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Georgia Power
the southern electric system

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Docket No. 50-366

HL-4599

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

Edwin I. Hatch Nuclear Plant - Unit 2
Licensee Event Report
False Low Reactor Water Level Signal Results in
Automatic Actuation of Engineered Safety Features

Gentlemen:

In accordance with the provisions of 10 CFR 50.73(a)(2)(iv), Georgia Power Company is submitting the enclosed Licensee Event Report (LER) concerning a false low reactor water level signal which occurred during the Spring 1994 Unit 2 refueling outage. The false signal resulted in an automatic actuation of engineered safety features.

Sincerely,

J. T. Beckham, Jr.

JKB/cr

Enclosure: LER 2-94-005

cc: *Georgia Power Company*
Mr. H. L. Sumner, 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. L. D. Wert, Senior Resident Inspector - Hatch

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 5

TITLE (4)
False Low Reactor Water Level Signal Results in Automatic Actuation of Engineered Safety Features

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	2 3	9 4	9 4	0 0 5	0 0	0 5	2 3	9 4		0 5 0 0 0
									FACILITY NAME	0 5 0 0 0

OPERATING MODE (9): 4

POWER LEVEL (10): 0 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 and Compliance Manager	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 04/23/94 at 1550 EDT, Unit 2 was in the Cold Shutdown mode near the end of a refueling outage. At that time, licensed Control Room operators observed that a full Reactor Protection System (RPS) actuation and a partial Group 2 Primary Containment Isolation System (PCIS) actuation had occurred. All control rods were already fully inserted; therefore, no control rod movement occurred. Licensed personnel reset the RPS signal within approximately three minutes and subsequently returned valves affected by the Group 2 PCIS actuation to their required lineups.

The cause of this event was an inadvertent action on the part of an Instrument and Control technician. The technician was performing a procedure to recalibrate a reactor water level instrument for cold shutdown conditions. During the calibration, he inadvertently bumped the valve handle on an instrument isolation valve which partially opened the valve and depressurized the associated instrument variable leg. This caused a false low reactor water level signal affecting both channels of RPS logic and one channel of PCIS logic.

Corrective actions for this event included resetting the RPS signal and returning affected systems to the lineups required for the plant condition. In addition, the responsible technician has been made aware of the consequences of his inappropriate action.

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 04/23/94 at 1550 EDT, Unit 2 was in the Cold Shutdown mode near the end of a refueling outage with all control rods (EIIS Code AA) fully inserted into the core. At that time, Instrument and Control (I&C) technicians were performing special purpose procedure 17SP-030194-OO-1-2S, "Reactor Water Level Cold Reference Leg Keepfill System Functional Test." This procedure requires reactor water level instrument 2B21-N093A to be calibrated under the temperature conditions which exist during Cold Shutdown. Part of the calibration process involves isolating the instrument from its variable and reference legs and opening its equalizing valve to remove air bubbles from the instrument. During the calibration, an I&C technician accidentally bumped the valve handle on an instrument isolation valve on the variable leg of reactor water level instrument 2B21-N093A which partially opened the valve. This momentarily depressurized the instrument variable leg and produced a low water level trip on other level instruments sharing the same variable leg. The other level instruments connected to that variable leg supply trip signals to the Reactor Protection System (RPS, EIIS Code JC) and the Primary Containment Isolation System (PCIS, EIIS Code JM). The logic combinations are such that the instruments on this variable leg can provide a full RPS actuation and a partial Group 2 PCIS actuation involving only outboard valves. Because of the plant configuration existing at the time, only two Group 2 PCIS valves closed, the outboard drywell floor drain and equipment drain sump isolation valves, 2G11-F004 and 2G11-F020. The I&C technician immediately realized the situation and quickly reclosed the isolation valve. No control rod movement occurred as a result of this event since all control rods were fully inserted prior to the event.

CAUSE OF EVENT

The cause of this event was an inadvertent action on the part of an I&C technician. While performing an instrument calibration, he accidentally bumped the valve handle on an instrument isolation valve on the variable leg of reactor water level instrument 2B21-N093A which partially opened the valve. This momentarily depressurized other level instruments on the shared variable instrument sensing leg, producing low reactor water level signals and resulting in the actuations described above.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This event is reportable per 10 CFR 50.73 (a)(2)(iv) because unplanned actuations of Engineered Safety Features (ESF) occurred. Specifically, a full RPS actuation occurred and two Group 2 PCIS valves automatically closed in response to a sensed low water level condition which resulted from momentarily depressurizing a reactor water level instrument sensing line.

