Georgia Power Company
333 Piedmont Avenue
Atlanta, Georgia 30308
Telephone 404 526-3195

Mailing Address:
40 Inverness Center Parkway
Post Office Box 1295
Birmingham, Alabama 35201
Telephone 205 868-5581

the southern electric system

W. G. Hairston, III
Senior Vice President
Nuclear Operations

HL-1476
001195

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

PLANT HATCH - UNIT 1
NRC DOCKETS 50-321
OPERATING LICENSES DPR-57
LICENSEE EVENT REPORT
REMOTE TRANSMISSION LINE FAILURE COUPLED WITH
SWITCHYARD BREAKER FAILURE CAUSES REACTOR SCRAM

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 the unanticipated actuation of some Engineered Safety Features (ESFs) and a condition that could have prevented an ESF from fully performing its safety function. This event occurred at Plant Hatch - Unit 1.

Sincerely,

W. G. Hairston, III

W. G. Hairston, III

SWR/ct

Enclosure: LER 50-321/1991-001

c: (See next page.)

IE22

Enclosure:

c: Georgia Power Company
Mr. H. L. Sumner, General Manager - Nuclear Plant
Mr. J. D. Heidt, Manager Engineering and Licensing - Hatch
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

001195

NRC Form 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

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

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
PLANT HATCH, UNIT 1	05000321	1 OF 8

TITLE (4)
REMOTE TRANSMISSION LINE FAILURE COUPLED WITH SWITCHYARD BREAKER FAILURE CAUSES 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)
									PLANT HATCH, UNIT 2	05000366
01	18	91	91	001	00	02	11	91		05000

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR (11)

OPERATING MODE (9): 1

POWER LEVEL: 100

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)
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	MANUFAC-TURER	REPORT TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORT TO NPRDS	
X	BJ	SC	W290	YES		X	BN	FU	B569	NO	
X	BD	DMP	B237	NO		X	FK	BKR	N/A	NO	

SUPPLEMENTAL REPORT EXPECTED (14)

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

ABSTRACT (16)

On 01/18/91, at approximately 1500 CST, Unit 1 was in the Run mode at an approximate power level of 2436 CMWT (approximately 100% rated thermal power). At that time, a remote power transmission line failed at its attachment point to the supporting electrical tower. In addition, a breaker in the Plant Hatch 230 kV switchyard failed to fully open in response to the sensed electrical fault conditions resulting in the trip of the Unit 1 main transformer auxiliary lockout relay. This resulted in main generator and main turbine trips, which, in turn, resulted in a reactor scram. This also resulted in a loss of power to the Unit 1 nonessential electrical busses and the prevention of the normal automatic transfer to the alternate supply due to the transfer logic. Power was manually restored to nonessential loads within approximately 2 to 3 minutes allowing for the initiation of operator actions to restore feedwater flow. In the interim, to recover reactor water level, the High Pressure Coolant Injection system (HPCI) automatically initiated. Although it exhibited erratic behavior in the automatic mode, it was successfully controlled manually and water level was restored and maintained with HPCI until feedwater flow was restored. Reactor pressure was controlled by the turbine bypass valves.

The root causes of the scram were the failure of the high tension power line and the failure of the main breaker to fully open. The root cause of the erratic HPCI operation was component failure in the HPCI speed controller.

Corrective actions included repairing the failed power line, repairing the failed breaker, and replacing the failed HPCI speed controller.

TEXT

PLANT AND SYSTEM IDENTIFICATION

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

SUMMARY OF EVENT

On 01/18/91, at approximately 1500 CST, Unit 1 was in the Run mode at an approximate power level of 2436 CMWT (approximately 100% rated thermal power). At that time, a power transmission line failed at its attachment point to the supporting electrical tower, which was located approximately 21.4 miles from Plant Hatch. In addition, a breaker in the Plant Hatch 230 kV switchyard failed to fully open in response to the sensed electrical fault conditions resulting in the trip of the Unit 1 main transformer auxiliary lockout relay. This resulted in main generator and main turbine trips, which, in turn, resulted in a reactor scram. This also resulted in a loss of power to the Unit 1 nonessential electrical busses and the prevention of the normal automatic transfer to the alternate supply due to the transfer logic. Power was manually restored to nonessential loads within approximately 2 to 3 minutes allowing for the initiation of operator actions to restore feedwater flow. In the interim, to recover reactor water level, the High Pressure Coolant Injection system (HPCI, EIIS Code BJ) automatically initiated on low level. Although it exhibited erratic behavior in the automatic mode, it was successfully controlled manually and water level was restored and maintained with HPCI until feedwater flow was restored. Reactor pressure was controlled by the turbine bypass valves (BPV, EIIS Code JI).

