its chassis to gain access to the paper roller mechanism, which unexpectedly disengaged from the recorder and fell out. The mechanism landed on the horizontal section of the control panel below.

The roller mechanism struck the CLOSE pushbuttons for FWS-MOV7A and FWS-MOV7B, the feedwater header outboard containment isolation valves. After seeing no adverse effects or abnormal conditions due to the dropped part, the operator finished the adjustment and reassembly of the chart recorder.

Within two minutes after the roller mechanism fell onto the control switches, a reactor low water level (Level 4) alarm sounded. The operator noted that water level continued to decrease, and announced that he was initiating a manual reactor scram. Analysis of computer data later found that the automatic scram signal actuated approximately four seconds prior to the manual signal

Following the scram, the reactor operator did not take the reactor mode switch from "RUN" to "SHUTDOWN" as required. (When the mode switch is not in "RUN," the MSIV isolation signal actuated by low reactor steam pressure (<849 psig) is bypassed.) When reactor steam pressure decreased to 849 psig with the mode switch still in the "RUN" position, the MSIVs isolated. Reactor pressure was subsequently controlled by manual operation of the main steam safety relief valves (SRVs) until the MSIVs were re-opened at approximately 6:54 p.m.

## CAUSAL ANALYSIS

A root cause analysis of this event was performed, focusing on both the cause of the reactor scram and the closure of the MSIVs.

An inspection of the chart recorder found that the pivot pin on the paper roller mechanism was loose, which allowed it to be easily disengaged from the recorder housing while the operator was adjusting the paper drive. The environmental conditions for this activity (i.e., the close proximity of control switches under the recorder) presented a risk for misoperation, in that no barriers were used for protection of the control switches on the horizontal benchboard section. Formal expectations for physical control and protection of components handled above the control panels were not programmatically or procedurally provided.

Following the scram, the reactor operator and the STA began evaluating the cause of the loss of feedwater flow. This distracted the reactor operator to the extent that he did not promptly complete the immediate required actions for responding to a scram, including the repositioning of the reactor mode switch to "SHUTDOWN." The operator reported to the Control Room Supervisor (CRS) a verbal confirmation that all control rods were

inserted, but he made no report about the mode switch. Neither did the CRS specifically request that report from the operator.

CORRECTIVE ACTION TO PREVENT RECURRENCE

The following actions have been taken to prevent recurrence of a similar event:

1. The control switches for the FWS-MOV7A and 7B have been covered to prevent inadvertent operation.

2. Inspections of other recorders in the Main Control Room and Auxiliary Control Room were conducted to verify the security of the paper driver mechanisms.

3. Specific standards and expectations have been established for activities in and above control panels.

Further actions are in progress (and are being tracked in the station's corrective action program):

4. The sequence of operator actions regarding initiation of a manual scram will be modified to use the reactor mode switch first. (Currently, the four divisional manual scram pushbuttons are used for this purpose.)

5. Actions will be taken to identify and cover other control switches that needed protection to prevent similar inadvertent actuations.

6. A more specific standard is being developed for reacting to a manual or automatic scram and utilizing verbal / procedural verifications early in the event. Enhanced training will be conducted on this standard.

7. Training will be conducted on minimizing and managing distractions and transients during shift turnover and during the initial moments of an event.

8. A "scram report" checklist has been formalized as a communication tool for the operators and CRS. Training will be conducted on the expectations regarding its use.

PREVIOUS EVENTS EVALUATION

No previous scrams occurring at River Bend have resulted from a similar initiating condition.

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SAFETY SIGNIFICANCE

All plant systems responded as designed. The response of the plant was consistent with the accident analysis in the Updated Safety Analysis Report. While the RCIC system was out of service, the HPCS system responded as designed to restore reactor water level. Reactor fuel thermal limits remained within specified limits. There were no radiological releases. As such, this event was of minimal safety significance with regard to health and safety of the public.

An analysis of the stresses placed on the feedwater isolation valves by this transient determined that there were no detrimental effects, and that the valves were safe for return to full power operations.

(NOTE: Energy Industry Component Identification codes are annotated as (**XX**).)