Georgia Power Company
40 Inverness Center Parkway
Post Office Box 1295
Birmingham, Alabama 35201
Telephone 205 877-7279

J. T. Beckham, Jr.
Vice President - Nuclear
Hatch Project

Georgia Power
the southern electric system

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Docket No. 50-366 HL-4836

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

Edwin I. Hatch Nuclear Plant - Unit 2
Licensee Event Report
Personnel Error Results in Automatic Reactor Shutdown

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 shutdown.

Sincerely,

J. T. Beckham, Jr.

OCV/eb

Enclosure: LER 50-366/1995-001

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. B. L. Holbrook, Senior Resident Inspector - Hatch

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NRC FORM 366 (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)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Edwin I. Hatch Nuclear Plant - Unit 2	0 5 0 0 0 3 6 6	1 OF 6

TITLE (4)
Personnel Error Results In Automatic Reactor Shutdown

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 NAME	DOCKET NUMBER(S)
0 4	1 1	9 5	9 5	0 0 1	0 0	0 5	0 4	9 5	Plant Hatch - Unit 1	0 5 0 0 0 3 2 1
									FACILITY NAME	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)	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 & 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

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 4/11/95 at 2051 EDT, Unit 2 was in the Run mode at a power level of 2436 CMWT (100% rated thermal power). At that time, the reactor automatically shut down due to an automatic trip of the main turbine on high reactor water level. Both reactor recirculation and feedwater pumps also tripped. Reactor pressure increased resulting in the opening of all safety/relief valves (SRVs) and main turbine bypass valves. As pressure decreased, the SRVs closed and the bypass valves automatically controlled pressure. Water level decreased due to steam discharge from the SRVs and the bypass valves and loss of the feedwater pumps. A Group 2 primary containment isolation system (PCIS) signal was received and the Group 2 valves closed. Level decreased to 37 inches below instrument zero causing initiation of the high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems, receipt of a partial Group 5 PCIS isolation signal, start of the Unit 1 and 2 standby gas treatment systems, and isolation of the Unit 1 and 2 secondary containments. Level was recovered with the HPCI and RCIC systems. The cause of this event was personnel error. An operator placed a jumper in the wrong location, shorting the power supply for reactor water level transmitter 2C32-N004B and causing the signal from the transmitter to decrease to zero. The reactor water level control system responded by increasing the flow rates from the feedwater pumps, causing water level to increase. Corrective actions include disciplining the operator, temporarily relieving him of licensed duties, providing him remedial training, and requiring double verification of jumper locations.

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 4/11/95 at 2051 EDT, Unit 2 was in the Run mode at a power level of 2436 CMWT (100% rated thermal power). Operations personnel were returning the reactor water cleanup (EIIS Code CE) system to service. Specifically, a licensed operator was installing jumpers to bypass the high differential flow isolation signal per plant procedure 34SO-G31-003-2S, "Reactor Water Cleanup System." At that time, the reactor automatically shutdown on turbine stop valve closure at greater than 30% of rated thermal power. This was a direct result of an automatic main turbine shutdown on high reactor water level. The reactor feedwater pumps (EIIS Code SJ) also automatically tripped on high water level. Both reactor recirculation pumps (EIIS Code AD) automatically tripped on turbine stop valve closure at greater than 30% rated thermal power.

Reactor vessel pressure increased immediately upon the automatic trip of the main turbine, reaching a maximum value of 1091 psig and resulting in a redundant automatic reactor shutdown signal on high reactor vessel pressure. All 11 main steam line safety/relief valves opened on high pressure, arming the low low set logic system per design. The three main turbine bypass valves (EIIS Code SO) also opened on high pressure. As reactor vessel pressure decreased due to the open safety/relief and main turbine bypass valves, the safety/relief valves closed. The low low set logic system decreased reactor pressure to approximately 818 psig, was reset by operations personnel, and the bypass valves were then used to automatically control pressure between 900 psig and 935 psig.

Water level decreased due to steam discharge from the safety/relief and main turbine bypass valves and loss of the reactor feedwater pumps. Within 29 seconds of the initial automatic reactor shutdown, a Group 2 primary containment isolation system (PCIS, EIIS Code JM) isolation signal and another automatic reactor shutdown signal were received on low (level 3) reactor water level. The Group 2 primary containment isolation valves (EIIS Code JM) closed per design. Water level continued to decrease and, at approximately three minutes after the initial automatic reactor shutdown, reached its minimum value of 37 inches below instrument zero (121 inches above the top of the active fuel). The high pressure coolant injection (EIIS Code BJ) and reactor core isolation cooling (EIIS Code BN) systems automatically started and injected makeup water to the reactor vessel per design. Additionally, a partial Group 5 (reactor water cleanup system) PCIS isolation signal was received, all four trains of the Unit 1 and Unit 2 standby gas treatment (EIIS Code BH) systems automatically started, and the Unit 1 and Unit 2 secondary containment outboard isolation

dampers automatically closed per design on low low (level 2) reactor water level. No Group 5 primary containment isolation valves actually closed because they had been closed prior to the event.

Reactor water level was recovered automatically with the high pressure coolant injection and reactor core isolation cooling systems. Operations personnel then shut down the high pressure coolant injection system and used the reactor core isolation cooling and control rod drive (EIIS Code AA) systems to maintain normal water level.

Operations personnel restarted the "A" reactor recirculation pump approximately 16 minutes following its automatic trip. The "B" pump was restarted approximately 23 minutes following its automatic trip. Reactor vessel bottom head temperature decreased less than 100 degrees Fahrenheit during the time the recirculation pumps were not operating. No cooldown rate or reactor pressure-versus-temperature limits were exceeded during this event.

