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Georgia Power
the southern electric system

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July 24, 1992

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

PLANT HATCH - UNIT 2
NRC DOCKET 50-366
OPERATING LICENSE NPF-5
LICENSEE EVENT REPORT
PERSONNEL ERROR RESULTS IN
AN AUTOMATIC REACTOR SCRAM

Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(iv), Georgia Power Company is submitting the enclosed Licensee Event Report (LER) concerning a personnel error which resulted in an automatic reactor scram. This event occurred at Plant Hatch - Unit 2.

Sincerely,

J. T. Beckham, Jr.

J. T. Beckham, Jr.

OCV/cr

Enclosure: LER 50-366/1992-009

cc: Georgia Power Company
Mr. H. L. Sumner, General Manager - Nuclear Plant
NORMS

U.S. Nuclear Regulatory Commission, Washington, D.C.
Mr. K. Jabbour, Licensing Project Manager - Hatch

U.S. Nuclear Regulatory Commission, Region II
Mr. S. D. Ebneter, Regional Administrator
Mr. L. D. Wert, Senior Resident Inspector - Hatch

JE22

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

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

LICENSEE EVENT REPORT (LER)

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TITLE (4)
PERSONNEL ERROR RESULTS IN AN AUTOMATIC REACTOR SCRAM

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQ NUM	REV	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
0 6	2 5	9 2	9 2	0 0 9	0 0	0 7	2 4	9 2	PLANT HATCH UNIT 1	0 5 0 0 0 3 2 1
										0 5 0 0 0

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR (11)			
POWER LEVEL	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)
		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	
STEVEN B. TIPPS, MANAGER NUCLEAR SAFETY AND COMPLIANCE, HATCH	912	367-7851

COMPLETE ONE LINE FOR EACH FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORT TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORT TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (16)

On 6/25/92, at 0100 CDT, Unit 2 was in the Run mode at 2436 CMWT (100 percent rated thermal power). At that time, a licensed plant operator was in the process of transferring bus 2R24-S018A from its alternate to its normal power supply in accordance with plant procedures when he inadvertently manipulated the wrong breaker control switch resulting in a loss of power to essential 600 V bus 2C. The loss of power resulted in loss of feedwater to the vessel, a half scram, automatic closure of various Primary Containment Isolation system (PCIS) valves, and automatic initiation of the Main Control Room Environmental Control system pressurization mode. The loss of feedwater caused a rapid decrease in reactor water level resulting in a low level scram, automatic closure of Group 2 and 5 PCIS valves, a trip of the Recirculation pumps, automatic initiation of the High Pressure Coolant Injection and the Reactor Core Isolation Cooling systems, and initiation of both units' Standby Gas Treatment systems. The lowest level reached was 113.4 inches above top of the active fuel. The reactor pressure during the event did not exceed the pre-event pressure. By 0103 CDT, reactor water level and pressure were stable and at 0105 CDT, the scram was reset. Following scram recovery, at 0945 CDT, a full Reactor Protection System (RPS) actuation was received when Intermediate Range Monitor "A" (IRM) spiked upscale. The control rods were fully inserted at this time. The scram was reset at 0947 CDT.

The cause of the loss of the 600 V bus was personnel error; the licensed operator inadvertently manipulated the wrong breaker control switch. The cause of the IRM spiking is unknown. The corrective actions for this event include counseling personnel and testing the IRMs during the next refueling outage.

TEXT

PLANT AND SYSTEM IDENTIFICATION

General Electric- Boiling Water Reactor
Energy Industry Identification System codes are identified in the text as (EIIS Code XX).

DESCRIPTION OF EVENT

On 6/25/92, at 0100 CDT, Unit 2 was in the Run mode at 2436 CMWT (100 percent rated thermal power). At that time, a licensed plant operator was in the process of transferring bus 2R24-S018A from its alternate to its normal power supply in accordance with procedure 34SO-R24-003-2S, "600 Volt Essential MCC 2E-A (2R24-S018A) and MCC 2E-B (2R24-S018B) Operation." In reaching for the control switch of the alternate supply breaker to 2R24-S018A in order to open the breaker, the operator inadvertently grasped the adjacent control switch which controlled the supply breaker to Division 1 essential 600 V bus 2C (2R23-S003). Consequently, when he took the control switch to trip, instead of opening the alternate supply breaker to bus 2R24-S018A as intended, the operator opened the supply breaker to 600 V bus 2C resulting in a loss of power to the bus.

The 2C 600 V bus supplies power to, among other things, the Reactor Protection system (RPS, EIIS Code JE) Bus "A" via the "A" Motor-Generator (MG) Set, and Instrument Bus 2A. Therefore, Instrument Bus 2A lost power as did RPS bus "A". Loss of power to RPS 'A' resulted in a half scram in the RPS "A" trip system, automatic closure of various inboard Group 2 Primary Containment Isolation System (PCIS, EIIS Code JM) valves, and automatic initiation of the Main Control Room Environmental Control system (MCRECS, EIIS Code VI) pressurization mode.

