GULF STATES UTILITIES COMPANY

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

AREA CODE 504 635-6094 346-8651

December 10, 1990
RBG- 34134
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. 90-040 for River Bend Station - Unit 1. This report is being submitted pursuant to 10CFR50.73.

Sincerely,

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

JAE/PDG/DEJ/JHM/REC/pg

cc: U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000
Arlington, TX 76011

NRC Resident Inspector
P.O. Box 1051
St. Francisville, LA 70775

INPO Records Center
1100 Circle 75 Parkway
Atlanta, GA 30339-3064

Mr. C. R. Oberg
Public Utility Commission of Texas
7800 Shoal Creek Blvd., Suite 400 North
Austin, TX 78757

IE22

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 0 3

TITLE (4)
Division I Balance-of-Plant Isolation due to an Error by Design Engineers

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
1 1	0 8	9 0	9 0	- 0 3 8	- 0 0	1 2	1 0	9 0		0 5 0 0 0

OPERATING MODE (9): 5
POWER LEVEL (10): 0 0

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)	X 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)	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)

At 1610 on 11/08/90 with the reactor in Operational Condition 5 (Refueling), while performing maintenance, the loosening of a common neutral connection in the 'A' reactor protection system (RPS) alternate circuit resulted in the momentary interruption of power to the RPS 'A' normal feed. This caused a Division I balance-of-plant (BOP) isolation and consequently, a momentary loss of shutdown cooling. This constitutes an engineered safety feature (ESF) actuation; therefore, this report is submitted pursuant to 10CFR50.73(a)(2)(iv).

The root cause of this event was an error by design engineers in the preparation of a field change notice (FCN). A new drawing issued by the original modification request was revised by the FCN, moving a termination to the wrong location. All electrical design engineers will receive training on this event with emphasis on attention to detail. This training will be completed by March 1, 1991.

During this event, all isolations occurred as designed. Upon restoration of valve, shutdown cooling was restored within 2 minutes and reactor vessel water temperature exhibited no change. Therefore, this event did not adversely affect the health and safety of the public.

REPORTED CONDITION

At 1610 on 11/08/90 with the reactor in Operational Condition 5 (Refueling), while performing maintenance, the loosening of a common neutral connection in the 'A' reactor protection system (RPS) alternate circuit resulted in the momentary interruption of power to the RPS 'A' normal feed. This resulted in a Division I balance-of-plant (BOP) isolation and an RPS half-scram. The Division I BOP isolation included the reactor water cleanup system (RWCU) system (*CE*) and valve 1SFC*MOV121 (loss of alternate shutdown cooling) for approximately 2 minutes. The Division I BOP isolation constitutes an engineered safety feature (ESF) actuation; therefore, this report is submitted pursuant to 10CFR50.73(a)(2)(iv).

INVESTIGATION

This event occurred during the implementation of a modification to add an annunciator relay (*30*), requiring modification of the 'A' RPS alternate circuitry. Field change notice (FCN) 2 to modification request (MR) 89-0056 required loosening the common neutral connection at terminal number JB407-3. When this was implemented, it resulted in a temporary loss of RPS 'A' normal power, resulting in the ESF actuations.

The root cause of this event was an error by design engineers in the preparation of FCN 2. A new drawing issued by this MR was revised by FCN 2, moving the termination at JB407-2 to JB407-3. The engineering task was complicated by the involvement of multiple shifts of design engineers. Nevertheless, the change was contrary to the elementary diagram, and ultimately resulted in the RPS actuation.

A review of previous LERs has revealed no similar events.

CORRECTIVE ACTION

A copy of the condition report documenting this event has been routed to the design engineers involved to remind them that careful attention is required on field change notices (FCNs) affecting non-safety-related circuits and thorough review of work performed on previous shifts is required prior to sign-off of documents. All electrical design engineers will receive training on this event with emphasis on attention to detail. This training will be completed by March 1, 1991.

SAFETY ASSESSMENT

During this event, all isolations occurred as designed. The Division I residual heat removal system (RHR) (*BO*), main steam line drains, main steam and radwaste sample systems (*KN*) were out of service;

therefore, these systems were unaffected by this event. Flow from the RHR "B" pump (*P*), providing alternate shutdown cooling flow via the spent fuel pool cooling assist mode, was interrupted upon isolation of 1SFC*MOV121. Shutdown cooling was restored within 2 minutes upon restoration of the valve and reactor vessel temperature exhibited no change. Therefore this event did not adversely affect the health and safety of the public.

