NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
PLANT HATCH, UNIT 1	0 5 0 0 0 3 2 1	1 OF 0 7

TITLE (4): FEEDWATER CONTROLLER FAILURE CAUSES REACTOR SCRAM ON LOW WATER LEVEL

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 NAMES	DOCKET NUMBER(S)
0 9	0 4	8 8	8 8	0 1 3	0 0	1 0	0 3	8 8	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)		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 AREA CODE	
Steven B. Tipps, Manager Nuclear Safety and Compliance, Hatch	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	S J	C N V	G 0 8 4	Y	X	B J	L I S	R 2 9 0	Y
X	S J	A M P	G 0 8 4	Y					

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 fifteen single-space typewritten lines) (16)

On 9/4/88, at 1456 CDT, Unit 1 was in the run mode at approximately 100 percent rate thermal power (2436 MWT). At that time, the 'A' Reactor Feed Pump controller malfunctioned, resulting in a decrease in reactor water level. By 1501 CDT, reactor water level had been restored to normal. Power level had been reduced to approximately 66 percent.

At 1515 CDT, the 'B' Reactor Feed Pump (RFP) controller failed downscale causing the 'B' RFP to decrease in speed. This resulted in reactor water level decreasing rapidly and the Reactor Protection System actuating on low water level. The transient included Primary Containment Isolation System valve Groups 2 and 5 isolations, and the actuation of the High Pressure Coolant Injection, Reactor Core Isolation and Standby Gas Treatment Systems. The reactor water level was subsequently returned to the normal range and startup begun.

The root causes of the event included: (1) a poorly soldered connection on the signal converter board within the 'A' RFP control loop (2) failure of the 'B' RFP controller caused by a non-responsive master feedwater controller amplifier.

The corrective actions for this event included resoldering a connection and replacing the controller amplifier.

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

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 9/4/88, at 1456 CDT, Unit 1 was in the run mode at approximately 100 percent rated thermal power (2436 MWT). At that time, the 'A' Reactor Feed Pump decreased to 2000 RPM and stabilized. As a result, reactor water level decreased to +32 inches. By 1501 CDT reactor water level had been restored to normal (+37 inches) using the 'B' Reactor Feed Pump. Power level had been reduced to approximately 65 percent.

At approximately 1515 CDT, the 'B' Reactor Feed Pump controller failed downscale. With the 'A' RFP operating at 2000 RPM, and the 'B' RFP directed to minimum speed by the failed controller, reactor water level decreased rapidly causing an actuation of the Reactor Protection System [RPS, EIIS Code JC] on low reactor water level at 1516 CDT. The transient included Primary Containment Isolation System [PCIS, EIIS Code JM] valve Group 2 and Group 5 isolations, plus the actuation of the High Pressure Coolant Injection [HPCI, EIIS Code BJ], Reactor Core Isolation Cooling [RCIC, EIIS Code BN] and Standby Gas Treatment [SBGT, EIIS Code BH] Systems. The RFP controllers were repaired and returned to service, and the reactor was returned to power operation on 9/5/88.

Description of Event

On 9/4/88, Unit 1 was in steady state operation at an approximate power level of 2436 MWT and the reactor mode switch was in the run position. At approximately 1456 CDT, licensed Plant Operators observed a decrease in reactor water level to approximately +32 inches above instrument zero (normal operating level is approximately +37 inches above instrument zero), followed by a high vibration alarm on the 'A' Reactor Feed Pump (1N21-C005A). The speed of 'A' RFP decreased to 2000 RPM and stabilized, indicating the 'A' RFP controller was malfunctioning.

At approximately 1457 CDT, due to the feedwater flow for 'A' RFP having decreased to less than approximately 20 percent of rated flow, both 'A' and 'B' Recirculation Pumps (RP, E.IS Code AD) (1B31-C001A & B) began to run back to the 44% speed limiter set point, as designed. The 'B' Recirculation Pump ran back to 44%. By 1501 CDT, reactor water level had been returned to within the normal range using the 'B' RFP (1N21-C005B). The 'A' Recirculation Pump had to be controlled manually by the licensed Plant Operator to reach 44%. By 1508 CDT, both pumps were at 44% speed. It was later determined that the cause for the 'A' Recirculation Pump not running back automatically to the 44% speed set point was the result of the speed limiter (1B31-K621A) being out of tolerance due to instrument drift. The instrument was adjusted and returned to service.

At 1515 CDT, with the plant now operating at 66% power level, the 'B' RFP controller (1C32-R601B) failed downscale causing the pump to runback to minimum speed. Reactor water level decreased rapidly and the reactor automatically scrammed on a low water level signal (+12 inches) at approximately 1516 CDT. Water level continued to decrease and both 'A' and 'B' Recirculation Pumps automatically tripped at -30 inches reactor water level, per design. PCIS valve Groups 2 and 5 isolated at +12 inches and -35 inches, per design. HPCI, RCIC, and SBGT (both Unit 1 and Unit 2 trains) systems also received auto initiation signals at -35 inches, and RCIC and SBGT successfully initiated immediately, per design. All of the above actions happened within two minutes of the 'B' RFP controller downscale failure.

