U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/88

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 8

TITLE (4): Reactor Scram Due To Main Generator Exciter Brush Failure

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 8	2 5	8 8	8 8	— 0 1 8 —	0 0	0 9	2 6	8 8		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): 1
POWER LEVEL (10): 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
L. A. England - Director-Nuclear Licensing	AREA CODE 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
X YES (If yes, complete EXPECTED SUBMISSION DATE)	NO		0 3	3 1	8 9

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

At 1232 on 8/25/88 with the unit at 100 percent power (Operational Condition 1), the reactor automatically scrammed due to a turbine control valve fast closure caused by a loss of main generator field excitation resulting in automatic main generator and turbine trips.

Immediately following the scram, reactor pressure spiked to a peak between 1100 and 1117 psig causing the five low-low set safety relief valves to cycle per design. The turbine bypass valves opened as required and the reactor recirculation pumps transferred to slow speed per design. Reactor water level initially decreased to +4 inches as indicated by the wide range instruments due to the reactor pressure spike. The high pressure core spray (HPCS) and reactor core isolation cooling (RCIC) systems injected as a result of a spurious low reactor water level 2 signal caused by a hydraulic perturbation in the reactor water level instrument reference lines. As a result of the feedwater flow continuing (due to the "A" feedwater control valve being in the manual mode at 50 percent open) in conjunction with the HPCS and RCIC injections, reactor water level rapidly increased to level 8 causing the HPCS injection valve and the RCIC steam supply valve to close and the reactor feedwater pumps to trip per design.

There was no significant adverse impact on the safe operation of the plant or to the health and safety of the public as a result of this event since the reactor scram placed the unit in the safe shutdown condition.

IE22
1/1

REPORTED CONDITION

At 1232 on 8/25/88 with the reactor at 100 percent power (Operational Condition 1), the main turbine (*TRB*) automatically tripped due to a generator (*TG*) trip on loss of field excitation, resulting in an automatic reactor scram due to the turbine control valve (*FCV*) fast closure.

Prior to the reactor scram, one of the main generator exciter brushes was identified as sparking. The control room exciter field volt meter (*EI*) was showing erratic readings between 33 to 70 DC volts while the voltage regulator (*RG*) was in the manual mode. Maintenance personnel were finalizing work details in preparation for replacing the worn exciter brushes when the main generator tripped on loss of field excitation.

Concurrent with the generator/turbine trip, the reactor water recirculation (*AD*) pumps (*P*) automatically transferred to the low frequency motor generator (LFMG) sets (*MG*) (slow speed) per design upon receiving an end-of-cycle recirculation pump trip (EOC RPT) signal.

Immediately following the scram reactor pressure spiked to a peak between 1100 and 1117 psig causing the five low-low set safety relief valves (SRVs) (*RV*) to cycle per design. The turbine bypass valves (*PCV*) also opened as required. The at-the-controls (ATC) operator maintained control of reactor pressure via use of the turbine bypass valves.

Reactor water level initially decreased due to the collapse of steam voids as a result of the reactor pressure spike. The lowest actual water level reached was +11 inches, +10 inches and +6 inches as indicated by the "A", "B", and "C" channel narrow range instruments (*LT*), respectively. The lowest wide range water level indication was +4 inches. However, the plant computer (*CPU*) showed evidence of a hydraulic perturbation on the wide range level instrumentation (*LT*) resulting in a low level spike in excess of -29 inches.

During the voltage transient caused by the generator trip, non-safety related 4.16 KV switchgear (*SWGR*) 1NNS-SWG1A failed to transfer (fast and slow) from normal station service transformer (*XPT*) 1STX-XNS1C to preferred station service transformer (*XPT*) 1RTX-XSR1C as a result of circuit breaker (*52*) 1NNS-ACB007 failing to close. This resulted in a loss of power to the high pressure core spray (HPCS) (*BG*) safety-related 4.16 KV bus (*EB*) 1E22*S004 and the non-safety related 4.16 KV bus (*SWGR*) 1NNS-SWG1C. Non-safety related 4.16 KV bus (*SWGR*) 1NNS-SWG1B also failed to fast transfer but successfully completed a slow transfer upon automatic closure of circuit breaker (*52*) 1NNS-ACB015.