CAUSE OF EVENT

The cause of this event was personnel error. A licensed operator placed a jumper on the wrong terminal points when attempting to bypass the reactor water cleanup (EIIS Code CE) system high differential flow isolation signal per plant procedure 34SO-G31-003-2S, "Reactor Water Cleanup System." Bypassing the high differential flow isolation signal is a standard practice when returning the reactor water cleanup system to service and is allowed by Unit 2 Technical Specifications Table 3.3.2-1, item 3, footnote l. The terminal points on which the jumper was to be placed were identified correctly in procedure 34SO-G31-003-2S and the terminal strip and points were labeled properly and conspicuously.

The jumper was mistakenly placed across the line and neutral terminals of the power supply for reactor water level transmitter 2C32-N004B. This transmitter provides the water level signal to the reactor water level control (EIIS Code JB) system. The jumper shorted the power supply, causing its output signal to decrease to near zero per design. The loss of power supply output caused the level signal from water level transmitter 2C32-N004B to decrease to zero. The reactor water level control system responded to the resulting false low water level signal by increasing the flow rates from the reactor feedwater pumps in an attempt to increase the low reactor water level condition seen by the logic. This caused actual water level to increase until it reached the automatic trip setpoints for the main turbine and reactor feedwater pumps.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This report is required by 10 CFR 50.73(a)(2)(iv) because of unplanned actuations of the reactor protection system (EIIS Code JC) and other engineered safety feature (ESF) systems. Specifically, the reactor protection system actuated on turbine stop valve closure at greater than 30% rated thermal power when the main turbine automatically tripped on high reactor water level. The reactor recirculation pumps also automatically tripped on turbine stop valve closure at greater than 30% power. Subsequent actuations of the reactor protection system occurred on high reactor pressure and low reactor water level as a result of the pressure and level transient resulting from the trip of the main turbine and the reactor feedwater pumps. Groups 2 and 5 (partial) of the primary containment isolation system actuated on low and low low reactor water level, respectively. The Group 2 primary containment isolation valves closed as required; however, no Group 5 valves isolated because they had been closed prior to the event. The high pressure coolant injection and reactor core isolation cooling systems automatically started and injected makeup water to the reactor vessel per design. All four trains of the Unit 1 and Unit 2 standby gas treatment systems automatically started and the Unit 1 and Unit 2 secondary containment outboard isolation dampers automatically closed per design on low low reactor water level.

The main turbine and reactor feedwater pumps automatically tripped on high reactor water level to prevent water induction from reactor water carryover into the main steam lines and subsequent turbine damage. The main turbine stop and control valves close rapidly on a main turbine automatic trip signal. Rapid closure of these valves causes a sudden reduction in steam flow from the reactor vessel which results in an increase in reactor vessel pressure. In order to limit the pressure transient on the vessel, and the increase in neutron and heat fluxes caused by the increase in reactor vessel pressure, main turbine stop valve closure at greater than 30% rated thermal power results in an automatic reactor shutdown. Rapid power reduction is further ensured by an automatic trip of the reactor recirculation pumps on main turbine stop valve closure at greater than 30% power.

In this event, all systems responded as designed and analyzed. An actual high reactor water level condition resulted in an automatic trip of the main turbine per design. This, in turn, caused an automatic shutdown of the reactor, and a trip of the recirculation pumps. The safety/relief and main turbine bypass valves opened to limit the pressure transient and to reduce pressure to less than its normal operating value. The maximum pressure during this event was 1091 psig. This is well within the American Society of Mechanical Engineers' Code limit for the reactor pressure vessel. The low low set logic system and the bypass valves automatically controlled pressure at less than 1000 psig following the initial transient.

Other ESF systems, including Groups 2 and 5 (partial) of the primary containment isolation system, the high pressure coolant injection and reactor core isolation cooling systems, the Unit 1 and Unit 2 standby gas treatment systems, and the Unit 1 and Unit 2 secondary containment isolation systems (outboard dampers), actuated on low or low low reactor water level per their design. Reactor water level was recovered automatically with the high pressure coolant injection and reactor core isolation cooling systems; automatic operation of these systems limited the minimum water level to 121 inches above the top of the active fuel. The reactor core isolation cooling and control rod drive (EIIS Code AA) systems were then used to maintain water level greater than 172 inches above the top of the active fuel.

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

CORRECTIVE ACTIONS

The operator was disciplined under Georgia Power Company's Positive Discipline Program. Additionally, he was temporarily relieved of licensed duties and given remedial training on the identification of terminal points and the placement of jumpers.

Operations management instituted a policy under which operations personnel are required to perform double verification of the location of terminal points prior to placing jumpers. This policy may be rescinded at the discretion of management.

ADDITIONAL INFORMATION

No systems other than those mentioned in this report were involved in this event.

No failed components caused or resulted from this event.

Previous similar events in the last two years in which personnel error caused unplanned ESF system actuations were reported in the following Licensee Event Reports:

50-321/1993-004, dated 05/14/93
50-321/1993-007, dated 05/21/93
50-321/1993-009, dated 06/10/93
50-321/1994-002, dated 04/19/94
50-321/1994-011, dated 10/18/94
50-321/1994-012, dated 11/14/94
50-366/1993-005, dated 06/10/93
50-366/1994-004, dated 05/17/94
50-366/1994-005, dated 05/23/94

Corrective actions for these events included training and disciplinary actions. These actions are intended to heighten attention to task performance in involved personnel as well as the general plant population. However, by their nature, these actions cannot completely eliminate the potential for task performance errors in any particular individual.