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J. T. Beckham, Jr.
Vice President - Nuclear
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Georgia Power
the southern electric system

September 28, 1995

Docket No. 50-321 HL-5037

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

Edwin I. Hatch Nuclear Plant - Unit 1
Licensee Event Report
Ground on 600-Volt Bus Affects
High Pressure Coolant Injection and Reactor Protection Systems

Gentlemen:

In accordance with the requirements of 10 CFR 50.73 (a)(2)(iv) and (v), Georgia Power Company is submitting the enclosed Licensee Event Report (LER) concerning a trip of nonessential loads powered from an alternating current 600-volt bus and a trip of the Reactor Protection System power supply.

Sincerely,

J. T. Beckham, Jr.

J. T. Beckham, Jr.

JKB/eb

Enclosure: LER 50-321/1995-005

cc: *Georgia Power Company*
Mr. H. L. Sumner, Jr., Nuclear Plant General Manager
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. B. L. Holbrook, Senior Resident Inspector - Hatch

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NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Edwin I. Hatch Nuclear Plant - Unit 1	0 5 0 0 0 3 2 1	1 OF 7

TITLE (4)
Ground on 600-Volt Bus Affects High Pressure Coolant Injection and Reactor Protection Systems

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAME	DOCKET NUMBER(S)
0 8	2 9	9 5	9 5	– 0 0 5	– 0 0	0 9	2 8	9 5	Plant Hatch, Unit 2	0 5 0 0 0 3 6 6
										0 5 0 0 0

OPERATING MODE (9): 1
POWER LEVEL (10): 1 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)	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)		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 (include area code)
Steven B. Tipps, Nuclear Safety and Compliance Manager, Hatch	AREA CODE 9 1 2 3 6 7 - 7 8 5 1

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
X	L R	B R K	0 0 0 0	No					

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 15 single-space typewritten lines) (16)

On 8/29/95 at 1930 EDT, Unit 1 was in the Run mode at a power level of 2436 CMWT (100 percent of rated thermal power) and Unit 2 was in the Run mode at a power level of 2175 CMWT (89.3 percent of rated thermal power). At that time, plant operators received indication that the nonessential loads on the "D" Unit 1 600-volt bus had tripped. Equipment deenergized included the Division II battery chargers for the Station Service Batteries (SSB). With the battery chargers deenergized, the associated batteries were declared inoperable. With the batteries declared inoperable, supported systems were also declared inoperable including the High Pressure Coolant Injection System (HPCI). At 2200 EDT, the feeder breaker to the motor-generator set of the Reactor Protection System power supply tripped, producing actuations of Engineered Safety Features including Group 2 and Group 5 Primary Containment Isolation System valves and all four trains of both units' Standby Gas Treatment Systems.

The cause of the nonessential load trip was an electrical ground in an elevator control circuit powered from the 600-volt bus. The ground is believed to have produced a spurious trip of the feeder breaker to the motor-generator set. Corrective actions for this event included isolating the electrical ground, repairing the grounded component, and moving the power source of the freight elevator. These actions are complete. Evaluations to identify similar circuits are in progress.

PLANT AND SYSTEM IDENTIFICATION

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

DESCRIPTION OF EVENT

On 8/29/95 at 1930 EDT, Unit 1 was in the Run mode at a power level of 2436 CMWT (100 percent of rated thermal power) and Unit 2 was in the Run mode at a power level of 2175 CMWT (89.3 percent of rated thermal power) in end-of-cycle coastdown. At that time, Control Room operators received annunciations indicating that nonessential loads from the Unit 1 'D' 600-volt bus (EIIS Code ED) had been deenergized. This condition is referred to as "nonessential load lockout." Equipment deenergized included the Unit 1 'B' Reactor Building Closed Cooling Water system pump (EIIS Code CC), the Unit 1 'B' Station Service Air Compressor (EIIS Code LF), and all three Division II chargers for the Unit 1 Division II Station Service Batteries (SSB, EIIS Code EJ). With the battery chargers inoperable, licensed personnel declared the Division II SSB inoperable also. Further, with the SSB declared inoperable, the operability of supported systems can not be assumed. Licensed personnel reviewed equipment powered from the SSB and identified the auxiliary oil pump for the High Pressure Coolant Injection (HPCI, EIIS Code BJ) system as being powered from the SSB, and thus considered it inoperable. Therefore, the HPCI system was declared inoperable at 2300 EDT. The 600-volt 'D' bus remained energized.