As a result of the loss of power to the 1E22*S004 bus, the division III standby service water (SSW) (*BI*) pump (*P*) (1SWP*P2C) automatically started from a low normal service water (NSW) system (*KG*) signal caused by the loss of power to the initiating instrumentation.

Additionally, the HPCS diesel generator (*EK*) received an automatic initiation signal due to the undervoltage condition on the 1E22*S004 bus. The HPCS diesel generator successfully started and its output breaker (*52*) automatically closed, restoring voltage to the bus per design.

Due to the loss of power to 1NNS-SWG1A as described above, the turbine plant component cooling water (TPCCW) system (*KB*) pumps (*P*) 1CCS-P1A and 1CCS-P1C tripped. Operations personnel verified that 1CCS-P1B automatically started. However, this pump alone was insufficient to maintain proper cooling to the plant instrument air system (*LD*) compressors (*CMP*). As a result, the instrument air compressors tripped on high temperature. Operations personnel restarted a second TPCCW pump after restoring power to 1NNS-SWG1A. The instrument air compressors were then restarted and continued to operate properly. The lowest instrument air header pressure indicated was 80 psig. Operations personnel reported no unanticipated cycling of any air operated valves (*V*) or dampers (*DMP*).

Additionally, power to reactor protection system (RPS) (*JC*) bus "A" (*EC*) was lost. RPS bus "B" continued to operate with power supplied from the "B" RPS M-G (*MG*) set. RPS bus "A" was manually transferred to alternate supply, restoring power.

The loss of power to RPS bus "A" also resulted in an automatic initiation of the standby gas treatment (SGTS) (*BH*) and annulus mixing (AM) (*VC*) systems and an automatic trip of the annulus pressure control (APC) system (*VC*). The three systems responded as designed upon the loss of RPS power.

A spurious high drywell pressure alarm (*PA*) also actuated as a result of the loss of power to RPS bus "A". The emergency response and information system (ERIS) computer (*CPU*) recordings verified that actual drywell pressure did not exceed 0.5 psid. The high drywell pressure trip setpoint is 1.68 psid.

The HPCS and reactor core isolation cooling (RCIC) (*BN*) systems received an automatic initiation signal and injected. These initiation signals resulted from a spurious reactor water level 2 signal caused by the hydraulic perturbation on the wide range reactor water level instrumentation previously described. The controller (*PMC*) for the "A" feedwater (*LC*) control valve (*FCV*) was in the manual mode at 50 percent position. As a result of feedwater flow continuing and the HPCS and RCIC injections, reactor water level rapidly increased to level 8 causing the HPCS injection valve (*INV*) and the RCIC steam supply valve (*ISV*) to automatically close and the three reactor feedwater pumps (*P*) to trip per design. The HPCS and RCIC systems responded as designed and injected to the reactor vessel for approximately 31 and 30 seconds, respectively. A notification of unusual event (NOUE) was entered at 1253 hours based on the emergency core cooling system (ECCS) injection into the reactor vessel. The NOUE was subsequently terminated at 1320 on 8/25/88.

At approximately one hour after the reactor scram, plant personnel reported that they observed heat radiating from the HPCS injection line upstream of the HPCS injection valve. The control room was notified and further investigation and evaluation began.

INVESTIGATION

An investigation of the main generator trip determined that deterioration of the exciter brushes led to the loss of exciter field voltage resulting in the subsequent generator trip. The root cause of the exciter brush failure was determined to be a deficiency in the preventive maintenance procedure. The investigation revealed that no specific requirement had been established within the procedure as to when to replace the exciter brushes. The exciter brushes were replaced prior to initiating subsequent startup procedures. A thorough investigation of the generator and turbine revealed no other damage.

