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 2	0 5 0 0 0 3 6 6	1 OF 0 9

TITLE (4)
DEFICIENT PROCEDURE CAUSES LOSS OF FEEDWATER RESULTING IN REACTOR SCRAM

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 5	2 7	8 8	8 8	0 1 7	0 0	0 6	2 7	8 8		0 5 0 0 0
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THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

OPERATING MODE (9): 1
POWER LEVEL (10): 0 9 8

[ ] 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	B N	3 3	L 2 0 0	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES (If yes, complete EXPECTED SUBMISSION DATE) [X] NO

EXPECTED SUBMISSION DATE (15): MONTH / DAY / YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On 05/27/88 at approximately 0407 CDT, Unit 2 was in the run mode at an approximate power level of 2384 MWt (approximately 98% of rated thermal power). In the process of filling and venting condensate pump 2N21-C001B (EIIS Code SJ) an air bubble was released into the condensate system tripping the condensate booster pumps (EIIS Code SJ) and reactor feed pumps (EIIS Code SJ) on low suction pressure. Reactor water level dropped and the reactor scrammed on low water level. In the event the Reactor Core Isolation Cooling (RCIC EIIS Code BN) system did not function as anticipated.

The root cause of this event was a deficient condensate and feedwater system operating procedure. The procedure did not give instructions for fillin and venting a condensate pump while the unit was on line.

Corrective actions for this event are: 1) replacing the limit switch on a RCIC valve, 2) verbally making involved personnel aware of cause and consequences of this event, and 3) scheduling revision to the condensate and feedwater system operating procedures for both units. The procedures will be revised to include instructions on filling and venting the condensate pumps.

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A. REQUIREMENT FOR THE REPORT:

This report is required by 10 CFR 50.73 (a)(2)(iv), because of the unanticipated actuation of the Reactor Protection System (RPS-EIIS Code JC) and of some Engineered Safety Features (ESFs). The ESFs that actuated were: Primary Containment Isolation System (PCIS-EIIS Code JM), the High Pressure Coolant Injection System (HPCI-EIIS Code BJ), and Standby Gas Treatment (SBGT-EIIS Code BH). The Safety Relief Valves (SRVs - EIIS Code JE) were manually actuated and HPCI was both manually and automatically initiated while all other ESF actuations were automatic per system design.

B. UNIT(s) STATUS AT TIME OF EVENT:

1. Power Level/Operating Mode:

Unit 2 was in steady state operation at an approximate power level of 2384 MWt (approximately 98 percent of rated thermal power). The reactor mode switch was in the run position.

2. Inoperable Equipment:

There was no inoperable equipment that contributed to this event.

C. DESCRIPTION OF EVENT:

1. Event:

On 5/27/88 at approximately 0400 CDT, Plant operations personnel were performing clearance 2-88-928 by filling and venting condensate pump (EIIS Code SJ) 2N21-C001B in preparation for placing the pump in service later in the day. To vent the suction side of the pump, the low pressure condensate vent valve, 2N21-F021B, was opened. Opening this valve establishes a path to the main condenser from the pump. The condenser was at a pressure of approximately 3 psia, while the isolated pump was at approximately atmospheric pressure (14.7 psia).

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Therefore, a suction was drawn on the pump well and its suction line. At approximately 0407 CDT, the condensate suction pump valve, 2N21-F001B, was partially opened to allow the piping and pump well to fill slowly with water.

At approximately the same time, reactor water level began to decrease due to a loss of condensate booster pump suction and a low suction pressure alarm was received in the main control room. Approximately four seconds later, the "A" and "B" condensate booster pumps (EIIS Code SJ) and the "A" and "B" Reactor Feedwater Pumps (RFPs EIIS Code SJ) tripped on low suction pressure. The "C" condensate booster pump, which had been in standby, autostarted per design on the loss of the "A" and "B" pumps.

As a result of the loss of feedwater the reactor water level decreased. Approximately eight seconds after the pumps tripped water level reached approximately 12.3 inches above instrument zero causing an automatic scram signal and a PCIS valve group 2 isolation to occur, per design. Approximately nine seconds after the pump tripped, the operator restarted the RFP "B" to inject into the vessel. Water level had reached approximately +0 inches above instrument zero at this time.

At approximately 0407 CDT, water level was at approximately 35 inches below instrument zero (Water Level 2). At this level HPCI, Reactor Core Isolation Cooling (RCIC - EIIS Code BN), Alternate Rod Insertion (ARI - EIIS Code AA), and SBGT initiated and PCIS valve group 5 isolated, per design.

Reactor water level reached its lowest level of approximately 66 inches below instrument zero (approximately 98 inches above the Top of Active Fuel [TAF]) at approximately 0408 CDT. Water level then increased due to HPCI and RFP "B" injecting into the vessel. RCIC, though it correctly started, failed to inject into the vessel.

When water level reached approximately 25 inches above instrument zero, the operator manually tripped HPCI and RCIC to slow the water level increase. Water level continued to increase and at approximately 0411 CDT, RFP "B" tripped, as designed, on a high water level of approximately 56 inches above instrument zero.

Water level continued to increase due to injection by the Control Rod Drive (CRD - EIIS Code AA) and water "swell". Water level increased to greater than 60 inches above instrument zero (full scale on the normal operating range water level instruments). The only water level indication on range was the shutdown flooding range level instrument which is not temperature or pressure compensated. At 0412 CDT, the Main Steam Isolation Valves (MSIVs - EIIS Code JM) were closed (by plant operations personnel as a conservative action), to prevent water entry into the Main Steam Lines (MSLs - EIIS Code SB). The MSIVs are located at approximately 123 inches above instrument zero.

