Entergy

Entergy Operations, Inc.
River Bend Station
5485 U.S. Highway 61
P. O. Box 220
St. Francisville, LA 70775
Tel 225 336 6225
Fax 225 635 5068

Rick J. King
Director
Nuclear Safety Assurance

June 26, 2003

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

Subject: River Bend Station
Docket No. 50-458
License No. NPF-47
Licensee Event Report 50-458 / 03-001-01

File Nos. G9.5, G9.25.1.3

RBG-46138
RBF1-03-0117

Ladies and Gentlemen:

In accordance with 10CFR50.73, enclosed is the subject Licensee Event Report. This report is a supplement to Licensee Event Report 50-458 / 03-001-00, issued on 4/22/03. Revisions are annotated by change bars in the right margin. There are no commitments in this document.

Sincerely,

Rick J. King

RJK/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. Prosanta Chowdhury
Program Manager – Surveillance Division
Louisiana DEQ
Office of Radiological Emergency Planning and Response
P. O. Box 82215
Baton Rouge, LA 70884-2215

NRC FORM 366 (1-2001)

U.S. NUCLEAR REGULATORY COMMISSION

# LICENSEE EVENT REPORT (LER)

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



| FACILITY NAME (1) | DOCKET NUMBER (2) | PAGE (3) |
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TITLE (4)
Unplanned Reactor Scram Due to Fluid Leak in Main Turbine Electrohydraulic Control System

| EVENT DATE (5) | | | LER NUMBER (6) | | | REPORT DATE (7) | | | OTHER FACILITIES INVOLVED (8) | |
|---|---|---|---|---|---|---|---|---|---|---|
| MO | DAY | YEAR | YEAR | SEQUENTIAL NUMBER | REV NO | MO | DAY | YEAR | FACILITY NAME | DOCKET NUMBER 05000 |
| 02 | 22 | 2003 | 2003 | 001 | 01 | 06 | 26 | 2003 | FACILITY NAME | DOCKET NUMBER 05000 |

| OPERATING MODE (9) | 1 |
|---|---|
| POWER LEVEL (10) | 90% |

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

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

LICENSEE CONTACT FOR THIS LER (12)

| NAME | TELEPHONE NUMBER (Include Area Code) |
|---|---|
| J.W. Leavines, Manager - Licensing | 225-381-4642 |

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

| CAUSE | SYSTEM | COMPONENT | MANU-FACTURER | REPORTABLE TO EPIX | CAUSE | SYSTEM | COMPONENT | MANU-FACTURER | REPORTABLE TO EPIX |
|---|---|---|---|---|---|---|---|---|---|
| X | SJ | V | Velan | Y | B | TG | (see p. 3) | | N |

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

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

At approximately 1:23 a.m. CST on February 22, 2003, with the plant operating at 90 percent power in end-of-cycle coastdown, an unplanned manual reactor scram was initiated. During the scram recovery, the reactor core isolation cooling (RCIC) system was manually actuated to assist in reactor water level control. This event is being reported in accordance with 10CFR50.73(A)(2)(iv)(a) as a valid actuation of the reactor protection system and RCIC.

The scram was necessitated by a fluid leak in the main turbine electrohydraulic control system that significantly depleted the system. Subsequent investigation determined that the leak occurred when a section of hydraulic tubing near the main turbine control valves developed a through-wall crack. An analysis of the failed tubing indicated that the likely cause was metal fatigue induced by system vibration. Additionally, pipe stress analysis verified that the tubing was overstressed due to deficiencies in design and configuration.

Plant safety systems responded normally to the manual scram and the RCIC initiation. This event was of minimal significance to the health and safety of the public.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

REPORTED CONDITION

At approximately 1:23 a.m. CST on February 22, 2003, with the plant operating at 90 percent power in end-of-cycle coastdown, an unplanned manual reactor scram was initiated. During the scram recovery, the reactor core isolation cooling (RCIC) system was manually actuated to assist in reactor water level control. This event is being reported in accordance with 10CFR50.73(A)(2)(iv)(a) as a valid actuation of the reactor protection system (RPS) and RCIC.

INVESTIGATION

Approximately ten minutes prior to the scram, an alarm actuated in the main control room indicating a low level in the main turbine (**TRB**) electrohydraulic control (EHC) oil reservoir. An operator was dispatched to investigate, and reported that the EHC pump (**P**) was cavitating and that there appeared to be smoke in the area of the turbine front standard. The scram was initiated in anticipation of losing EHC system function. The main turbine unloaded normally as steam pressure decreased, and the main generator output breaker tripped on reverse power as designed. The EHC system was then shut down.

Following the scram, reactor water level lowered to the low alarm setpoint (Level 3). The feedwater level setpoint setdown, reactor recirculation pumps downshift, and the suppression pool cooling system trip all occurred as expected. Approximately 35 seconds later, reactor water level rose to the high alarm setpoint (Level 8). Due to a pre-existing deficiency, the "C" feedwater regulating valve (**FCV**) was not able to close to less than 80 percent open. A high reactor water alarm (Level 8) signal was received, tripping all three main feedwater pumps as designed. RCIC was initiated to provide water level control. The "A" loop of the residual heat removal system was placed into the suppression pool cooling mode as required for RCIC operation.