The root causes of the scram were the failure of the high tension power line and the failure of the main breaker to fully open. The root cause of the erratic HPCI operation was component failure in the HPCI speed controller.

Corrective actions included repairing the failed power line, repairing the failed breaker, and replacing the failed HPCI speed controller.

DESCRIPTION OF THE EVENT

On 01/18/91, at approximately 1500 CST, Unit 1 was in normal full power operation when the phase 1 portion of a 230 kV transmission line failed at its attachment point to the supporting electrical tower, located approximately 21.4 miles from the plant. As the phase 1 line fell away from the break location, it briefly contacted phase 2, creating a phase 1-2 fault. This fault was sensed and resulted in a trip signal being sent to power circuit breakers (PCB, EIIS Code FK) 490 and 500 in the Plant Hatch 230 kV switchyard. Phase 2 of PCB 500 did not successfully open at this time due to a failed current limiting resistor in the breaker control circuit. However, as phase 1 opened, the phase 1-2 fault cleared since the current loop between phases 1 and 2 had been interrupted. At this point in the transient, the electrical output of Plant Hatch Unit 1 was flowing through other circuits in the switchyard, and for the next few milliseconds, no other electrical fault existed.

TEXT

Continued movement of the failed phase 1 line resulted in it again contacting phase 2 and this time contacting the electrical tower as well. This provided a conducting path between phase 2 and the electrical tower, creating a phase 2 to ground fault. This fault current was supplied through the still-closed pole of phase 2 in PCB 500. The resumption of fault current through this breaker resulted in the breaker failure relay being tripped. The breaker failure relay tripped a lockout relay in the Unit 1 main transformer control circuitry which, in turn, initiated a main generator and main turbine trip. This turbine trip occurred at greater than 30% reactor power, resulting in a reactor scram per design. The main turbine bypass valves opened, as designed, to mitigate the reactor pressure transient, limiting reactor pressure during the transient to a maximum of approximately 1095 psig. No safety relief valves (SRV, EIIS Code JE) opened and post-event investigation confirmed none were required to open.

With the main transformer disconnected from the grid, nonessential site electrical loads were automatically disconnected since they are normally supplied through the main transformer. The nonessential loads are designed to automatically transfer to their alternate supply. However, in order for this transfer to occur, the transfer logic circuit must sense all phases of both PCBs 500 and 510 open. Because one phase of PCB 500 did not open and indication for the PCB was lost (explained later in this report), automatic transfer to the alternate supply was prevented and Unit 1 nonessential loads remained deenergized. Among these loads were the condensate pumps (EIIS Code SD), condensate booster pumps (EIIS Code SJ), main circulating water pumps (EIIS Code SD), and reactor recirculation system motor-generator sets (EIIS Code AD).

Without the condensate booster pumps, the reactor feedpumps tripped on low suction pressure. Lacking normal reactor feedwater flow, reactor water level decreased to approximately 12 inches above instrument zero by approximately 15:01 CST, initiating a Primary Containment Isolation System (PCIS, EIIS Code JM) Group II isolation and another scram due to low reactor water level. All required Group II valves closed as designed. At approximately 15:03 CST, reactor water level reached 35 inches below instrument zero, resulting in an automatic initiation of the HPCI and the Reactor Core Isolation Cooling systems (RCIC, EIIS Code BN), as well as PCIS Group V and secondary containment isolations. The PCIS Group V isolation occurred as designed. Both Unit 1 and Unit 2 Standby Gas Treatment Systems (SGTS, EIIS Code BH) started and functioned as designed. Upon receipt of the auto start signal, the HPCI system demonstrated erratic operation resulting in more than one HPCI turbine overspeed trip. The control room operators took manual control of HPCI. The RCIC system functioned normally. Thus, with the HPCI and RCIC systems injecting, reactor water level recovered. The minimum water level observed during the transient was approximately 38 inches below instrument zero, which is approximately 126.5 inches above the top of active fuel.

TEXT

A post-event review team observed that a secondary containment damper, 1T41-F043B, failed to close when the secondary containment isolation signal was received. However, the inoperable damper did not prevent successful secondary containment isolation because the redundant damper in that same duct did close.

At approximately 2 to 3 minutes after the initial scram, power was manually restored to nonessential site loads. At this point, activities were initiated to restore normal feedwater flow.

Between 15:05 CST and 15:10 CST the control room received a report of a fire in the switchyard. The fire brigade responded, but found the fire had already been extinguished by a member of the switchyard maintenance crew using a portable carbon dioxide fire extinguisher. The source of the fire was a damaged current limiting resistor in the control circuitry of the failed PCB 500. The damage to the control circuitry had already caused the loss of PCB indication alluded to earlier in this report.