Although HPCI received an auto initiation signal, it did not immediately come to rated speed and inject into the reactor vessel due to the presence of water in the steam supply line, which caused the turbine to trip on a high turbine exhaust pressure signal. (It was later verified that the level switch at the HPCI inlet drain pot was not functioning which allowed water to accumulate.) The first two unsuccessful HPCI auto start attempts resulted in the water being forced out of the steam supply piping. The HPCI system then initiated and injected as required on the third auto start signal (which was within 30 seconds of the initial actuation signal); no manual intervention by the plant operations personnel was necessary. The 'B' RFP was placed in manual control and used with HPCI and RCIC to return reactor water level to within the normal range (+32 inches to +42 inches). The scram signal was reset at approximately 1519 CDT. Reactor water level was stabilized by 1600 CDT.

Cause of Event

The immediate cause for the reactor scram on low reactor water level was the failure of both the 'A' and 'B' RFP controllers.

The root cause for the failure of the 'A' RFP controller was determined to be equipment failure. Specifically, an inadequately soldered connection (cold joint) on the manufacturer supplied signal converter board appeared to be the primary cause of the controller's observed failure mode (a decreasing output signal). Additionally a loose capacitor, due to a loose termination screw, on the signal converter board may have been a contributing factor in the controller's failure. Other circuit boards in the 'A' RFP controller were inspected for inadequately soldered connections and loose connections and no other problems were found.

The root cause for the failure of the 'B' RFP controller has been concluded to be equipment failure. Initial investigation of the controller's failure revealed a loose cable connection between the controller (1C32-R601B) and the controller amplifier unit (1C32-K674). However, it could not be conclusively determined if this loose connection existed while the controller was in the control room panel or resulted when the controller was removed, following the scram, for investigation of its failure. With the cable connection secured, a complete instrument loop check was performed and demonstrated no discrepancies. However, after further investigation and observation of the controller's functioning, the master feedwater controller amplifier (1C32-K637) appeared to be malfunctioning, causing the same 'B' RFP controller problem, and was concluded to be the most probable root cause of the 'B' RFP controller failure.

The immediate cause of the two HPCI turbine trips was a high turbine exhaust pressure signal caused by water in the HPCI steam supply lines. The presence of water in the HPCI steam supply lines was due to the following two failures: (1) clogged HPCI steam trap and/or strainer and (2) a failed dual station snap switch in the inlet drain pot level switch (1E41-N014). Condensed steam had accumulated in the inlet drain pot enough to have backed up into the HPCI steam supply lines. Due to its maintenance history, the dual station snap switch had previously been determined to be sensitive to the high temperature environment where it was located, and replacement with a new device which has a higher tolerance for heat had already been scheduled for installation during the current Unit 1 outage. The root cause of the failure of the switch in this event was its heat sensitivity.

Reportability Analysis and Safety Assessment

This report is required per 10 CFR 50.73(a)(2)(iv) because an unplanned actuation of the Reactor Protection System (RPS) and Engineered Safety Features (ESF) occurred. Specifically, the RPS was initiated automatically on low reactor water level. The other ESFs which activated during this event were the Primary Containment Isolation System valve Group 2 and Group 5, plus the High Pressure Coolant Injection and the Standby Gas Treatment Systems.

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

The RPS provides timely protection against the onset and consequences of conditions that could threaten the integrity of the fuel barriers and the nuclear system process barrier. A reactor scram initiated by a low water level condition, protects the fuel by reducing the fission heat generation within the core.

In this event, the decrease in vessel level was a direct result of the failures of the 'A' and 'B' RFP controllers. The RPS functioned per design. Reactor water level was restored by using a Reactor Feed Pump, plus the High Pressure Coolant injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems.

These prompt corrective actions rapidly terminated power operation and restored monitored plant parameters (such as reactor water level) to their nominal values.

The High Pressure Cooling Injection System, designed to ensure adequate core coverage in low reactor water level conditions, in this event injected within 30 seconds of demand even with the existence of water in the HPCI steam supply lines. Had HPCI not injected, other Engineered Safety Features, such as the Automatic Depressurization System [ADS, EIIS Code JE], Low Pressure Coolant Injection [LPCI, EIIS Code BO], and Core Spray [CS, EIIS Code BM] systems were available to supply water to the reactor.