NOTE: Energy Industry Identification System Codes are identified in the text as (*XX*).

GULF STATES UTILITIES COMPANY

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

AREA CODE 504 635-6094 346-8651

December 10, 1990
RBG- 34132
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. 90-038 for River Bend Station - Unit 1. This report is being submitted pursuant to 10CFR50.73.

Sincerely,

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

IAE/PDG/DEJ/DCH/JFM/pg

cc: U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000
Arlington, TX 76011

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P.O. Box 1051
St. Francisville, IA 70775

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1100 Circle 75 Parkway
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Mr. C. R. Oberg
Public Utility Commission of Texas
7800 Shoal Creek Blvd., Suite 400 North
Austin, TX 78757

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NRC FORM 366 (6-89) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
RIVER BEND STATION	0 5 0 0 0 4 6 8	1 OF 0 3

TITLE (4) Isolations During Realignment of Load Centers due to Insufficient Communication Among Personnel and Failure to Utilize the Procedure

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
1 1	0 9	9 0	9 0	0 3 9	0 0	1 2	1 0	9 0		0 5 0 0 0

OPERATING MODE (9)	5	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
POWER LEVEL (10)	1 0	20.402(b)	20.405(c)	X 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)	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 5 - 4 1 1 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 11/09/90 at 2036 with the unit in Operational Condition 5 (Refueling), with the water level greater than 23 feet above the reactor vessel flange, power was lost to 480 volt load center 1NJS-SWG1D causing loss of the 'B' reactor protection system (RPS) motor generator (MG) set. This resulted in isolations of the reactor water cleanup system (RWCU), and the Division II containment isolation valves, resulting in a loss of shutdown cooling. This report is submitted pursuant to 10CFR50.73(a)(2)(iv) to document these engineered safety feature (ESF) actuations.

Following completion of maintenance activities, operations personnel were in the process of realigning those load centers normally powered from the 'B' 13.8KV bus back to 1NPS-SWG1B. The loss of power occurred because a nuclear equipment operator did not perform the verifications required by procedure. This event was caused by inadequate communication between the control operating foreman (COF) and the NEO, and the NEO not using the procedure to perform required verifications. Training on this event will be provided to Operations personnel.

All actuations occurred as designed upon loss of power. Shutdown cooling was restored in about 6 minutes and no increase in reactor vessel temperature was observed. Therefore, this event did not adversely affect the health and safety of the public.

REPORTED CONDITION

On 11/09/90 at 2036 with the unit in Operational Condition 5 (Refueling), with the water level greater than 23 feet above the reactor vessel flange, power was lost to 480 volt load center 1NJS-SWG1D (*SWGR*) causing loss of the 'B' reactor protection system (RPS) motor generator (MG) set (*MG*). This resulted in isolation of the reactor water cleanup system (RWCU) (*CE*), and the Division II containment isolation valves (*ISV*), resulting in a loss of shutdown cooling. These isolations constitute actuations of engineered safety features (ESFs); therefore, this report is submitted pursuant to 10CFR50.73(a)(2)(iv).

INVESTIGATION

Prior to the event, maintenance on 1NPS-SWG1B (*SWGR*) required it to be de-energized and all 480 volt load centers to be cross-tied to 1NPS-SWG1A (*SWGR*). Following completion of maintenance activities, operations personnel were in the process of splitting the 480 volt load centers by realigning those load centers normally powered from the 'B' 13.8KV bus back to 1NPS-SWG1B (*SWGR*). When the nuclear equipment operator (NEO) arrived at load centers 1NJS-SWG1C and 1NJS-SWG1D, he closed normal supply breaker (*52*) 1NJS-SWG1D-ACB62 and then opened bus tie breaker 1NJS-SWG1D-ACB52 which caused a loss of power to 1NJS-SWG1D. The loss of power occurred because breaker (*52*) 1NPS-SWG1B-ACB33 was still open. This is the 13.8KV supply breaker to 480V transformer (*XFMR*) 1NJS-X1D supplying load center 1NJS-SWG1D. Operations personnel closed 1NPS-SWG1B-ACB33 (*52*) to restore power to 1NJS-SWG1D.

This event was caused by inadequate communication between the control operating foreman (COF) and the NEO, and the NEO not using the procedure to perform required verifications.

