GULF STATES UTILITIES COMPANY

RIVER BEND STATION POST OFFICE BOX 220 ST. FRANCISVILLE, LOUISIANA 70775

AREA CODE 504 635-6094 346-8651

September 4, 1992

RBG- 37,446

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

Sincerely,

W.H. Odell
Manager - Oversight
River Bend Nuclear Group

WAE/PDG/FRC/JHM/JPM/JLB/kvm

IE22

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NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

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

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.

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 1 0

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 5 0 0 0
0 3	2 6	9 2	9 2	— 0 0 7 —	0 1	0 9	0 4	9 2		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): 5

POWER LEVEL (10): 0

[ ] 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)	[ ] 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	5 0 4 3 8 1 - 4 1 4 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

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) 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 46 affected MOVs will be reworked so that the limit switches and torque switches cannot be bypassed by hot shorts in the control room.

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 46 motor operated valves (not 50 as reported in Rev. 0 of this LER) 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*) - 16 MOVs

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

. Automatic depressurization system (ADS) - 1 MOV

. Chilled water systems - control building and turbine building (*KM*) - 1 MOV

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 46 MOVs that are susceptible as described above. Fifteen residual heat removal MOVs will be modified by MR 92-0040, 16 service water MOVs will be modified by MR 92-0043, 13 reactor core isolation cooling valves will be modified by MR 92-0044, one ADS MOV will be modified by MR 92-0042, and one chilled water system MOV will be modified by MR 92-0041. These modifications will be implemented during the fourth refueling outage as listed by maintenance work orders on Attachment 1 except the following 12 MOVs which will be deferred until RF-5:

1HVN*MOV22A	1SWP*MOV73A
1SWP*MOV74A	1SWP*MOV74B
1SWP*MOV77A	1SWP*MOV81A
1SWP*MOV501A	1SWP*MOV504A
1SWP*MOV506A	1SWP*MOV506B
1SWP*MOV510A	1SWP*MOV511A

SAFETY ASSESSMENT

Analysis Methodology

The valves listed in Revision 0 of LER 92-007 included RHR, RCIC and standby service water MOVs, while the MOVs shown in Attachment 2 of this analysis are primarily in the service water system. This analysis which makes use of the previous work done for LER 92-007 Revision 0 is described below. The ESK for each MOV 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 operation procedures. The MOVs were compared against the Revision 0 Probabilistic Risk Assessment (PRA) for River Bend. Those MOVs that were not found in the PRA were evaluated to determine if spurious operation could impact the operation of the systems. The 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 removed from further consideration. The remaining MOVs were analyzed to determine the potential core damage frequency due to hot shorts resulting from main control room (MCR) panel fires.

The analysis performed for LER 92-007 used the Revision 0 PRA model whereas this analysis uses a new and updated Revision 1 PRA model. The Revision 1 PRA model differs from the Revision 0 model in that it incorporates changes in hardware and changes in modeling assumptions. The most significant changes has to do with the impact of the new closed service water system and assumptions that govern the success of the standby service water system.

The remaining MOVs were grouped according to the MCR panels that contained their respective wiring. Panels P870 and P731 contained wiring for all of the remaining valves and were the only panels which were evaluated. The frequency of a fire in any one MCR panel was then established using the information from the Kuosheng 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-05 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 was given 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 the revision 1 PRA. These integrated fault tree models included frontline systems and all support systems, including HVAC required for operation of the frontline systems.

Results of Analysis

The Safety Assessment performed for Revision 0 of this document used the River Bend PRA Revision 0 model. The analysis performed for this assessment uses a new and refined River Bend PRA Revision 1 model. As seen in the table below, refinements in the model have caused both the total CDF and the hot shorts CDF to decrease.

	Total CDF [/year]	Hot Shorts CDF [/year]	Hot Snort Percent of Total CDF [%]
Rev 0 Analysis	6.1E-05	7.0E-06	12
This Analysis	1.5E-05	4.5E-06	30

While the refinements in the model have caused the relative importance of the hot shorts phenomenon to increase, it should be noted that this increase is not significant because of the statistical uncertainty associated with the analysis.