The licensed reactor operator assigned to control reactor water level reduced the flowrate and then tripped the HPIC and RCIC systems to stabilize reactor water level at approximately 43 inches above instrument zero. This occurred at approximately 15:23 CST. Although not recognized at the time, position indication was lost on the RCIC inboard injection valve, 1E51-F013, as it began to close as designed on a RCIC trip signal. As decay heat caused reactor pressure to increase and the bypass valves opened to control reactor pressure, reactor water level again began to decrease. At approximately 15:35 CST, with reactor water level at approximately 15 inches above instrument zero and decreasing slowly, RCIC was restarted. It was at this time that the licensed operator noticed that position indication had been lost on 1E51-F013. The operating crew concluded that the valve was only partially open since it appeared to them that reactor water level was not being restored as quickly as expected. The operators decided to secure RCIC and investigate the valve indication problem. A plant equipment operator was dispatched to the valve breaker, but the breaker would not reset.

At approximately 15:40 CST, reactor water level reached approximately 12.5 inches above instrument zero, and another scram signal was received due to low reactor water level. Plant operators then started HPCI in the manual control mode and recovered reactor vessel water level. HPCI would have been restarted earlier to recover level, but operations personnel believed the restoration of feedwater was imminent. By approximately 15:47 CST, a reactor feed pump was returned to service, and water level was controlled thereafter using the normal reactor level control system.

TEXT

CAUSE OF THE EVENT

The direct cause of the first scram was a turbine trip which occurred above 30% rated thermal power. The root causes of the first scram were the failed remote power transmission line and the failed current limiting resistor in PCB 500. The failed current limiting resistor prevented the trip signal from reaching phase 2 of PCB 500 which caused the redundant breaker, PCB 510, to open to clear the fault. The opening of PCB 510 resulted in a main turbine trip, which led to the scram.

The cause of the failure of the HPCI system to properly inject in the automatic mode was a failed speed controller. Post-event maintenance attempted to recalibrate the controller, but the controller would not retain its recalibration. However, it should be noted that the failure mode of the speed controller, due to the configuration of the control logic, did not preclude successful control of HPCI in the manual mode.

The cause of the lost valve indication on 1E51-F013 was a blown fuse in the valve actuator control power circuit. This interrupted control power to the valve, preventing the valve from being moved. The fuse failure was investigated by the architect/engineer and was determined to have been a random failure.

The cause of the failure of damper 1T41-F043B was water-induced corrosion on the pneumatic actuator. It appears that due to the location of this damper on the outside of the reactor building, the damper was exposed to environmental conditions which fostered the condensation and accumulation of moisture and the resultant corrosion over time. It should be noted that the air quality, specifically dewpoint, particulate content, and oil content, has been sampled as part of Georgia Power Company's response to Generic Letter 88-14 regarding instrument air supply system problems affecting safety related equipment. The results of the sampling met the requirements ANSI/ISA S7.3-1975, Reaffirmed 1981. As part of the ongoing Plant Hatch program for maintaining proper instrument air quality, air quality is periodically sampled and evaluated against the same standard.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This event is reportable per 10 CFR 50.73(a)(2)(iv) because an unplanned actuation of the Reactor Protection System (RPS, EIIS Code JC) and Engineered Safety Features (ESF) occurred. Specifically, tripping of a main transformer lockout relay resulted in a turbine trip above 30% rated thermal power, which in turn resulted in a reactor scram. This event is also reportable per 10 CFR 50.73(a)(2)(v) because an event occurred which could have prevented the fulfillment of the safety function of a system which is intended to mitigate the consequences of an accident. Specifically, a failed HPCI speed controller caused the HPCI system to trip repeatedly due to overspeed, with the result that operators were required to take manual control of HPCI in order to restore reactor water level.

TEXT

The Reactor Protection System (RPS, EIIS Code JC) automatically initiates a reactor scram to ensure the radioactive materials barriers, such as fuel cladding and pressure system boundary, are maintained, and to mitigate the consequences of transients and accidents. Closure of the turbine stop valves can result in an addition of positive reactivity to the core as the reactor pressure rise collapses steam voids. Turbine stop valve closure initiates a scram prior to the Neutron Monitoring System or Reactor High Pressure; however, it is required to provide a satisfactory margin to core thermal-hydraulic limits. The high-pressure scram in conjunction with the pressure relief system is adequate to preclude overpressurizing the pressure system boundary; however, the turbine stop valve closure scram provides additional margin. In the event described in this report, the RPS actuated per design. Reactor pressure was controlled by the turbine bypass valves alone. Consequently, reactor pressure was maintained well below the vessel design pressure.