An investigation into the cause of the HPCS and RCIC injections revealed the source of the initiation signals to be a spurious hydraulic perturbation in the reactor water level instrumentation reference lines and not an actual low reactor water level 2 signal. The hydraulic perturbation was caused by the 100 percent turbine trip induced reactor steam dome pressure spike. The pressure spike was immediately transmitted to the four reactor level reference lines located near the top of the reactor vessel but was not immediately sensed by the narrow and wide range variable line taps located lower in the reactor vessel. Using the Sequence of Events Report from the plant process computer, it was determined that a HPCS low water level signal was received on channels "G", "L", and "R" for 20, 38 and 42 milliseconds, respectively. In addition, the RCIC initiation signals occurred at the same time. The instruments used to measure reactor water level for these initiation logics are Rosemount type 1154 transmitters which are fast acting instruments with no electronic dampening. As a result, the instruments caused the automatic initiation based upon sensing the pressure spike of very short duration.

Using ERIS data, HPCS and RCIC instrumentation showed significant spikes 300 milliseconds after the scram. ERIS monitors these instrument signals in 100 millisecond intervals and showed reactor water level spike magnitudes of -19, -29, -16, and -4 inches. The magnitude of the signals recorded by ERIS cannot be considered absolute due to the ERIS sample rate time intervals.

Additional evidence which shows that an actual low reactor water level 2 was not reached includes the fact that signals were not received on the following reactor water level instruments: 1) RPS level 3 scram, 2) nuclear steam supply shutoff system (*JM*) (NSSSS) level 2 balance of plant (BOP) and reactor water cleanup system (RWCU) (*CE*) equipment isolation, 3) anticipated transient without scram (ATWS) level 2 trip and 4) feedwater low level annunciator/computer points. The above signals are generated by Rosemount 1152 transmitters which have built in electronic dampening to slow the response of the instruments. These instruments also share common reference lines with the Rosemount type 1154 transmitters. Therefore, it is concluded that the initiation signals for HPCS and RCIC were generated as a result of the lack of electronic dampening in the Rosemount 1154 transmitters.

During the transient which occurred concurrent with the reactor scram, 1NNS-SWG1A failed to transfer from normal station service transformer 1STX-XNS1C to preferred station service transformer 1RTX-XSR1C. An investigation has revealed the following events led to the failure to transfer. As a result of the flashover of the main generator exciter brushes, the main generator output voltage decreased. This caused the voltage at the Fancy Point Substation to decrease and undervoltage relay (*59*) 59R-1NNSA08 to de-energize. This undervoltage relay must be energized in order for circuit breaker 1NNS-ACB07 to close to complete either a fast or slow bus transfer. Undervoltage relay 59R-INNSA08 monitors voltage on the off-site power line (*EA*) from the Fancy Point Substation which supplies preferred station service transformer 1RTX-XSR1C and ensures that sufficient voltage is available from the off-site power source prior to allowing the automatic transfer to take place.

When the generator output circuit breaker opened, the voltage at Fancy Point Substation began to stabilize. However, 10 cycles after the generator output breakers opened, relay (*62*) 62XG-1SPGN07 energized by design and disabled the fast closure portion of the 1NNS-ACB07 closing circuit. By the time the voltage at Fancy Point stabilized and undervoltage relay 59R-1NNSA08 re-energized, relay 62XG-1SPGN07 had disabled the fast closure portion of 1NNS-ACB07.

In order for 1NNS-ACB07 to close on a slow transfer, relay (*94*) 94B-1NNSA08 must be energized. Relay 94B-1NNSA08 is energized (after a short time delay) when both undervoltage relays (*27*) 27-1-1NNSA17 and 27-2-1NNSA17 sense a low voltage on the 1NNS-SWG1A bus. According to Operations personnel on duty, annunciator 0067 on panel 1H13-P808 had alarmed. This indicates that only one of the two 27 relays operated properly. Hence, 1NNS-ACB07 was also prevented from closing on a slow transfer. Additional evidence that relay 94B-1NNSA08 was

NRC Form 366A (9-83)

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

not energized is that time delay relay (*62*) 62-1NNSA08 which activates 94B-1NNSA08 did not indicate a target, and the process computer point which is also activated by the 62 relay did not operate. The 27-1 and 27-2 relays addressed in this evaluation are not used in safety related applications at River Bend Station (RBS).