Instrument Bus 2A provides power to the controls for the minimum flow isolation valves and flow control valves for the Condensate and Feedwater System (EIIS Code SJ) Condensate pumps, Condensate Booster pumps, and Reactor Feedwater pumps (RFPs). Loss of power to these controls caused the valves to fail open diverting more than half of the rated system flow directly to the Main Condenser (EIIS Code SG) hotwell.

The open minimum flow valves also caused a low suction pressure condition at the RFPs. As designed, the standby Condensate Booster pump automatically started but was not able to eliminate the low suction pressure condition because of the open minimum flow valves. Consequently, following a designed time delay, RFP "A" tripped on low suction pressure. The low suction pressure condition still did not clear due to the open minimum flow valves. Consequently, following a second time delay, RFP "B" tripped on low suction pressure.

Coincident with the feedwater transient, Reactor Recirculation System (EIIS Code AD) pump "A" received a scoop tube lockup signal and pump "B" ran back to minimum speed due to the loss of power to Instrument Bus 2A. The

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resultant core flow reducti. caused an increase in steam void formation in the core resulting in a momentary swell in reactor water level of approximately three inches. Subsequent to the swell, reactor water level decreased rapidly due to the lack of sufficient feedwater injection. Approximately 28 seconds after the loss of power to the 600 V bus, the reactor water level reached 12.3 inches above instrument zero (170.7 inches above the top of the active fuel) resulting in an automatic scram and automatic closure of the outboard Group 2 PCIS valves. Reactor water level continued to decrease. When the level reached 35 inches below instrument zero, the High Pressure Coolant Injection (HPCI, EIIS Code BJ) system and the Reactor Core Isolation Cooling (RCIC, EIIS Code BN) system automatically initiated as designed. A Group 5 PCIS actuation and a trip of the Recirculation pumps also occurred as designed. Additionally, the Standby Gas Treatment Systems (SGTS, EIIS Code BH) on Unit 1 and Unit 2 automatically started and the Secondary Containment ventilation systems on each unit isolated.

Reactor water level turned at 45 inches below instrument zero (113.4 inches above the top of the active fuel) due to HPCI and RCIC injection.

By this point in the event, a licensed operator had reclosed the supply ker to 600 V bus 2C energizing the bus and restoring power to the Condensate a Feedwater System minimum flow controls. The RFP trips were then manually reset and the "A" RFP was started. HPCI and RCIC were subsequently secured and level was restored to and maintained at the normal level of 37 inches above instrument zero using the "A" RFP.

During the event, reactor pressure did not increase above the pre-event level of 1000 psig. Pressure decreased subsequent to the reactor scram and was controlled at approximately 950 psig by the Electro-Hydraulic Control (EHC, EIIS Code JI) System using the bypass valves. Conseque ly, the Safety Relief Valves (SRVs, EIIS Code SB) were not required to open and, therefore, did not open during the event.

By 0103 CDT, reactor water level and reactor pressure were stable. At 0104 CDT, RPS bus "A" was transferred to its alternate power supply. At 0105 CDT, the scram was reset. By 0205 CDT, the Group 2 and 5 PCIS actuation signals were reset and the MCREC system was restored to its normal mode of operation.

Following recovery from the scram, at 0945 T, Unit 2 was in Hot Shutdown with all rods fully inserted and the reactor cool t temperature greater than 212 degrees Fahrenheit. Intermediate Range Moni r "H" (IRM, EIIS Code IG) which inputs to the "B" RPS trip system had previously failed upscale and had been bypassed as allowed by the Technical Specifications. Also, IRM "F" which also inputs to the "B" RPS trip system had drifted upscale causing a half scram in the "B" RPS trip system. It could not be bypassed since only one IRM in a trip system can be bypassed at the same time. Consequently, at 0945 CDT, a full RPS actuation was received when IRM "A" which inputs to the "A" RPS trip system intermittently spiked upscale. IRM "F" subsequently drifted back to its normal range and the RPS actuation was reset at 0947 CDT.

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CAUSE OF EVENT

The cause of the event was cognitive personnel error on the part of a licensed plant operator. Specifically, the individual, in attempting to perform a control board operation, inadvertently turned the wrong control switch, resulting in de-energization of Division 1 essential 600 V bus 2C. Loss of power to the bus resulted ultimately in a scram on low reactor water level.

The cause of IRM "A" spiking upscale is unknown at this time. Time domain reflectometer (TDR) testing was performed on the instrument loop in an attempt to identify any grounds or faulty connections. The diagnostic testing of the IRM detector cable indicated that an intermittent ground in the cable may exist in Primary Containment or in the Primary Containment penetration. During the next scheduled refueling outage, the cable will be checked and any necessary repairs made.

The cause of IRM "H" being upscale prior to the event is unknown at this time. A TDR test was performed on the detector cable with satisfactory results. When the cable was re-connected to the IRM pre-amp following the diagnostic test, the IRM indicated normal. However, it continues to spike intermittently. The cause cannot be determined until Primary Containment can be accessed. Consequently, during the next scheduled refueling outage, the condition will be investigated and any needed repairs will be made.