Immediately after the event, Operations personnel and electricians from the Maintenance department investigated and found that the relays for the nonessential load lockout were deenergized and that the fuses which feed the potential transformers supplying these relays were blown. The fuses were replaced at 2000 EDT, but immediately blew. Troubleshooting activities were continuing when, at 2200 EDT, the 600-volt feeder breaker to the Unit 1 'B' Reactor Protection System (EIIIS Code JE) power supply motor-generator set tripped. This breaker is also supplied from Unit 1 600-volt bus 'D'. The nonessential load lockout does not affect this breaker. When the breaker tripped, the motor-generator coasted down until its own output breakers tripped on underfrequency per design. The trip of the output breakers caused Group 2, Group 5, and outboard small-bore Group 1 Primary Containment Isolation System (PCIS, EIIS Code JM) valves to receive an automatic isolation signal. The Main Control Room Environmental Control System (MCRECS, EIIS Code VI) entered the pressurization mode; all four trains of both units' Standby Gas Treatment Systems (SGTS, EIIS Code BH) initiated; the Primary Containment Hydrogen and Oxygen Analyzers (EIIS Code IK) isolated; the Reactor Water Cleanup (RWCU, EIIS Code CE) system, the Fission Product Monitoring (FPM, EIIS Code IJ) system, and the operating Steam Packing Exhauster tripped. The operations staff placed the affected RPS bus on its alternate supply, and systems affected by this trip were returned to their normal configurations by 2245 EDT.

Troubleshooting activities were continuing when, at 2330 EDT, a plant equipment operator noticed a freight elevator ventilation fan motor over heating and the freight elevator cycling up and down. He reported this fact to the Shift Supervisor who, realizing the freight elevator was powered from the affected 600-volt bus, directed that the elevator be deenergized. When this action was completed, the fuses for the potential transformers were replaced at 0035 EDT and remained intact. With the fuses intact, the relays controlling the nonessential load lockout were energized. Licensed personnel reset the lockout, allowing restoration of power to the nonessential loads on the 600-volt bus. With these loads being powered, the battery chargers were energized, so the SSB and the affected systems were then declared operable.

CAUSE OF EVENT

These events were caused by an arcing ground on a freight elevator brake solenoid, which resulted in blown fuses in the 600-volt bus protection circuitry. The blown fuses resulted in a trip of the nonessential load lockout logic on the Unit 1 'D' 600-volt bus. It is also believed that the ground caused a subsequent trip of the Unit 1 'B' RPS motor-generator set feeder breaker.

The ground on the freight elevator component which tripped the lockout on the 600-volt bus occurred on the elevator's emergency brake solenoid. The freight elevator emergency brake solenoid is normally energized when the elevator is in operation. The solenoid power is supplied from the Unit 1 'D' 600-volt bus through an autotransformer and rectifier circuit. The autotransformer and rectifier circuit produces the necessary DC current for the elevator controls. When the ground occurred in the brake solenoid, sufficient DC current flowed through the potential transformer to drive it into saturation. The saturation produced an overcurrent condition high enough to activate the potential transformer's overcurrent fuses. When the fuses activated, the 600-volt bus undercurrent devices operated and initiated the nonessential load shed logic.

The ground is believed to have caused the trip of the RPS motor-generator set feeder breaker. The affected breaker is equipped with a microprocessor-based trip unit and is also located on the same bus affected by the ground. The ground apparently affected the trip unit such that its microprocessor actuated the breaker.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This event is reportable per 10 CFR 50.73 (a)(2)(v) because an event occurred in which a single train safety system, namely the HPCI system, was declared inoperable. This event is also reportable per 10 CFR 50.73 (a)(2)(iv) because an event occurred in which Engineered Safety Features experienced unplanned, automatic actuations as described above.

The HPCI system consists of a steam turbine-driven pump and the necessary piping and valves to transfer water from the suppression pool to the reactor vessel. The system is designed to inject water to the reactor vessel over a wide range of reactor pressures from 150 psig through full rated pressure. The HPCI system starts and injects automatically whenever reactor water level decrease or high drywell pressure indicates the possibility of an abnormal loss of coolant inventory. The HPCI system is especially designed to replace lost reactor coolant inventory in cases where a small line break occurs which does not result in full depressurization of the reactor vessel. In order for HPCI to function, a DC-powered auxiliary oil pump must operate to provide hydraulic control pressure for the system. Based on the loss of the battery chargers, the battery which supplies the auxiliary oil pump was declared inoperable, so the HPCI system was declared inoperable also. However, since the battery was still charged and capable of supplying power to the auxiliary oil pump, the HPCI system was capable of an automatic initiation and injection. In addition, the backup systems for the HPCI system (described below) were available during the event and could have injected water to provide core cooling had this been necessary.