Station Operating Procedure (SOP)-0047 Section 5.3 covers restoring cross-tied 480 volt load centers. The first two steps of this section of the procedure require the operator to verify that the 13.8KV breaker supplying the 480 volt transformers is closed and to verify proper voltage on the low side of the transformer. The NEO did not verify these items because the instructions he received from the COF led him to believe that all of the 13.8KV breakers were closed. The NEO did not have a copy of the procedure with him because he considered this to be a routine evolution.

CORRECTIVE ACTION

Operations Department personnel will be trained on this event with an emphasis being placed on: 1) Assuring that instructions given to

personnel are concise. 2) Reinforcing the importance of the use and prior review of procedures in connection with routine evolutions. 3) Detailed briefings prior to performing planned configuration changes. This training will be performed during licensed operator requalification training and will be completed by March 31, 1991.

SAFETY ASSESSMENT

All actuations occurred as designed upon loss of power. Shutdown cooling was restored in about 6 minutes and no increase in reactor vessel temperature was observed. Therefore, this event did not adversely affect the health and safety of the public.

NOTE: Energy Industry Identification System Codes are identified in the text as (*XX*).

GULF STATES UTILITIES COMPANY

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

AREA CODE 504 635-6094 346-8651

December 10, 1990
RBG- 34133
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. 90-039 for River Bend Station - Unit 1. This report is being submitted pursuant to 10CFR50.73.

Sincerely,

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

WHO/JAE/PDG/DEJ/DCH/VCC/pg

cc: U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000
Arlington, TX 76011

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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)
Beaver Valley Power Station Unit 2	0 5 0 0 0 4 1 2	1 OF 0 5

TITLE (4)

Engineered Safety Features Actuations Caused By Partial Loss of Offsite Power Due to High Winds

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)
1 1	0 5	9 0	9 0	0 1 9	0 1	1 2	1 4	9 0	N/A	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)	POWER LEVEL (10)	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
1	1 0 0	20.402(b)	20.405(c)	X 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)	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
T.P. Noonan, General Manager Nuclear Operations	4 1 2 6 4 3 - 1 2 5 8

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	
C	F K	X X X X	X X X X	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)

On 11/05/90, with the Unit in Cold Shutdown, the 2A System Station Service Transformer (SSST) was being supplied by offsite power (No. 2 138KV Bus). At 1802 hours, a fault occurred on the No. 2 138 KV Bus, causing a loss of power to the "A" Train Normal and Emergency 4KV Busses. The No.1 Emergency Diesel Generator started and loaded the 2AE bus. This loss of power also caused a loss of power to Unit 2 Control Room Radiation Monitor, 2RMC*RQI201, resulting in a Control Room Emergency Breathing Air Pressurization System (CREBAPS) actuation. The cause for this event was adverse weather conditions (high winds). The air bottles were isolated at 1807 hours, after verifying a spurious signal actuation. At 1835 hours, the CREBAPS signal was reset and the air bottles were unisolated. At 1904 hours, power was restored to the 2A SSST. The Nuclear Regulatory Commission was notified at 2024 hours. There were no safety implications as a result of this event. The electrical protection circuitry functioned to restore the "A" Train Emergency 4KV Bus. Core cooling capability was available through this transient, as the 21B Residual Heat Pump was started immediately upon the loss of the 21A Residual Heat Pump. CREBAPS air bottle pressure remained above the Technical Specification limit the entire time period.

