GULF STATES UTILITIES COMPANY

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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. 92-007 for River Bend Station - Unit 1. This report is submitted pursuant 10CFR50.73.

Sincerely,

DM Lorfing
for W.H. Odell
Manager - Oversight
River Bend Nuclear Group

LAE/PDG/EMC/DCH/RGG/WDS/kvm

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NRC FORM 366
(6-89)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

APPROVED OMB NO. 3150-0104
EXPIRES ..0/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
RIVER BEND STATION	0 5 0 0 0 4 5 8	1 OF 0 9

TITLE (4): MOTOR OPERATED VALVE VULNERABILITY TO HOT SHORTS DISCOVERED DURING REVIEW OF NRC INFORMATION NOTICE 92-18

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 3	2 6	9 2	9 2	0 0 7	0 0	0 4	2 7	9 2		0 5 0 0 0
										0 5 0 0 0

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

OPERATING MODE (9)				
POWER LEVEL (10): 1 0 0	[ ] 20.402(b)	[ ] 20.405(c)	[ ] 50.73(a)(2)(iv)	[ ] 73.71(b)
	[ ] 20.405(a)(1)(i)	[ ] 50.36(c)(1)	[X] 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)	[ ] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	
	[ ] 20.405(a)(1)(iv)	[X] 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)
L.A. ENGLAND, DIRECTOR - NUCLEAR LICENSING	504 381-4145

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
[ ] YES (If yes, complete EXPECTED SUBMISSION DATE)	[X] NO				

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

On 3/26/92, during a review of NRC Information Notice 92-18, a design deficiency was identified in the control circuits for motor operated valves (MOVs) (*20*) required for alternate shutdown of the plant. These control circuits could operate spuriously during a control room fire. If a fire in the control room forces reactor operators to evacuate the control room, these MOVs can be operated from the remote shutdown panel. However, energized short circuits ("hot shorts") combined with the absence of thermal overload protection, could permit bypassing of the torque switch and limit switches, and thus cause valve damage before operators are able to transfer control of the valves to the remote shutdown panel. This design is contrary to the River Bend Station Fire Hazards Analysis and constitutes a condition outside the design basis. Therefore, this report is submitted pursuant to 10CFR50.73(a)(2)(ii)(B).

The control circuit design deficiency identified by NRC Information Notice 92-18 is an emerging issue in the nuclear industry. A contributing factor was the lack of thermal overload protection as specified in Regulatory Guide 1.106. Typical control circuits are designed with thermal overload protection to protect the motor operator. The special application of a motor operated valve required for alternate shutdown combined with the Regulatory Guide 1.106 design to bypass the thermal overloads resulted in a design deficiency.

The control circuitry for the 50 affected MOVs will be reworked so that the limit switches and torque switches cannot be bypassed by hot shorts.

REPORTED CONDITION

On 3/26/92, during a review of NRC Information Notice 92-18, a design deficiency was identified in the control circuits for motor operated valves (MOVs) required for alternate shutdown of the plant. These control circuits could operate spuriously during a control room fire. If a fire in the control room forces reactor operators to evacuate the control room, these MOVs can be operated from the remote shutdown panel. However, energized short circuits ("hot shorts") combined with the absence of thermal overload protection, could permit bypassing of the torque switch and limit switches, and thus cause valve damage before operators are able to transfer control of the valves to the remote shutdown panel. This design is contrary to the River Bend Station Fire Hazards Analysis and constitutes a condition outside the design basis. Therefore, this report is submitted pursuant to 10CFR50.73(a)(2)(ii)(B).

INVESTIGATION

A review of the design bases for motor operated valves with remote shutdown capabilities was performed in conjunction with the evaluation of Information Notice 92-18 "Potential for Loss of Remote Shutdown Capability During a Control Room Fire". The review included research of the USAR, the SER, the Technical Specifications, Regulatory Guides, 10CFR50 Appendix R, Fire Hazard Analysis (FHA) Criteria 240.201, Stone and Webster Engineering technical guidelines, and sample elementary diagrams (ESKs) of suspect motor operated valves.

USAR section 9.5.1 Appendix 9B, section 9B.4.12 refers to River Bend compliance with 10CFR50 Appendix R, section III.L "Alternative and Dedicated Shutdown Capability". This section states:

> "The equipment required for these alternative methods has been analyzed to assure that it is independent of the fire area being evaluated, or that acceptable fire protection is provided."

The River Bend Station Fire Hazards Analysis (FHA), criteria 240.201 identifies the main control room (Fire area C-25) as an area where alternative shutdown capability is provided. FHA table 3 (Method IE- Main Control Room Fire Required Items) lists specific equipment (both active and passive) as being required and independent of a fire in the control room. The review of the Fire Hazards Analysis identified fifty (50) motor operated valves that are susceptible to the hot short failure mode described in Information Notice 92-18. These valves are listed in Attachment 1. The affected systems are as follows:

. Residual heat removal (*BO*) - 15 MOVs

. Standby service water (*BI*) - 19 MOVs

. Reactor core isolation cooling (*BN*) - 13 MOVs

. Automatic depressurization system (ADS) - 1 MOV
. Chilled water systems - control building and turbine building (*KM*) - 2 MOVs

Pursuant to the guidance in Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor Operated Valves," thermal overload protection is not provided for safety related MOVs in the safety related stroke direction at River Bend. The thermal overload protection is bypassed to ensure that the overload protection does not prevent MOVs from performing their safety-related functions during an accident. Table 1.8-1 of the USAR states that River Bend complies with this regulatory guide.