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

Corrective Actions

A complete instrument loop check was performed on the 'A' RFP control circuit to ensure that the controller was working properly. A loose termination screw was found and tightened on capacitor C11 in the signal converter. In addition, an inadequately soldered connection (cold joint) was identified and repaired

Control amplifier (1C32-K637) in the master feedwater controller circuit was replaced to correct the malfunction of the 'B' RFP controller which was functioning erratically and sluggishly. This control amplifier was replaced, with feedwater responding per design.

The dual station snap switch, which is part of the HPCI steam supply drain pot level switch, was replaced and functionally tested to verify proper operation of the drain valve, 1E41-F054. The scheduled replacement for the existing level switch, per Design Change Request 87-007, has a higher level of tolerance to heat. The new device, the F.C.I. High Temperature Liquid Level and Interface Control (Model HT66), is scheduled for installation during the current Unit 1 outage. In addition, all Unit 1 HPCI steam trap and strainers are scheduled for inspection and cleaning during this outage. The Unit 2 HPCI steam traps and strainers were inspected and cleaned when the HPCI inlet drain pot level switch (2E41-N014) was replaced with the new, high temperature level control device during the last Unit 2 outage.

The procedures 52PM-E41-001-0S, "HPCI System Maintenance", and 52PM-E51-004-0S, "RCIC System Maintenance", will be revised to include a 12 month maintenance inspection of the drain pot strainers and steam traps. This inspection will check for and remove flow blockage, plus determine if these components are working properly. These procedures will be validated by the end of this outage (approximately 12/9/88).

Additional Information

1. FAILED COMPONENT(s) IDENTIFICATION

a. MPL: 'A' RFP controller (1C32-R601A)
Manufacturer: General Electric — Root Cause Code: X
Model Number: 3S7513TC108A2 — Component Code: CNV
Type: N/A — Manufacturer Code: G084
EIIS Code: SJ
Reportable to NPRDS: Yes

b. MPL: Master Feedwater Controller Amplifier (1C32-K637)
Manufacturer: General Electric
Model Number: 543-03 — Root Cause Code: X
Type: N/A — Component Code: AMP
EIIS Code: SJ — Manufacturer Code: G084
Reportable to NPRDS: Yes

c. MPL: HPCI Inlet Drain Pot level Switch (1E41-N014)
Manufacturer: Robertshaw — Root Cause Code: X
Model Number: 85239A1 — Component Code: LIS
Type: N/A — Manufacturer Code: R290
EIIS Code: BJ
Reportable to NPRDS: Yes

2. PREVIOUS SIMILAR EVENTS

There were two previous events similar to the one described in this LER. They were reported in LER 50-321/1987-013 (dated 9/2/87) and LER 50-366/1988-020 (dated 9/6/88). These two LERs describe events where feedwater was lost due either to tripping of the Reactor Feed Pumps on low suction pressure or a failure in the master feedwater control circuitry. Both events resulted in a decrease in Reactor Feed Pump flow and subsequent reactor scrams.

In LER 50-321/1987-013, feedwater was lost when two capacitors, in the master feedwater controller amplifier, short circuited causing a loss of voltage output signal to the feed pump controllers. The corrective action for this event was the replacement of the defective capacitors in the controller amplifier with capacitors from another vendor.

In LER 50-366/1988-020, feedwater was lost when a fuse blew in an electrical circuit containing Condensate and Feedwater system controllers. The corrective actions for this event included: 1) replacing a failed fuse in panel 2H11-P622, 2) replacing water level transmitter 2B21-N081B and 3) initiating a design review of feedwater control circuitry to identify "ganged" circuits.

The corrective actions for these events would not have prevented the event described in LER 50-321/1988-013 because the root causes of the events were different.

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the southern electric system

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October 3, 1988

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

PLANT HATCH - UNIT 1
NRC DOCKET 50-321
OPERATING LICENSE DPR-57
LICENSEE EVENT REPORT
FEEDWATER CONTROLLER FAILURE CAUSES
REACTOR SCRAM ON LOW WATER LEVEL

Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(i.), Georgia Power Company is submitting the enclosed Licensee Event Report (LER) concerning the unanticipated actuation of some Engineered Safety Features (ESFs). This event occurred at Plant Hatch - Unit 1.

Sincerely,

R. P. McDonald

DPR/ct

Enclosure: LER 50-321/1988-013

c: (see next page)

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Georgia Power

U. S. Nuclear Regulatory Commission
October 3, 1988
Page Two

c: Georgia Power Company
Mr. H. C. Nix, General Manager - Plant Hatch
Mr. L. T. Gucwa, Manager, Licensing and Engineering - Hatch
GO-NORMS

U. S. Nuclear Regulatory Commission, Washington, D. C.
Mr. L. P. Crocker, Licensing Project Manager - Hatch

U. S. Nuclear Regulatory Commission, Region II
Dr. J. N. Grace, Regional Administrator
Mr. J. E. Menning, Senior Resident Inspector - Hatch

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