DESCRIPTION OF EVENT

On 11/05/90, with the Unit in Cold Shutdown at reactor coolant system (RCS) pressure and temperature of 100 PSIG and 84F respectively, Train "A" Priority was in effect. Train "A" Priority signifies that the Train "A" related components are being used to satisfy all Technical Specification required operable components, including the No. 1 Emergency Diesel Generator and that no maintenance activities are permitted on these components. The "A" Train Normal and Emergency 4160 Volt (4 KV) Busses were being supplied offsite power from the No. 2 138KV Bus through the 2A System Station Service Transformer (SSST) (Figure 1). During normal operations, these busses are supplied by the Unit through the 2C Unit Service Station Transformer (USST). Upon a loss of power to the USST, a fast-bus transfer to the SSST is initiated. At 1700 hours, System Operations notified the Control Room of severe wind warnings. At 1802 hours, a fault occurred on the No. 2 138 KV Bus, causing a loss of power to the 2A SSST and the 2A, 2AE and 2B 4KV Busses (the "B" Train Emergency Bus and the No. 2 Emergency Diesel Generator were available and operable at all times). This resulted in a loss of power to the following running components: 21C Charging Pump (pump was racked in on the 2AE 4KV Bus), 21A Residual Heat Removal Pump, 21A Component Cooling Water Pump, and the 21C Service Water System Pump (also racked in on the 2AE 4KV Bus). The No.1 Emergency Diesel Generator started and loaded the 2AE bus. The 21A Component Cooling Water Pump started during the diesel generator loading sequence. The 21C Charging Pump was manually started, since the 21A Charging Pump was also racked on the 2AE 4KV Bus, but its control switch was in Pull-To-Lock, and the 21C Charging Pump will not receive the automatic start signal if the preferred pump is also on the bus (design feature). The 21A Service Water Pump was manually started. The 21C Service Water Pump did not start due to the same design feature previously discussed for the Charging Pumps. The 21B Residual Heat Removal Pump was manually started (powered from the 2DF 4KV Bus, which was unaffected) approximately 30 seconds after the loss of the 21A Residual Heat Removal Pump. No increase in RCS pressure or temperature were observed. Following verification of Emergency Diesel Generator capacity the 21A Residual Heat removal Pump was restarted at 1803 hours, and the 21B Residual Heat Removal Pump was manually shutdown. System Operations was contacted regarding the loss of the 138 KV Bus. System Operations reported that a Traveling Operator had been dispatched to investigate the fault. This loss of power also caused a subsequent loss of power to the Unit 2 Control Room Radiation Monitor, 2RMC*RQI201, as it receives 120VAC power from the 2AE 4KV Bus. The deenergizing of the radiation monitor resulted in a Control Room Emergency Breathing

Air Pressurization System (CREBAPS) actuation. The air bottles were isolated at 1807 hours, after verifying a spurious signal actuation, placing Unit 1 (Control Rooms are in a common envelope) into Technical Specification 3.0.3. At 1904 hours, System Operations verified acceptability for the restoration of normal offsite power to the 2A SSST. The No. 1 Emergency Diesel generator was restored to standby after restoring and paralleling 2AE 4KV and 2A 4KV power.

CAUSE OF THE EVENT

The cause for this event was adverse weather conditions (high winds). The spurious fault was self-clearing and 138 KV power was restored automatically.

CORRECTIVE ACTIONS

The following corrective actions have been taken as a result of this event:

1. The 21B Residual Heat Removal Pump was started approximately 30 seconds following the loss of power to the 21A Residual Heat Removal Pump.

2. The air bottles were isolated at 1807 hours, after verifying a spurious signal actuation. This placed Unit 1 (Control Rooms are in a common envelope) into Technical Specification 3.0.3.

3. At 1835 hours, the CREBAPS signal was reset and the CREBAPS air bottles were unisolated. This allowed Unit 1 to exit Technical Specification 3.0.3.

4. At 1904 hours, offsite power was restored to the 2A SSST. The 2A and 2AE 4KV Busses were subsequently restored to offsite power through the 2A SSST. The No. 1 Emergency diesel generator was returned to standby service.

REPORTABILITY

The Nuclear Regulatory Commission was notified at 2024 hours in accordance with 10CFR50.72.b.2.ii. This written report is being submitted in accordance with 10CFR50.73.a.2.iv, as an event involving an Engineered Safety Features (ESF) System Actuation.

SAFETY IMPLICATIONS

There were no safety implications as a result of this event. The electrical protection circuitry functioned as designed resulting in the starting and loading of the No. 1 Emergency Diesel Generator. Core cooling capability was available throughout this transient, as the 21B Residual Heat Pump was started immediately upon the loss of the 21A Residual Heat Pump. Due to the recent core reload and the time from shutdown, minimal decay heat was present. No increases in RCS pressure or temperature were observed. CREBAPS air bottle pressure remained above the Technical Specification minimum limit the entire time period prior to isolation.

DIESEL GENERATOR RELIABILITY

In accordance with the Station Commitment to NRC Generic Letter 84-15, the reliability of the Diesel Generators based on the criteria of NUMARC 87-00, Appendix D, "EDG Reliability Program", are included.