ROOT CAUSE

The control circuit design deficiency identified by NRC Information Notice 92-18 is an emerging issue in the nuclear industry. A contributing factor was the lack of thermal overload protection (a configuration specified in Regulatory Guide 1.106). Typical control circuits are designed with thermal overload protection to protect the motor operator. The special application of a motor operated valve required for alternate shutdown combined with the Regulatory Guide 1.106 design to bypass the thermal overloads resulted in a design deficiency.

A review of previous LERs revealed no similar events.

CORRECTIVE ACTION

Analysis of the sample of ESKs and associated wiring drawings for the motor control centers and remote shutdown panel revealed a method to rewire the control circuitry of a motor operated valve so that the torque and limit switches in the valve operators are not bypassed by the hot short. Analysis of the wiring diagrams indicates the modification can be performed with no additional field cable installation. This modification technique would require wiring changes at the motor control center and the remote shutdown panel. No wiring revisions are required at the torque and limit switches in the valve operators. The LLRT and signature testing on the MOVs associated with Generic Letter 89-10 will not be impacted by this modification.

The corrective action for this condition is to rework the control circuitry wiring for the fifty (50) MOVs that are susceptible as described above. Fifteen residual heat removal MOVs will be modified by MR 92-0040, nineteen service water MOVs will be modified by MR 92-0043, thirteen reactor core isolation cooling valves will be modified by MR 92-0044, one ADS MOV will be modified by MR 92-0042, and two chilled water system MOVs will be modified by MR 92-0041. These modifications will be implemented during the fourth refueling outage, now in progress.

SAFETY ASSESSMENT

Analysis Methodology

The valves are primarily in the residual heat removal, reactor core isolation cooling, and standby service water systems. The ESK for each of the MOVs was reviewed to identify the main control room panels that contained wiring associated with each valve such that a fire in the panel could result in a hot short affecting the valve. The function of each valve for safe shutdown was then identified, along with its normal and safe shutdown position. This information was gathered from a review of system and abnormal operating procedures. The MOVs were also compared against the Revision 0 Probabilistic Risk Assessment for River Bend [Reference 1]. Those MOVs that were not found in the PRA were evaluated to determine if spurious operation could impact the operation of the systems. Spurious operation of many of these MOVs was analyzed in Reference 2 and shown to be insignificant. These valves were removed from further consideration in this analysis. The remaining valves that did not appear in the PRA model were analyzed and found to be conservatively removed from the PRA model (i.e., no credit was taken for their success in the model). These valves were also removed from further consideration. The remaining MOVs, 38 in all, were analyzed to determine the potential core damage frequency due to hot shorts resulting from main control room (MCR) panel fires. These MOVs are followed by asterisks in Attachment 1.

The 38 MOVs were grouped according to the MCR panels that contained their wiring. A total of 15 MCR panels were identified. The panels affecting the majority of valves were P601 and P715 which impacted nearly all of the RHR and RCIC valves and P870 and P731 which impacted nearly all of the SSW valves. The frequency of a fire in any one MCR panel was then established using the information from the Kosheng PRA [Reference 3] to estimate the frequency of a MCR panel fire. From this reference, the frequency of an MCR panel fire that had the potential to induce hot shorts was determined to be 9.35E-5 per year. This value includes credit for automatic fire suppression systems but not for manual fire suppression.

The next step in the analysis was the development of an event tree to define the successful and non-successful combinations of mitigating systems available to achieve a safe shutdown. Per the RBS FHA and the instructions in AOP-31, limited credit was given for Division II equipment and no credit for non-safety related systems. The systems included in the event tree were essentially only the Division I and III ECCS systems. The event tree identified a total of 51 possible sequences with a total of 18 sequences ending in core damage.

Before the core damage sequences in the event trees could be quantified, the conditional probability of a hot short disabling a valve given a MCR panel fire had to be determined. Reference 2 gives a generic probability value of 0.1 of a hot short occurring in a panel with a fire present. Therefore, this is the conditional failure probability assigned to each of the valves having wiring located in the panel of interest. Integrated fault tree models for each sequence were then assembled using the models from Reference 1. These integrated fault tree models included not only frontline systems, but also all support systems, including HVAC required for operation of the frontline systems. The fault tree models included hardware faults, maintenance and test unavailabilities and human errors. The core damage sequences were then quantified for each of the panels in the MCR to determine the core damage frequency due to hot shorts.

Results of Analysis

The total core damage frequency (CDF) resulting from a main control room panel fire in the panels of interest to this analysis was estimated to be 1.02E-5 per year. As a means of comparison, the total core damage frequency due to an MCR fire in any panel was determined to be 2.9E-5 per year. Therefore the hot short phenomena would only contribute 7.55E-6 per year or 26% of the total CDF due to MCR fires. Also, from Reference 1, the total Revision 0 PRA model CDF due to internal events for River Bend is 6.13E-5 per year. Therefore, the hot short CDF is only 12.3% of the CDF due to internal events.