Due to modeling changes and actual hardware changes to the service water system, the valves which are most important from a core damage standpoint have changed slightly from those previously identified. This analysis shows that 8 of the service water valves out of the 19 valves in attachment 2 contribute essentially all of the risk from hot shorts. The other 11 valves do not have a significant impact on risk. Those 8 risk important valves, in order of importance are:

1. 1SWP*MOV96A
2. 1SWP*MOV55A
3. 1SWP*MOV171
4. 1SWP*MOV172
5. 1SWP*MOV507A
6. 1SWP*MOV81A
7. 1SWP*MOV502A
8. 1SWP*MOV503A

Modifying the circuitry of the 8 valves specified above virtually eliminates the core damage contribution from hot shorts in the main control room. Modifying circuitry of the 4 most important valves (1 through 4 above) reduces the CDF from hot shorts to 9.4E-07/year or to about 6 percent of the CDF from internal events. For the rest of the valves contained in Attachment 1, the hot short phenomenon is not a significant risk contributor.

Since this analysis has been performed, River Bend has taken an aggressive approach to modifying the valves most important to risk, as the circuitry of the top 5 valves has been modified. The CDF due to hot shorts in the remaining valves is less than 9.4E-07/year. It is therefore acceptable to defer the modifications of the remaining valves until RF-5.

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RIVER BEND STATION	0 5 0 0 0 4 5 8	9 2	— 0 0 7 —	0 1	0 6 OF 1 0

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

ATTACHMENT 1

Motor Operated Valve Control Circuits Work Completed in RF4 or to be Deferred to RF5 as indicated:

RESIDUAL HEAT REMOVAL (MR 92-0040):

	MWO #	
1E12*MOVF003A	R150846	COMPLETED
1E12*MOVF004A	R150847	COMPLETED
1E12*MOVF006A	R150848	COMPLETED
1E12*MOVF008	R150849	COMPLETED
1E12*MOVF011A	R150850	COMPLETED
1E12*MOVF023	R150851	COMPLETED
1E12*MOVF024A	R150852	COMPLETED
1E12*MOVF027A	R150853	COMPLETED
1E12*MOVF040	R150854	COMPLETED
1E12*MOVF042A	R150855	COMPLETED
1E12*MOVF047A	R150856	COMPLETED
1E12*MOVF048A	R150857	COMPLETED
1E12*MOVF053A	R150858	COMPLETED
1E12*MOVF064A	R150859	COMPLETED
1E12*MOVF068A	R152089	COMPLETED

REACTOR CORE ISOLATION COOLING (MR 92-0044):

	MWO #	
1E51*MOVC002	R150845	COMPLETED
1E51*MOVF010	R150833	COMPLETED
1E51*MOVF013	R150834	COMPLETED
1E51*MOVF019	R150835	COMPLETED
1E51*MOVF022	R150836	COMPLETED
1E51*MOVF031	R150837	COMPLETED
1E51*MOVF045	R150838	COMPLETED
1E51*MOVF046	R150839	COMPLETED
1E51*MOVF063	R150840	COMPLETED
1E51*MOVF064	R150841	COMPLETED
1E51*MOVF068	R150842	COMPLETED
1E51*MOVF077	R150843	COMPLETED
1E51*MOVF078	R150844	COMPLETED

NRC FORM 366A (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

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

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.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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	9 2	— 0 0 7 —	0 1	0 6 OF 1 0

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

ATTACHMENT 1

Motor Operated Valve Control Circuits Work Completed in RF4 or to be Deferred to RF5 as indicated:

RESIDUAL HEAT REMOVAL (MR 92-0040):

	MWO #	
1E12*MOVF003A	R150846	COMPLETED
1E12*MOVF004A	R150847	COMPLETED
1E12*MOVF006A	R150848	COMPLETED
1E12*MOVF008	R150849	COMPLETED
1E12*MOVF011A	R150850	COMPLETED
1E12*MOVF023	R150851	COMPLETED
1E12*MOVF024A	R150852	COMPLETED
1E12*MOVF027A	R150853	COMPLETED
1E12*MOVF040	R150854	COMPLETED
1E12*MOVF042A	R150855	COMPLETED
1E12*MOVF047A	R150856	COMPLETED
1E12*MOVF048A	R150857	COMPLETED
1E12*MOVF053A	R150858	COMPLETED
1E12*MOVF064A	R150859	COMPLETED
1E12*MOVF068A	R152089	COMPLETED

