GULF STATES UTILITIES COMPANY
RIVER BEND STATION POST OFFICE BOX 220 ST. FRANCISVILLE, LOUISIANA 70775
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December 17, 1993
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U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

SUBJECT: River Bend Station - Unit 1
Docket No. 50-458
License No. NPF-47
Licensee Event Report 50-458/93-026-00

Gentlemen:

In accordance with 10CFR50.73(a)(2)(iv), enclosed is the subject report concerning an isolation of the reactor core isolation cooling system.

Very truly yours,

LL Dietrich

for James. J. Fisicaro
Manager - Safety Assessment
and Quality Verification
River Bend Nuclear Group

Enclosure

210110

JE22

cc: U.S. Nuclear Regulatory Commission
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NRC Resident Inspector
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Louisiana Department of Environmental Quality
Nuclear Energy Division
P.O. Box 82135
Baton Rouge, LA 70884-2135
ATTN: Administrator

NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
RIVER BEND STATION	05000458	1 OF 4

TITLE (4) ISOLATION OF THE REACTOR CORE ISOLATION COOLING SYSTEM DUE TO AN APPARENT FAILURE OF A RELAY

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
11	17	93	93	-- 026 --	00	12	17	93		05000
									FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
			20.402(b)		20.405(c)	X	50.73(a)(2)(iv)		73.71(b)
POWER LEVEL (10)	100		20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v)		73.71(c)
			20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		OTHER
			20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		(Specify in Abstract below and in Text, NRC Form 366A)
			20.405(a)(1)(iv)		50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		
			20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
DAVID N. LORFING, SUPERVISOR - NUCLEAR LICENSING	(504) 381-4157

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
X	YES (If yes, complete EXPECTED SUBMISSION DATE).	NO		02	28	94

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

On November 17, 1993 at 2302, with the reactor in Operational Condition 1 (Power Operation) at 100 percent power, a containment isolation valve (1E51*MOVF064) inadvertently closed during a surveillance test on the reactor core isolation cooling system (RCIC). This event is reportable as an engineered safety feature actuation (ESF) pursuant to 10CFR50.73(a)(2)(iv).

GSU has concluded that a relay failure occurred which led to the isolation of the valve. However, the cause of the relay failure, and thus the root cause of the event, requires additional evaluation. The three relays associated with this RCIC isolation have been replaced and shipped to the manufacturer for failure analysis. GSU will provide a supplement to this report following this evaluation to document the results. Following replacement of the relays, the surveillance test procedure was successfully performed with the replacement relays.

All other plant equipment functioned as required during this event. Throughout this event, core cooling capability was assured since the high pressure core spray system was available.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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RIVER BEND STATION	05000458	93	-- 026 --	00	2 OF 4

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

REPORTED CONDITION

On November 17, 1993 at 2302, with the reactor in Operational Condition 1 (Power Operation) at 100 percent power, a containment isolation valve (1E51*MOVF064) (*20*) inadvertently closed during a surveillance test on the reactor core isolation cooling system (RCIC) (*BN*). This event is reportable as an engineered safety feature actuation (ESF) pursuant to 10CFR50.73(a)(2)(iv).

INVESTIGATION

The isolation occurred during the restoration portion of surveillance test procedure (STP)-207-4501, which performs a monthly channel functional test on the isolation functions of the main steam tunnel ambient temperature-high instrumentation as required by the plant Technical Specifications. The valve that stroked closed was the RCIC steam supply line outboard containment isolation valve, 1E51*MOVF064. During the STP, the isolation logic bypass switch was placed in the "bypass" position to prevent inadvertent ESF actuations. The isolation of 1E51*MOVF064 is a designed ESF isolation for high temperature conditions in the main steam tunnel. In this event, during the restoration portion of the STP, the isolation occurred when the isolation logic bypass switch was moved from the "bypass" position to the "normal" position. No high temperature conditions existed that would have initiated the isolation.

In response to sensed high temperatures, either of the parallel relays E31A*K2A or E31A*K25A are designed to energize. In series with these two relays is the bypass switch and relay E51A*K100. When relay E51A*K100 is energized by either of the parallel relays, it initiates isolation of the RCIC valve. Therefore, a failure of either relay E31A*K2A or relay E31A*K25A could cause the isolation of the valve.