The CDFs for fires in the different panels also vary widely. A fire in panel P870 resulted in a total CDF of 3.92E-6 per year and a CDF due to hot shorts of 3.63E-6 per year while a fire in panel P731 resulted in a total CDF of 3.62E-6 per year and a CDF due to hot shorts of 3.33E-6 per year. Therefore, these two panels alone accounted for a CDF of 6.96E-6 per year or 92 % of the total due to hot shorts. This is understandable since these panels affect the MOVs in the standby service water system which supplies cooling functions to the majority of the frontline mitigating systems. In contrast the CDF for a fire in panel P601, which affects RHR and RCIC valves, is only 3.26E-7 per year or 3% of the total.

Importance ranking of components whose failure contributed to the CDF was done for each of the panels. The most important contributors to CDF as measured by the Fussel-Vesely factor were the standby service water MOVs 171, 172, 74A, and 74B. These valves all are associated with service water to the Division I auxiliary building unit coolers and the HPCS unit cooler. The most important contributors as measured by risk achievement (i.e., the components that would most affect CDF if their failure frequency increased) are the frequency of the panel fire itself and the fans, dampers, valves and controls associated with switchgear room cooling. These results point out the perceived importance of HVAC and in particular switchgear room cooling in the Revision 0 PRA models. It should be noted that the PRA is currently being revised and one area of improvement is the dependence on HVAC. Analysis has been performed that allows for recovery of loss of switchgear room cooling and this will be included in the future models. Therefore, the current results are conservative and it is expected that both the interval events CDF and the MCR fire CDF will decrease when the Rev. 1 PRA models are used.

Conclusions

The core damage frequency due to hot shorts generated by fires in main control panels at River Bend contributes only 12.3% to the total core damage frequency. This does not represent a significant increase in plant risk since this increase is within the bounds of the uncertainty of the analysis. These results are conservative. In addition, this contribution to the total core damage frequency is not as significant as the contribution due to fires in MCR panels (without considering the hot short phenomena). As a requirement of Generic Letter 88-20, Supplement 4, River Bend will shortly be performing the Individual Plant Examination (IPE) for external events which will include a detailed fire risk analysis. This issue of hot shorts in MCR panels will be addressed in that analysis in a more detailed and less conservative manner than the analysis performed here. It is expected that the results of that analysis will identify the true importance of this issue in relation to overall fire risk.

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

ATTACHMENT 1

Motor Operated Valve Control Circuits to be Reworked

RESIDUAL HEAT REMOVAL:

1E12*MOVF003A*
1E12*MOVF004A*
1E12*MOVF006A*
1E12*MOVF008*
1E12*MOVF011A
1E12*MOVF023
1E12*MOVF024A*
1E12*MOVF027A*
1E12*MOVF040
1E12*MOVF042A*
1E12*MOVF047A*
1E12*MOVF048A*
1E12*MOVF053A*
1E12*MOVF064A*
1E12*MOVF068A*

REACTOR CORE ISOLATION COOLING:

1E51*MOVC002*
1E51*MOVF010*
1E51*MOVF013*
1E51*MOVF019*
1E51*MOVF022
1E51*MOVF031*
1E51*MOVF045*
1E51*MOVF046*
1E51*MOVF063*
1E51*MOVF064*
1E51*MOVF068*
1E51*MOVF077
1E51*MOVF078

ATTACHMENT 1 (CONT.)

CHILLED WATER SYSTEMS:

1HVK*MOV20A*
1HVN*MOV22A

AUTOMATIC DEPRESSURIZATION SYSTEM:

1SVV*MOV1A

STANDBY SERVICE WATER:

1SWP*MOV40A
1SWP*MOV40C
1SWP*MOV55A*
1SWP*MOV73A*
1SWP*MOV74A*
1SWP*MOV74B*
1SWP*MOV77A*
1SWP*MOV81A
1SWP*MOV96A*
1SWP*MOV171*
1SWP*MOV172*
1SWP*MOV501A
1SWP*MOV502A*
1SWP*MOV503A*
1SWP*MOV504A*
1SWP*MOV506A*
1SWP*MOV506B*
1SWP*MOV507A*
1SWP*MOV510A*

Note: The 38 MOVs followed by *s were analyzed to determine the potential core damage frequency due to hot shorts resulting from MCR panel fires.

NRC FORM 366A (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

APPROVED OMB NO. 3150-0104
EXPIRES 4/30/92

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

References

1. "Analysis of Core Damage Frequency from Internal Events: River Bend Station, Unit 1 for Individual Plant Examination", PRA/Radiological Analysis Group, Report Number EA-RA-91-0004-MP, Revision 0, February 28, 1992.

2. Memorandum from J. S. Miller to C. M. Coones, "Spurious Signals During a Fire Event", EA-M-90-017, January 18, 1989.

3. "Probabilistic Risk Assessment Kosheng Nuclear Power Station, Unit 1", July 1985.