Georgia Powe. 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

HL-1527
001338

March 18, 1991

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

PLANT HATCH - UNITS 1
NRC DOCKETS 50-321
OPERATING LICENSES DPR-57
LICENSEE EVENT REPORT
REACTOR WATER LEVEL INSTRUMENT PERTURBATION
RESULTS IN REACTOR SCRAM WHILE IN COLD 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 reactor water level instrument perturbation which resulted in a reactor scram while in cold shutdown. This event occurred at Plant Hatch - Unit 1.

Sincerely,

J. T. Beckham, Jr.

J. T. Beckham, Jr.

OCV/cr

Enclosure: LER 50-321/1991-005

c: (See next page.)

IE22
1/1

Georgia Power

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

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

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

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)
REACTOR WATER LEVEL INSTRUMENT PERTURBATION RESULTS IN REACTOR SCRAM WHILE IN COLD SHUTDOWN

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 5 0 0 0
0 2	1 7	9 1	9 1	0 0 5	0 0	0 3	1 8	9 1		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 (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)
[ ]	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

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

ABSTRACT (16)

On 2/17/91, at approximately 1840 CST, Unit 1 was in Cold Shutdown with reactor coolant temperature at approximately 118 degrees F and an indicated reactor water level of approximately 50 inches above instrument zero (approximately 214 inches above top of active fuel). At that time, instrumentation and control (I&C) technicians were attempting to rectify a mismatch between redundant reactor water level indications. While they were returning reactor water level instrument 1C32-N004B to service, a pressure spike was experienced on the instrument sensing line. The same sensing line also feeds two other reactor water level instruments, 1B21-N680C/D. The latter two instruments sensed the spike and actuated the Reactor Protection System (RPS, EIIS Code JE), resulting in a scram signal due to sensed low reactor water level. The low reactor water level signal also resulted in an automatic isolation of the Shutdown Cooling (SDC) system (RHR, EIIS Code BO). The false low reactor water level signal cleared immediately. The scram signal was reset within approximately 3 minutes, and shutdown cooling was restored by approximately 1900 CST. Reactor water temperature reached a maximum of approximately 138 degrees F per control room indications.

The root cause of the event could not be conclusively determined, but it is believed that an air bubble trapped in the instrument sensing line either moved or collapsed causing the instrument spike.

Corrective actions for the event included completing the maintenance activity on reactor water level instrumentation and returning SDC to service.

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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 2/17/91, at approximately 1840 CST, Unit 1 was in Cold Shutdown with reactor coolant temperature at approximately 118 degrees F and an indicated reactor water level of approximately 50 inches above instrument zero (approximately 214 inches above top of active fuel). At that time, instrumentation and control (I&C) technicians were attempting to rectify a mismatch between redundant reactor water level indications. While they were returning reactor water level instrument 1C32-N004B to service, a pressure spike was experienced on the instrument sensing line. The same sensing line also feeds two other reactor water level instruments, 1B21-N680C/D. The latter two instruments sensed the spike and actuated the Reactor Protection System (RPS, EIIS Code JE), resulting in a scram signal due to sensed low reactor water level. The low reactor water level signal also resulted in an automatic isolation of the Shutdown Cooling (SDC) system (RHR, EIIS Code BO). The false low reactor water level signal cleared immediately. The scram signal was reset within approximately 3 minutes, and shutdown cooling was restored by approximately 1900 CST. Reactor water temperature reached a maximum of approximately 138 degrees F per control room indications.

The root cause of the event could not be conclusively determined, but it is believed that an air bubble trapped in the instrument sensing line either moved or collapsed causing the instrument spike.

Corrective actions for the event included completing the maintenance activity on reactor water level instrumentation and returning SDC to service.

DESCRIPTION OF EVENT

On 2/17/91, at approximately 0450 CST, Unit 1 was in Cold Shutdown. At that time, plant operators noted a mismatch of approximately 15 inches between redundant reactor water level indications. The condition was reported on Deficiency Card 1-91-0746 in accordance with plant administrative control procedures.

To inspect for possible causes of the mismatch, and to verify the physical integrity of the lines, a walkdown inspection was performed of the pressure sensing lines, including those in the drywell. Neither leaks, nor any other evidence of structural damage was found. Therefore, I&C technicians began backfilling the "B" loop instruments in accordance with work instructions in Maintenance Work Order 1-91-0802 and procedure 57CP-C32-006-1S, "Reactor Water Level Loop Calibration." At approximately 1840 CST with an indicated reactor water level of approximately 50 inches above instrument zero and reactor coolant temperature at 118 degrees F as measured at the inlet to the RHR heat exchanger,

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I&C technicians began opening an instrument valve to return instrument 1C32-N004B to service. Using normal operating practice, an I&C technician slightly opened the valve, waited a few seconds for pressures to equalize, and then resumed slowly opening the valve. As the level instrument was being valved into service, control room operators received indication that a full scram had occurred due to a low reactor water level signal. Since the reactor was already shutdown with all control rods fully inserted, no rod motion occurred.