Inspections of the 27-1 and 27-2 relays revealed no apparent problems although the contacts were slightly dirty. Problems have occurred in the past from dirty or oxidized contacts on these undervoltage relays. The current preventive maintenance (PM) frequency is 24 months per the vendor's recommendations. The undervoltage relays which were suspected of not properly operating were replaced. The remaining relays and timers in the transfer circuit were successfully tested.

An evaluation of the loss of power to RPS bus "A" revealed that the motor generator set output breaker tripped as a result of the voltage and frequency transients that were generated concurrent with the exciter brush failure. In addition to providing overcurrent protection, the M-G set output breaker (*52*) will trip on overvoltage or underfrequency. The RPS bus is further protected by two EPA breakers (*52*) in series with the M-G set breaker. The EPA breakers will also trip on overvoltage, undervoltage, or underfrequency.

The operator stated that prior to reset of the M-G set output breaker, the reset button had to be depressed. An analysis of the output breaker trip circuit revealed that actuation of the overvoltage relay (*59*) causes the circuit breaker undervoltage trip device (*27*) to trip the breaker and energize relay 3K, which energizes relay 2K, which is then sealed in through the reset pushbutton. Neither the underfrequency relay nor the breaker overcurrent function will energize the 2K relay requiring the reset button to be depressed prior to resetting the output breaker. Therefore, it was determined that the M-G set output breaker had tripped due to an overvoltage condition.

It cannot be determined conclusively from the evidence available if the M-G set breaker tripped prior to the EPA breaker trip or as a result of it. Since all three breakers are in series, the resulting effect on the RPS bus would have ultimately been the same.

To support restart, a preliminary evaluation was conducted to determine the impact of the elevated temperatures in the HPCS injection line. Temperatures were taken to determine the profile and maximum temperatures that the system had seen. This conservative evaluation concluded that there was no impact on the integrity of the piping system and in fact, the system could sustain at least one additional transient of this type without affecting the designed life of the piping.

Additionally, samples of the water in the injection line confirmed that the water did come from the reactor vessel. Further investigation is continuing to determine how and why the reactor water was allowed to enter the injection line.

A review of previous LERs submitted by River Bend Station revealed no previous reactor scrams as a result of a failure of the main generator exciter or brushes.

CORRECTIVE ACTION

To assure that the main generator exciter brushes are replaced before deteriorating to a point which arcing occurs, the preventive maintenance procedure is being revised to establish a specific frequency at which the brushes are to be replaced. The required procedure changes are scheduled to be completed by 10/31/88.

During the forced outage resulting from the reactor scram, maintenance personnel were trained by the vendor representative as to when and how to change the main generator exciter brushes. Following replacement of the brushes, an inspection was performed at 1800 rpm before synchronizing the generator to offsite power (*FK*) to ensure proper operation of the exciter brushes, holders, and collector rings. No abnormal conditions were observed.

An engineering evaluation of the HPCS and RCIC initiation logic is presently being conducted to determine appropriate corrective actions for preventing unnecessary level 2 initiations caused by 100 percent turbine trip induced reactor water level instrument spikes.

The undervoltage relays which led to the failure to transfer of switchgear 1NNS-SWG1A have been replaced. The current preventive maintenance tasks for cleaning the undervoltage relay contacts are scheduled on a 24 month frequency per the vendor's recommendations. Maintenance is presently revising these tasks to require a 6 month frequency. The required revisions to the preventive maintenance tasks are currently scheduled to be completed by 10/31/88.

As noted above, investigation into how the reactor water entered the HPCS injection line is still continuing. Further corrective action, if required, will be developed when this investigation is complete. A supplemental report will be provided by 3/31/89.

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SAFETY ASSESSMENT

There was no significant adverse impact on the safe operation of the plant or the health and safety of the public as a result of this event since the reactor scram placed the unit in the safe shutdown conditions. The HPCS and RCIC systems, while unnecessary, did function properly to provide reactor water makeup and all other automatic safety system actions performed as designed.

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

GULF STATES UTILITIES COMPANY

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AREA CODE 504   635-6094   346-8651

September 26, 1988
RBG-28881
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. 88-018 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/DAS/ch

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