The Reactor Protection System is designed to protect the integrity of the fuel cladding and nuclear process barriers by initiating a reactor shutdown when certain process conditions are sensed such as high reactor power, high reactor pressure, low reactor water level, turbine trip, etc. The RPS trip on low reactor water level (Level 3) is sensed by four level instruments designated 2B21-N680A, B, C, and D. Instruments 2B21-N680C and D share a common variable sensing line with instrument 2B21-N093A, the instrument whose isolation valve handle was accidentally bumped and partially opened by the I&C technician. Instruments 2B21-N680C and D supply trip signals to opposite channels of RPS trip logic. Therefore, when the variable inputs of these instruments were depressurized, a low reactor water level trip signal entered both channels of RPS logic, resulting in a full RPS actuation. Consistent with the conditions for being in Mode 4, all control rods were fully inserted into the core before the RPS actuation occurred. Consequently, no control rods moved, nor were any required to move.

The Primary Containment Isolation System provides automatic isolation capability of Primary Containment penetrations to preclude release of radioactive material in the event of an accident. Group 2 PCIS valves are generally located in lines which penetrate the Primary Containment but which do not communicate directly with the reactor coolant system. Group 2 PCIS actuations occur when sensed process conditions indicate the possibility of a leak in the reactor coolant system. The same reactor water level instruments which provide trip signals to the RPS also provide trip signals to the PCIS logic. The two particular instruments which share the variable sensing line with 2B21-N093A (i.e., 2B21-N680C and D) supply PCIS trip signals to outboard Group 2 PCIS logic. Thus the false low reactor water level signal on these two instruments resulted in an isolation signal being sent to outboard Group 2 PCIS valves. Because of the plant configuration which existed at the time, only two Group 2 PCIS valves closed. They were 2G11-F004 and 2G11-F020, the outboard drywell floor drain and equipment drain sump isolation valves. All other Group 2 PCIS valves which would have been affected were either already closed, were properly removed from service, or else had their isolation logic temporarily defeated at the time of the actuation because of an in progress Logic System Functional Test (LSFT).

All affected ESF systems responded per design given the signal which was introduced when the variable sensing line was depressurized. No unexpected actuations occurred, and all actuations which did not occur as expected were accounted for by procedurally controlled activities in progress at the time of the event.

Had this event occurred with the "B" Residual Heat Removal (RHR, EIIS Code BO) system operating in the Shutdown Cooling (SDC) mode, SDC injection valve 2E11-F015B (which is a PCIS valve affected by Group 2 PCIS logic) would have received an automatic closure signal. However, procedure 34AB-C71-001-2S, "Scram Procedure," requires licensed personnel to confirm all PCIS valve actuations occurred as designed. In addition, this procedure tells the user that the SDC injection valve will close if a scram occurs on low reactor water level while in the SDC mode of RHR. Therefore, administrative controls currently in place would have resulted in licensed operators quickly detecting closure of the SDC injection valve and implementing corrective action per 34AB-E11-001-1S, "Loss Of Shutdown Cooling."

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

CORRECTIVE ACTIONS

Corrective actions for this event included the following:

1. Licensed personnel reset the RPS trip signal and returned the affected valves to their required lineups.

2. This event has been discussed with the responsible I&C technician, and he is aware of the consequences of his inadvertent action.

ADDITIONAL INFORMATION

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

2. Failed Equipment Information: No failed equipment contributed to or resulted from this event.

3. Previous Similar Events: Events reported in the past two years in which inappropriate personnel actions resulted in unplanned ESF actuations are described in the following LERs:

   50-321/1992-009, dated 04/23/92
   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 03/25/94
   50-366/1992-009, dated 07/24/92
   50-366/1992-014, dated 09/15/92
   50-366/1992-021, dated 11/10/92
   50-366/1992-023, dated 12/14/92
   50-366/1992-026, dated 12/21/92
   50-366/1993-005, dated 06/10/93
   50-366/1994-004, dated 05/17/94

   Corrective actions for these events included replacing fuses, disciplining personnel, issuing a plantwide memorandum on prompt reporting of unexpected equipment responses, revising procedures, training personnel on issues pertinent to specific events (e.g., jumper placement, recognizing relay status), discussing events with personnel in shift meetings, revising plant drawings, and changing alarm setpoints. These actions would not have prevented this event because the situations and personnel involved were unique to each event. Training and disciplinary 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.