The HPCI system is provided to assure the reactor is adequately cooled to limit fuel-clad temperature in the event of a small break in the nuclear boiler system causing a loss of coolant which does not result in rapid depressurization of the reactor vessel. The Automatic Depressurization System (ADS, EIIS Code JE) is a backup for the HPCI system. Upon ADS initiation, the reactor vessel is depressurized to a point where either the Low Pressure Coolant Injection system (LPCI, EIIS Code BO) or the Core Spray System (CSS, EIIS Code BM) can operate to maintain adequate core cooling. In this event, due to a controller malfunction in the HPCI system, HPCI tripped more than once on turbine overspeed and had to be manually controlled in order to complete an injection into the reactor pressure vessel. During this event, the ADS, LPCI system, and CSS remained operable. Based upon the Unit 1 Final Safety Analysis Report (FSAR), either loop of the CSS or the LPCI system can supply sufficient cooling to the reactor for any rupture of the nuclear system boundary up to and including the Design Basis Accident (DBA).

The function of the secondary containment is to limit ground level releases of airborne radioactive materials, and to provide a means for the controlled, elevated release of the building atmosphere so that off-site doses from a design basis fuel handling or loss of coolant accident (LOCA) will be below the limits stated in 10 CFR 100. The secondary containment system consists of three subsystems: the reactor building, the Standby Gas Treatment System, and the main stack. In this event, secondary containment damper 1T41-F043B failed to close upon receipt of an isolation signal. However, as previously noted, the redundant damper in the same duct closed so that secondary containment integrity was successfully established.

Based on the above analysis, it is concluded that this event had no adverse impact on nuclear safety. Since the scram occurred from full power operation, it is concluded that the event would not have been more severe had it occurred under other operating conditions.

TEXT

CORRECTIVE ACTIONS

Corrective actions for this event included:

1. Repairing the Bullen 230kV transmission line. This action is complete.

2. Repairing PCB 500. This action is complete.

3. Replacing the HPCI speed controller and performing the HPCI operability test per 34SV-E41-002-1S, "HPCI Pump Operability." This action is complete.

4. Replacing the failed fuse in the control power circuit for 1E51-F013. This action is complete.

5. Repairing the failed damper actuator for 1T41-F043B. This action is complete.

6. One additional secondary containment isolation damper actuator located outside the reactor building will be inspected during the 1991 Unit 1 outage to determine if moisture-induced corrosion is evident. The remaining similar dampers will be inspected if corrosion is found. This action will be complete by 12/17/91.

ADDITIONAL INFORMATION

1. Other Systems Affected: No systems other than those described in this report were affected by the event.

2. Previous Similar Events: No events were identified in which failures in the switchyard led to reactor trips in the past two years. One event during this time frame was identified in which the HPCI system received a valid automatic initiation signal but failed to inject into the reactor vessel. This event was described in LER 50-366/1991-001, Revision 1. Corrective action for that event (related to the HPCI system failure) included replacing thermal overload relays for the HPCI injection valve motor starter. That corrective action would not have prevented this event because the causes of the injection system failures were different.

3. Failed Components Identification:

   A. Master Parts List Number: 1E41-R764-1
      Manufacturer: Woodward Governor Company
      Model Number: EG-M
      Type: Electronic Governor Magnetic Pickup
      Manufacturer Code: W290
      EIIS System Code: BJ
      Reportable to NPRDS: Yes
      Root Cause Code: X
      EIIS Component Code: SC

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (5) YEAR	SEQ NUM	REV	PAGE (3)
PLANT HATCH, UNIT 1	05000321	91	001	00	8 OF 8

TEXT

B. Master Parts List Number: 1E51-F231
Manufacturer: Bussman
Model Number: KTN-R-6
Type: Fuse
Manufacturer Code: B569
EIIS System Code: BN
Reportable to NPRDS: No
Root Cause Code: X
EIIS Component Code: FU

C. Master Parts List Number: 1T41-F043B
Manufacturer: Bettis Corporation
Model Number: 522C-SR-72
Type: Air Operated Damper Actuator
Manufacturer Code: B237
EIIS System Code: BD
Reportable to NPRDS: No
Root Cause Code: X
EIIS Component Code: DMP

D. Master Parts List Number: None
Manufacturer: Unknown (unreadable on burned resistor)
Model Number: 132A1230-32
Type: Current Limiting Resistor
Manufacturer Code: Unknown
EIIS System Code: FK
Reportable to NPRDS: No
Root Cause Code: X
EIIS Component Code: BKR