NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
RIVER BEND STATION	0 5 0 0 0 4 5 8	1 OF 0 2

TITLE (4)
Manual Reactor Scram Due to High Unidentified Drywell Leakage

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 NAMES	DOCKET NUMBER(S)
0 1	1 2	8 7	8 7	0 0 2	0 0	0 2	1 1	8 7		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1
POWER LEVEL (10): 0 1 7

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
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 (Specify in Abstract below and in Text, NRC Form 366A)
20.405(a)(1)(iii)	X 50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
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 (AREA CODE)
E. R. Grant- Director, Nuclear Licensing	5 0 4 6 3 5 - 6 9 9 5

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
X	A D	V	A 3 9 1	N	X	B G	V	A 5 8 5	N
X	B O	V	A 5 8 5	N	X	C E	V	V 0 8 5	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 fifteen single-space typewritten lines) (16)

At 1532 on 1/12/87 with the unit at full power, a reactor shutdown was initiated as required by Technical Specification 3.4.3.2.b Action Statement for unidentified drywell leakage exceeding 5 gpm. The unidentified leakage began increasing at approximately 0000 on 1/12/87. At 0308 an "unusual event" was declared, and Operations began reactor shutdown procedures at 0347 in compliance with Technical Specification 3.4.3.2.b. Thereafter, the leakage remained in the 8.0 to 10.5 gpm range. Power was reduced to 17 percent prior to initiating a manual scram per approved shutdown procedures to comply with the Technical Specification Action Statement.

The unidentified drywell leak rate increasing to greater than 5 gpm was caused by packing leaks on several valves with the "B" Residual Heat Removal (RHR) testable injection check valve (1E12*AOVF041B) being the primary contributor. Packing on all valves responsible for contributing to the excessive leakage was subsequently replaced or sealed with an approved sealing compound. The valves were then inspected for leakage when the plant was returned to rated pressure. Little or no leakage was observed. The plant has continued to operate to this date with the unidentified drywell leakage remaining well below the limit allowed by the Technical Specification. At no time was the health and safety of the public affected as a result of this event.

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FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
RIVER BEND STATION	0 5 0 0 0 4 5 8	8 7	— 0 0 2	— 0 0	0 2 OF 0 2

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

REPORTED CONDITION

At 1532 on 1/12/87 with the unit at full power, a reactor shutdown was initiated as required by Technical Specification 3.4.3.2.b Action Statement for unidentified drywell leakage exceeding 5 gpm. The unidentified leakage began increasing at approximately 0000 on 1/12/87. At 0308 an "unusual event" was declared, and Operations began reactor shutdown procedures at 0347 in compliance with Technical Specification 3.4.3.2.b. Thereafter, the leakage remained in the 8.0 to 10.5 gpm range. Power was reduced to 17 percent prior to initiating a manual scram per approved shutdown procedures to comply with the Technical Specification Action Statement.

INVESTIGATION AND CORRECTIVE ACTION

Later investigation showed that the high drywell leak rate was caused by several packing leaks on valves located within the drywell structure. The most significant leak was on the "B" Residual Heat Removal (RHR) testable injection check valve (1E12*AOVF041B). Three other valves were noted as minor contributors to the leakage. These included, High Pressure Core Spray testable injection check valve (1E22*AOVF005), Reactor Recirculation System suction valve (1B33*MOVF023B), and Reactor Water Cleanup system suction isolation bypass valve (1G33*VF103). With the exception of 1G33*VF103, the valves were repacked subsequently utilizing approved maintenance procedures. Valve 1G33*VF103 could not be isolated from the reactor vessel to allow packing replacement and was therefore sealed by injecting an approved nuclear grade sealing compound into its packing chamber. When the reactor was returned to rated pressure, the valves were reinspected to verify that leakage had stopped. This verification showed little or no leakage present, and the unidentified drywell leak rate has since remained well below the limit allowed by the Technical Specifications.

SAFETY ASSESSMENT

The 5 gpm unidentified leak rate limit is based upon the predicted and experimentally observed behavior of cracks in pipes. The evidence obtained from these experiments suggests that, for leakage somewhat greater than the 5 gpm limit, the probability is small that the imperfection or crack associated with this leakage would grow rapidly. Since the leakage was determined to originate from valve packing and not from piping imperfections or cracks, there was no impact on the health and safety of the public as a result of this event.

GULF STATES UTILITIES COMPANY
RIVER BEND STATION POST OFFICE BOX 220 ST. FRANCISVILLE, LOUISIANA 70775
AREA CODE 504 635-6094 346-8651

February 11, 1987
RBG- 35406
File Nos. G9.5, G9.25.1.3

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

Gentlemen:

River Bend Station - Unit 1
Docket No. 50-458

Please find enclosed Licensee Event Report No. 87-002 for River Bend Station - Unit 1. This report is being submitted pursuant to 10CFR50.73.

Sincerely,

J. E. Booker

J. E. Booker
Manager-River Bend Oversight
River Bend Nuclear Group

JEB/TFP/PDG/RRS/je

cc: U.S. Nuclear Regulatory Commission
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