The cause of IRM "F" drifting upscale is unknown. A TDR diagnostic test was performed on it with satisfactory results. Also, a functional test was satisfactorily performed in accordance with procedure 57SV-C51-004-2S, "IRM Instrument Functional Test The IRM is now operating properly.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This report is required pursuant to 10 CFR 50.73(a)(2)(iv) because the event resulted in several unplanned automatic ESF actuations. Specifically, the loss of power to 600 V bus 2C resulted in an automatic reactor scram on low water level and several other automatic ESF actuations.

The plant responded as designed to the loss of feedwater event. The loss of feedwater injection caused a rapid decrease in the reactor water level. Approximately 28 seconds after the initiating event, the water level reached 12.3 inches above instrument zero resulting in an automatic scram and an isolation of Group 2 PCIS valves. All control rods fully inserted. The reactor water level continued to decrease as expected and within several seconds reached 35 inches below instrument zero resulting in automatic isolation of Group 5 PCIS valves, an automatic trip of both Recirculation pumps, and automatic initiation of the HPCI and RCIC systems. Also, the Standby Gas Treatment Systems (SGTS, EIIS Code BH) on Unit 1 and Unit 2 automatically started and the Secondary Containment ventilation systems on each unit isolated. All systems functioned per design. The HPCI and RCIC systems reached rated flow and began restoring water level. The lowest level reached during the event was 45 inches below instrument zero (113.4 inches above the top of the active fuel).

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As the reactor water level approached the normal range, RFP "A" was s[illegible]ted and the HPCI and RCIC systems were secured. Subsequently, level was restored and maintained at the normal level (37 inches above instrument zero) using RFP "A".

During the event, reactor pressure did not exceed the pre-event level of 1000 psig. The reactor pressure decreased as a result of the scram. Approximately one minute after the scram, the main turbine was manually tripped and the bypass valves opened, controlling pressure at [illegible]50 psig. The SRVs were not required to open and, therefore, did not operate during the event

The loss of Division 1 essential 600 V bus 2C did not prevent any safety system from performing its intended safety function. 600 V bus 2C is completely independent and redundant to the Division 2 essential 600 V bus (600 bus 2D, 2R23-S004). ESF systems are divisionally independent and redundant, including power supplies. Consequently, a loss of one division of power supply could not prevent an ESF system from performing its intended safety function. Upon the loss of 600 V bus 2C, the Division 1 trains of the following systems lost power: RPS "A", PCIS, Process Radiation Monitoring system (EIIS Code IL), Neutron Monitoring system (EIIS Code IG), and the Offgas Radiation Monitoring system (EIIS Code IL). Where possible, these systems are designed to fail-safe. That is, upon loss of power, the system performs its intended safety function. Consequently, upon loss of 600 V bus 2C, RPS "A" trip system actuated resulting in a half scram, the Group 2 PCIS air-operated inboard valves failed closed, and MCRECS automatically transferred to the pressurization mode. The PCIS motor operated inboard valves failed as is. However, the redundant, independent outboard valves as well as the Division 2 ESF trains were available to perform their safety function if the need arose.

Neutron Monitoring System IRMs provide a trip to RPS upon sensing a high neutron flux condition in order [illegible]gate the consequences of a power transient by initiating a scram. In this [illegible]ent, two IRMs spuriously failed upscale resulting in a full RPS actuation. [illegible] control rods were already fully inserted at the time of the event; therefore, no rod movement resulted from the actuation. Had a valid high neutron flux condition, or any other valid signal, requiring a scram occurred, the RPS was al[illegible] in the "safe" condition.

In summary, the ESF systems [illegible]nctioned as designed, maintaining the plant well [illegible]in its safety limits. [illegible]s therefore concluded that this event had no [illegible]e impact on nuclear safety. This safety assessment applies to all [illegible]ng conditions.

CORRECTIVE ACTIONS

[illegible] responsible individual was counseled.

A time domain reflectometer test was performed on IRMs "A", "F", and "H" in an at[illegible] to locate grounds or bad connections in the instrument loops.

During the next refueling outage scheduled to begin 9/16/92, the problems associated with IRMs "A" and "H" will be investigated and repairs will be made as necessary.

TEXT

ADDITIONAL INFORMATION

Similar events in the previous two years in which personnel error resulted in a scram were reported in the following LERs:

50-321/90-11, dated 6/22/91
50-321/91-07, dated 3/27/91
50-321/91-17, dated 10/9/91
50-321/92-09, dated 4/23/92
50-366/91-05, dated 3/15/91

Corrective actions for these previous similar events included counseling of personnel. Personnel counseling is intended to heighten one's awareness in a specific area and helps in preventing additional personnel errors; however, it is understood that personnel counseling does not totally eliminate such errors; consequently, corrective actions for these previous similar events would not necessarily have prevented this scram.

One previous similar event occurred in the last two years in which an IRM spiked upscale spuriously resulting in an RPS actuation. The event was reported in LER 50-3[illegible], dated 6/3/91. In that event, the cause of the spurious spike could not be determined; therefore, no corrective actions to prevent recurrence could be taken.

Failed Component Information: At this time, no actual failed components have been identified.