Subsequent to the RCIC initiation, main feedwater pump "A" was restarted to provide level control. Three minutes after the "A" main feedwater pump was restarted, reactor water level reached Level 8 for a second time. This automatically closed the RCIC system injection valve and tripped the feedwater pump. Five minutes later, operators restarted main feedwater pump "B" and reactor water level increased to Level 8 for the third time approximately 4 minutes after the pump start. Reactor water level control using RCIC was established at 2:01 a.m. The RCIC system was shut down at 4:00 p.m. on the day of the event after the main feedwater system was put back into service.

During the refueling outage subsequent to the reported event, internal inspections of the "C" feedwater regulating valve, the "C" main feed pump discharge check valve

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(**V**), and the downstream isolation valve (**ISV**) were performed. Part of a washer missing from the discharge check valve was found inside the feedwater regulating valve that was preventing it from closing past the 80 percent open position. Additionally, the disc nut from the check valve was missing, and was found lodged in the seat of the isolation valve. This was preventing the isolation valve from closing completely. These conditions were determined to be significant contributing factors to the difficulty in maintaining reactor water level less than Level 8 during this event.

CAUSAL ANALYSIS AND CORRECTIVE ACTIONS

The investigation determined that the leak occurred when a section of hydraulic tubing (1" diameter, 0.083" wall thickness type 304 stainless steel) near the main turbine control valves developed a through-wall crack. An analysis of the failed tubing indicated that the likely cause was metal fatigue induced by system vibration. The failure occurred in an area where the heat-affected zones from closely spaced welds overlapped. Additionally, subsequent pipe stress analysis verified that the tubing was overstressed.

The tubing had been modified in March 2000 during the addition of pressure pulsation dampening accumulators to the system. No stress analysis was performed for the as-left configuration of the tubing. Contributing factors were determined to be:

1. Non-safety related piping designs were not typically treated with the same level of rigor as safety related systems.

2. There was an over-reliance on the technical completeness of a vendor recommended solution.

Further inspection of the feedwater system was conducted during the recent refueling outage to determine the cause of the Level 8 pump trips that occurred during the scram recovery. As described above, an inspection of the "C" feedwater pump discharge check valve (Velan Valve Co., model no. B22-7114P-02TS) following the scram determined that the disc, the disc nut, and the washer came loose after the cotter pin inserted in the nut came out. The corresponding check valves for the "A" and "B" pumps were inspected during the refueling outage in March 2003. The cotter pin was missing in the "B" valve, and the disc nut was loose but still in place. The inspection of the "A" valve was satisfactory. The cotter pins from the "B" and "C" valves could not be located. It is suspected they were dislodged by flow-induced vibration.

The parts list for the pump discharge check valve calls for a 1/8" diameter cotter pin, although the cotter pin hole is 3/16" in diameter. The vendor confirmed that a larger

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cotter pin would provide a more secure fitting, and recommended staking the disc stud to increase the security of the disc nut. An engineering evaluation was processed to allow increasing the diameter of the cotter pin.

CORRECTIVE ACTION TO PREVENT RECURRENCE

The following corrective actions have been completed:

1. Repairs to the affected tubing on turbine control valve number 1 were completed prior to startup from the forced outage. The repairs included increasing the wall thickness of the tubing.

2. During the subsequent refueling outage, the turbine control valve EHC tubing was reconfigured based on a system stress analysis to eliminate the overstress condition, and was modified to increase the tubing wall thickness.

3. The "B" and "C" feedwater pump discharge check valves were reassembled with 3/16" cotter pins, and the disc studs were staked. The 1/8" cotter pin in the "A" valve was replaced with a 3/16" cotter pin.

SAFETY SIGNIFICANCE

The plant was shutdown following the discovery of a leak in the main turbine EHC system. A manual reactor scram was initiated and all safety systems operated per design. Prior to the event, reactor power was at 90 percent in end-of-cycle coastdown. The core was operating with approximately 15 percent margin to thermal power margins. No power excursion or pressure excursion were seen in post-trip data. Therefore, the safety limits specified in Technical Specification were not challenged.

Reactor water level was maintained above the Level 2 setpoint at which emergency core cooling systems are actuated. Water level was controlled and maintained by the RCIC system. Reactor feedwater pumps remained available following their automatic shutdown upon reaching Level 8, although the decision was made to utilize RCIC for an expeditious recovery. Reactor water level was returned to a normal post-shutdown range in a timely manner.

There was no release of steam to the containment, drywell, or suppression pool; therefore, there was no challenge to containment integrity. The reactor vessel pressure did not exceed the first safety relief valve set point of 1133 psig, or the high pressure

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scram setpoint (nominally 1094.7 psig); therefore, there was no challenge to the reactor coolant pressure boundary.

As none of the barriers to fission product release were challenged, there was no adverse effect on nuclear safety. This event had minimal effect on the health and safety of the public.

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