As a result of the low water level signal, the RHR SDC suction valve 1E11-F008 automatically closed, isolating the RHR SDC system and tripping the RHR pump per system design. All Primary Containment Isolation System Group 2 valves which would have closed as a result of the low level signal were already closed.

Licensed plant operators then entered procedure 34AB-OPS-044-1S, "Loss of Shutdown Cooling," and raised reactor vessel water level to approximately 53 inches above instrument zero (indicated level) to ensure sufficient inventory was present to facilitate natural circulation cooling of the core. After determining the source of the actuation and verifying that the work which caused the trip was complete, plant operators returned SDC to operation by approximately 1900 CST. Subsequent to SDC being placed back in service, the maximum reactor coolant temperature read on the RHR heat exchanger was 138 degrees F.

CAUSE OF THE EVENT

The low reactor water level signal which caused the scram came from two other level instruments, 1B21-N680C/D, located on the same sensing line as 1C32-N004B. These two instruments, though located on a common sensing line, feed opposite channels of RPS trip logic. Therefore, a single perturbation affecting the common sensing line has the capability to insert a half scram into both channels of RPS logic, causing a full scram (a fail-safe design feature). In this event, it is believed that an air bubble trapped in the sensing line caused a pressure spike as the instrument was valved into service. The pressure spike on the common sensing line caused the two RPS channels to momentarily sense a false low reactor water level, thereby initiating an RPS actuation and SDC isolation.

Since subsequent evaluation revealed that the actual reactor water level was approximately 15 inches lower than the indicated level of 50 inches above instrument zero, actual water level was apparently sufficiently close to the instrument setpoint of 12.3 inches above instrument zero for a small perturbation in the sensing line to cause the instruments to actuate.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This event is reportable per 10 CFR 50.73(a)(2)(iv) because an engineered safety feature, the Reactor Protection System, experienced an unplanned, automatic actuation due to a sensed low reactor water level, and because the same signal caused an unplanned automatic isolation of the RHR SDC system.

The Reactor Protection System automatically initiates a reactor scram to ensure the radioactive materials barriers, such as fuel cladding and pressure system

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boundary, are maintained, and to mitigate the consequences of transients and accidents. One of the initiating signals of an RPS actuation is low reactor water level. This signal is designed to shut down the reactor while maintaining sufficient reactor coolant inventory to assure adequate coverage and cooling of the fuel. In this event, RPS functioned as designed. When a low reactor water level was sensed in both channels of RPS logic due to the pressure spike on the common sensing line, a full scram signal was generated. At the time of the event, all control rods were fully inserted. Thus, no control rod movement occurred as a result of the scram.

The Shutdown Cooling mode of the RHR system is designed to remove decay heat from the core when the reactor is shut down. SDC extracts reactor vessel inventory through the Reactor Recirculation System (EIIS Code AD) piping, circulates it through a heat exchanger, and then returns it to the reactor vessel. In this event, the low reactor water level signal caused, per design, an isolation of SDC suction valve 1E11-F008 thereby interrupting SDC operation. However, SDC was restored to operation within approximately 20 minutes. Subsequent to SDC being placed back in service, the maximum reactor coolant temperature read on the RHR heat exchanger was 138 degrees F.

It should be noted the above 138 degrees F reading may not be completely indicative of actual average reactor coolant temperature since no forced circulation occurred during the time period SDC was not in operation and since the temperature was read sometime after SDC was placed back in service. To confirm that coolant temperature did not significantly increase, Southern Nuclear Operating Company Fuels group performed a calculation based on a Hatch decay heat model, with conservative assumptions, to estimate the maximum reactor coolant temperature during the 20-minute SDC loss. The calculation results showed a maximum coolant temperature of 160 degrees F, a reading well below 212 degrees F. The calculation was performed using a normal water level, as opposed to 53", since actual level was lower.

Based on the above analysis, it is concluded that this event had no adverse impact on nuclear safety.

CORRECTIVE ACTION

Corrective actions for this event included completing the maintenance activity on reactor water level instruments and restoring the RHR SDC system to service.

ADDITIONAL INFORMATION

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

2. Previous Similar Events: One event occurring in the past 2 years was identified in which work on pressure or level transmitters resulted in a reactor scram due to disturbances on common sensing lines. This event was reported in LER 50-366/1990-003, dated 4/23/90. Corrective actions for this event included revising procedures 57CP-CAL-103-2S, "ITT Barton Model 764 Differential Pressure Transmitter," and 57CP-CAL-104-1/2S, "ITT Barton Model

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763 Differential Pressure Transmitter." These corrective actions would not have prevented this event because the instruments involved were different and the procedures were different.

3. Failed Components Identification: No failed components contributed to this event.