GSU's troubleshooting of the isolation revealed the following:

- Baseline voltage drop readings did not reveal any unusual voltages
- Visual inspection of the relays did not indicate any deficiencies
- Subsequent performance of the STP did not reproduce the original trip

Therefore, the initial troubleshooting investigation revealed no indications of the cause of failure.

Further investigation has revealed that the cause of this event is equipment failure. Following the initial troubleshooting evaluation, all three relays were replaced. Testing was performed on the relays which revealed that relay E31A*K2A had an erratically performing contact. This relay is not tested during STP-207-4501 and should not have energized. Failures of the other two relays could have caused the isolation, but could not be reproduced during testing. Failure of relay 1E51A*K100 is unlikely due to the fact that if its contact had failed closed, the valve isolation would have occurred regardless of bypass switch position. It is more likely that the contact for either relay E31A*K2A or E31A*K25A failed closed. Once the bypass switch was restored to normal position, the energized relay (K2A or K25A) would energize K100, and cause the isolation. Based on the above, GSU concludes that relay failure is the cause of the isolation; however, the nature of the failure is not known.

A search of the maintenance history did not identify any previous failures of either relay E31A*K2A or E31A*K25A. These relays had never been replaced prior to this event. The specified service lifetime of these types of relays, based on manufacturer and model numbers, is 40 years. A search of the condition report database did not reveal any previous cases of RCIC isolations during the restoration phase of an STP.

GSU has reviewed the Nuclear Plant Reliability Data System (NPRDS) to identify similar failures of these relays throughout the nuclear industry. The search criteria specified failures of relays to open. The results of this review has revealed seven cases. Causes cited for these failures are stuck contacts, corrosion, high resistance and indeterminate.

The operating experience described above, maintenance history, condition report database search, and NPRDS review, has not provided definitive insights concerning the failure mode.

Other potential causes investigated were personnel error, procedural inadequacy, and environmental factors. At the time of the isolation, Operations was restoring the bypass switch to the "normal" position from the "bypass" position. This is a routine action by Operations personnel, performed at least twice a day. As required by the STP, Instrumentation and Controls (I&C) personnel verified that no RCIC isolation annunciation was active prior to movement of the switch. Therefore, there is no indication that a personnel error occurred, and the procedure contains the appropriate verification to prevent an isolation. Review of environmental factors revealed that the relays are located in the control room, a mild environment. There were no unusual characteristics about the relay location or switch location.

ROOT CAUSE

GSU has concluded, based on the evaluation, that a relay failure occurred which led to the closure of the containment isolation valve. However, the cause of the relay failure, and the identification of the relay that failed requires additional evaluation. GSU will provide a supplement to this report following this evaluation to document the results.

Recent LERs have documented RCIC isolations attributed to a temperature switch malfunction (LER 93-022-00) and personnel error (LER 93-018-01). The relay failure involved in this event would not be mitigated by those STP changes that were cited as corrective actions for LERs 93-018-01 and 93-022-00. The review of previous LERs revealed none similar to LER 93-026.

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

CORRECTIVE ACTION

The three relays associated with this RCIC isolation have been replaced. The STP was successfully performed as a part of the retest requirements for these relays.

The relays have been shipped to the manufacturer for failure analysis. GSU will provide a supplement to this report following this evaluation to document the results.

To investigate River Bend Station failure rates due to all causes (not limited to failures to open), a search through the material demand history for those safety related relays having the same manufacturer and model numbers as E31*K2A and E31*K25A, shows that 16 of these relays have been replaced. This is just over 1 percent of the safety related relays of these types installed at the plant. Failures have been attributed to contact failure, open coils, or equipment failure. Additional corrective actions will be evaluated following the failure analysis by the manufacturer and provided in the supplemental report.

SAFETY ASSESSMENT

All other plant equipment functioned as required during this event. Throughout the duration of this event, core cooling capability was assured since the high pressure core spray system was available.

Note: Energy Industry Identification System codes are indicated in the text as (*XX*).