The backup for the HPCI system is the Automatic Depressurization System (ADS) together with two low pressure injection systems, the Residual Heat Removal/Low Pressure Coolant Injection (RHR/LPCI, EIIS Code BO) system and the Core Spray (CS, EIIS Code BM) system. The CS system is composed of two independent, redundant, 100 percent capacity subsystems. Each subsystem consists of a motor driven pump, its own dedicated spray sparger located above the core, and piping and valves to transfer water from the suppression pool to the sparger. Upon receipt of an initiation signal, the CS pumps in both subsystems start. Once ADS has reduced reactor pressure sufficiently, CS system flow begins. LPCI is an independent operating mode of the RHR system. There are two independent, redundant, 100 percent capacity LPCI subsystems, each consisting of two motor driven pumps and piping and valves to transfer water from the suppression pool to the reactor through the Reactor Recirculation (EIIS Code AD) System. Upon receipt of an initiation signal, all four LPCI pumps automatically start. RHR system valves in the LPCI flow path are automatically positioned to ensure the proper flow path for water from the suppression pool to inject into the recirculation loops. Once ADS has reduced reactor pressure sufficiently, the LPCI flow to the reactor through the recirculation loop begins. The divisionally separated initiation logic systems for RHR/LPCI and CS incorporate "crossover" circuitry allowing each division to trigger an initiation of the other division. With this design, any one operable division of logic can produce a full actuation in both divisions of all the pumps and valves necessary for injection to the reactor vessel.

In this event, one loop of RHR/LPCI was affected by the loss of the battery chargers in that the injection valve (which is powered by the battery via an inverter) would eventually have been inoperable as the SSB discharged. No loss of safety function occurred, however, because the other division was unaffected, and would have been available should an accident have occurred during the time the affected train was inoperable. In addition, the battery which powers the inverter for the LPCI injection valves was still charged and functioning even though the charger for the battery was deenergized. During this time, the plant was complying with the required actions for Technical Specifications section 3.8.4, Condition D, assuring that the plant would have reached mode 3 before the battery was discharged to the point that the affected LPCI valve would have been actually incapable of opening. The CS system was also unaffected by this event and would have been available had it been needed.

The RPS power supply system is designed to supply stable 120-volt AC power to a variety of plant instrumentation systems including the Process Radiation Monitoring System, the Neutron Monitoring System, the Reactor Protection System, the Primary Containment Isolation System, and the Offgas Radiation Monitoring System. A high degree of power stability is achieved by using two motor-generator sets to condition the power supplied by the RPS power supply system. The electrical output of each motor-generator set energizes one of two RPS busses. In this event, the feeder breaker to the motor-generator set tripped apparently due to the ground on the 600-volt bus. When the breaker tripped, the motor-generator set coasted down until its own output breakers tripped on underfrequency. Upon loss of power or control signal, systems powered by the RPS deenergize to their "safe" configuration (i.e., they initiate their emergency or accident functions). All systems affected by this event responded per design for a power interruption and licensed personnel verified this per procedure immediately after the event occurred. Had a design basis accident occurred in conjunction with the trip of the RPS power supply, plant systems powered from RPS would already have been in their emergency configurations and no further automatic actuations of these systems would have been required to mitigate the accident.

Based on this analysis, it is concluded that this event had no adverse impact on nuclear safety. This analysis is applicable to all power levels.

CORRECTIVE ACTIONS

1. The ground in the freight elevator component was isolated and all affected loads were reenergized using the normal power supply. This action has been completed.

2. The grounded brake solenoid in the freight elevator has been repaired and the elevator returned to service. This action has been completed.

3. The power supply for the freight elevator has been temporarily moved to a different electrical bus which does not supply essential, safety-related loads. This removal from the class 1E source has been completed and the power supply will remain on a non-safety related source until Corrective Action #4 described below has been completed.

4. The Architect/Engineer for Plant Hatch has been requested to evaluate a permanent means of supplying power to the freight elevator through an electrical bus which does not supply essential, safety-related loads. This evaluation will be completed by 12/15/95. Additional actions based on this evaluation will be taken, as necessary.

5. The Architect/Engineer for Plant Hatch has been requested to identify similar DC circuits and evaluate a means for providing isolation between 600-volt ungrounded systems and any DC circuits which they may supply. Such isolation will reduce the likelihood of the type of unwanted trips experienced in this event. This evaluation will be completed by 12/15/95.

6. The 600-volt feeder breaker for the RPS motor-generator set will be modified to incorporate a noise filter. This will reduce the likelihood of further spurious trips of this breaker. This action will be completed by 12/1/95.

NRC FORM 366A (5-92)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, 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 YEAR	REVISION NUMBER	PAGE (3)
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TEXT (If more space is required, use additional copies of NRC Form 366A)(17)

ADDITIONAL INFORMATION

1. No systems other than those already mentioned in this report were affected by this event.

2. Failed Component Information:

Master Parts List Number: 1U11-E004
Serial Number: 500-63
Manufacturer Code: 0000
Reportable to NPRDS: No
EIIS Component Code: BRK

Manufacturer: F. S. Payne Co.
Type: Brake Solenoid
EIIS System Code: LR
Root Cause Code: X

3. Previous Similar Events: There have been no events reported in the past two years in which a ground on an electrical system produced the type of plant responses experienced in this event.