	Last 20 Demands	Last 100 Demands
Diesel Generator 2-1	1.00	0.976 *
Diesel Generator 2-2	1.00	1.00 **

* - Reliability based on 42 Demands.
** - Reliability based on 37 Demands.

A "Demand" is considered a start of the diesel generator for normal monthly surveillance tests, refueling surveillance tests, and unexpected loss of voltage (undervoltage) starts.

PREVIOUS OCCURRENCES

The following are previously reported events involving CREBAPS actuations:

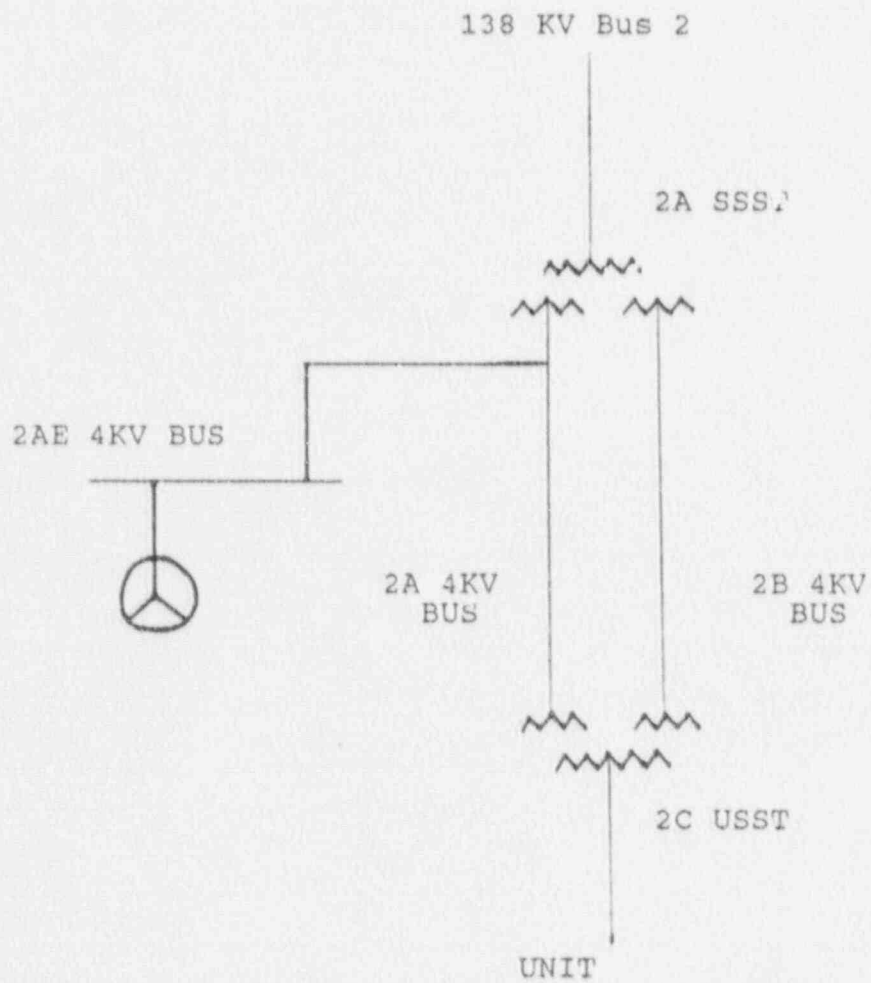
LER 88-019-00 "Inadvertent CREBAPS Actuation"
LER 89-002-00 "Inadvertent Control Room Pressurization (CREBAPS) Actuation"

The following are previously reported events a loss of power to 4KV emergency busses:

LER 87-022-00 "Automatic Start - No.1 Emergency Diesel Generator on Loss of AC Power to 2AE Emergency Bus"
LER 88-004-00 "Diesel Generator Actuation Due to Spurious Overcurrent Signal"
LER 88-005-00 "Overcurrent Relay Trip Leads to ESF Actuation"
LER 88-007-00 "Reactor Trip Due To Reactor Coolant Pump Trip Caused By a Loss Of 4KV Bus 2A Loads"
LER 89-012-00 "Loss Of Power To Train "A" Emergency Bus"

A review of the five events listed above shows four events due to component failures and one event due to personnel error during relay testing which resulted in the diesel generator loading.

Figure 1


138 KV Bus 2
2A SSST
2AE 4KV BUS
2A 4KV BUS
2B 4KV BUS
2C USST
UNIT