REACTOR CORE ISOLATION COOLING (MR 92-0044):

	MWO #	
1E51*MOVC002	R150845	COMPLETED
1E51*MOVF010	R150833	COMPLETED
1E51*MOVF013	R150834	COMPLETED
1E51*MOVF019	R150835	COMPLETED
1E51*MOVF022	R150836	COMPLETED
1E51*MOVF031	R150837	COMPLETED
1E51*MOVF045	R150838	COMPLETED
1E51*MOVF046	R150839	COMPLETED
1E51*MOVF063	R150840	COMPLETED
1E51*MOVF064	R150841	COMPLETED
1E51*MOVF068	R150842	COMPLETED
1E51*MOVF077	R150843	COMPLETED
1E51*MOVF078	R150844	COMPLETED

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR - SEQUENTIAL NUMBER - REVISION NUMBER	PAGE (3)
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TEXT (If more space is required, use additional NRC Form 366A's) (17)

ATTACHMENT 1 (CONTINUED)

CHILLED WATER SYSTEMS (MR 92-0041):

1HVN*MOV22A	MWO-R159661	DEFERRED TO RF-5

AUTOMATIC DEPRESSURIZATION SYSTEM (MR 92-0042):

1SVV*MOV1A	MWO-R152085	COMPLETED

STANDBY SERVICE WATER (MR 92-0043):

1SWP*MOV55A	MWO-R159662	COMPLETED
1SWP*MOV73A		DEFERRED TO RF-5
1SWP*MOV74A		DEFERRED TO RF-5
1SWP*MOV74B		DEFERRED TO RF-5
1SWP*MOV77A		DEFERRED TO RF-5
1SWP*MOV81A		DEFERRED TO RF-5
1SWP*MOV96A	MWO-R159670	COMPLETED
1SWP*MOV171	MWO-R159671	COMPLETED
1SWP*MOV172	MWO-R159672	COMPLETED
1SWP*MOV501A		DEFERRED TO RF-5
1SWP*MOV504A		DEFERRED TO RF-5
1SWP*MOV506A		DEFERRED TO RF-5
1SWP*MOV506B		DEFERRED TO RF-5
1SWP*MOV507A	MWO-R159679	COMPLETED
1SWP*MOV510A		DEFERRED TO RF-5
1SWP*MOV511A		DEFERRED TO RF-5 (ADDED TO LIST DURING MR IMPLEMENTATION)

NRC FORM 366A (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

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

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.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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	9 2	— 0 0 7 —	0 1	0 8 OF 1 0

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

ATTACHMENT 1 (CONTINUED)

The following valves were originally listed as requiring alteration but have been subsequently determined to not need the modifications:

1HVK*MOV20A 1SWP*MOV40A 1SWP*MOV40C	MODIFICATION NOT REQUIRED SINCE CONTROL OF THE MOV NOT IMPACTED BY THE HOT SHORT PHENOMENON (NO CONTROL ROOM INTERFACE EXCEPT INDICATING LIGHTS).
1SWP*MOV502A 1SWP*MOV503A	MODIFICATION NOT REQUIRED DUE TO MR 92-0034 (FIRE HAZARD ANALYSIS REVISION).

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	9 2	— 0 0 7 —	0 1	0 9 OF 1 0

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

ATTACHMENT 2

MOV'S CONSIDERED IN SAFETY ASSESSMENT

1HVN*MOV22A	1SWP*MOV55A	1SWP*MOV503A
1SWP*MOV73A	1SWP*MOV74A	1SWP*MOV74B
1SWP*MOV77A	1SWP*MOV81A	1SWP*MOV96A
1SWP*MOV171	1SWP*MOV172	1SWP*MOV501
1SWP*MOV504A	1SWP*MOV506A	1SWP*MOV506B
1SWP*MOV507A	1SWP*MOV510A	1SWP*MOV511A
1SWP*MOV502A		

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 Kuosheng Nuclear Power Station, Unit 1", July 1985.