The highest water level reached during the event was approximately 74 inches above instrument zero. The closure of the MSIVs isolated the reactor from the main condenser. Therefore, the reactor pressure started to rise.

Plant operations personnel notified the NRC of the scram, in accordance with the requirements of 10 CFR 50.72, at approximately 0430 CDT.

To control reactor pressure, per the Emergency Operating Procedures (EOPs), the operator individually manually cycled SRVs. The "M" SRV was cycled at approximately 0430 CDT and the "B" SRV was cycled at approximately 0440 CDT.

Each SRV opening also reduced reactor water level. At approximately 0443 CDT, the operator attempted to manually place RCIC into service but was unable to develop sufficient turbine speed. At approximately 0449 CDT, the operator again tried unsuccessfully to place RCIC into service.

To control reactor pressure, the operator continued to individually manually cycle SRVs. At approximately 0451 CDT he cycled the "G" SRV. The "F" SRV was cycled to control pressure at approximately 0500 CDT.

The water level had dropped to approximately 17 inches above instrument zero at approximately 0501 CDT. To increase water level, the operator again attempted to start RCIC but again RCIC could not develop sufficient speed. At approximately 050[illegible] CDT, the operator manually initiated HPCI and injected into the vessel. HPCI automatically tripped at approximately 0505 CDT on a high water level of approximately 51 inches above instrument zero.

To control reactor pressure, the "D" SRV was cycled at approximately 0601 CDT. To control reactor pressure and water level, the operator manually initiated HPCI at approximately 0718 CDT and again at 0830 CDT.

At approximately 0900 CDT, the inboard MSIVs were reopened, (the outboard MSIVs were opened at approximately 0509 CDT) and at approximately 1050 CDT operations personnel were able to place the RWCU system in service. With the CRD system injecting water, the reactor was in a stable condition at approximately 1050 CDT.

2. Other Systems Affected:

No safety systems other than those described were affected by this event. The only safety system with secondary functions in this event is the SRVs. Its other functions are the Automatic Depressurization System (ADS EIIS Code JE) and the Low Low Set (LLS EIIS Code JE) modes of operation. These modes were not required during this event.

3. Method of Discovery:

This event was discovered by licensed operations personnel by observation of control room indications.

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4. Operator Actions:

Operations personnel performed the following actions:

1. Operated valves to fill and vent the "B" condensate pump.

2. Responded to the event using the Emergency Operating Procedures.

3. Made a 10 CFR 50.72 report to the NRC.

5. Auto/Manual Safety System Response:

No manual or automatic safety system actuations occurred other than those described.

D. CAUSE OF EVENT:

1. Immediate Cause:

The immediate cause of this event is loss of sufficient flow from condensate pumps "A" and "C" causing the condensate booster pumps and reactor feedwater pumps to trip on low suction pressure. When the condensate pump "B" suction valve, 2N21-F001B, was opened, the air inside the pump well and piping, at an approximate pressure of 14.7 psia, expanded down the common suction pipe for the three condensate pumps (the pressure in the suction pipe was approximately 3 psia) causing the pumps to lose flow.

2. Root/Intermediate Cause:

The intermediate cause of this event is cognitive personnel error. The licensed Senior Reactor Operator (SRO) who wrote the amplifying instruction to clearance 2-88-928 did not anticipate the potential for air binding of the operating condensate pumps using his chosen restoration methodology. This evolution had been performed only once previously during power operations.

The root cause of this event is a deficient condensate and feedwater system operating procedure. Procedure 34SO-N21-007-2S did not give instructions for filling and venting a condensate pump while the unit was on line.

E. ANALYSIS OF EVENT:

The RPS provides timely protection against the onset and consequences of conditions that could threaten the integrities of fuel barriers and the nuclear system process barrier.

A low water level in the reactor vessel indicates that the reactor is potentially in danger of being inadequately cooled. Should reactor water level decrease too far, fuel damage could result. A reactor scram, initiated by a low water level condition, protects the fuel by reducing the fission heat generation within the core.

The HPCI system is provided to assure that the reactor core is adequately cooled to limit fuel clad temperatures in the event that reactor water level decreases without a rapid depressurization of the reactor vessel. The HPCI system maintains sufficient reactor vessel water level inventory until the vessel is depressurized.

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The SRVs are installed to prevent overpressurization of the reactor vessel. They also allow operations personnel to depressurize the reactor vessel and to control reactor pressure if the reactor is isolated from the main condenser.

In this event, the decrease in the vessel level was correctly sensed by the RPS and appropriate protective actions (such as the scram and the HPCI initiation and injection) occurred. These corrective actions not only rapidly terminated power operations, but also restored monitored plant parameters (such as reactor water level) to their nominal values.

Based on the above information and since this event happened at near rated power, it is concluded that this event had no adverse impact on nuclear plant safety. Additionally, it is concluded that the consequences of this event would not be more severe under other operating conditions.

F. CORRECTIVE ACTIONS:

The corrective actions for this event included:

1. Replacing the limit switch on valve 2E51-F045. The limit switch failed to pick-up the relay, which provides the ramp switch signal to the RCIC Woodward Controller, which resulted in the Woodward Controller not responding to speed demands.

2. Involved personnel were verbally made aware of the cause and consequences of this event

3. Scheduling revision of the Unit 1 and Unit 2 condensate and feedwater system operating procedures (34SO-N21-007-1S and 34SO-N21-007-2S) by approximately 11/01/88. The revisions will include specific instructions on filling and venting the condensate pumps during power operations as an infrequently performed evolution.

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G. ADDITIONAL INFORMATION:

1. Failed Component(s) Identification:

MPL (Plant Index Identifier): 2E51-F045
Manufacturer: Limitorque
Model Number: B/M10403
Type: Geared Limit Switch
EIIS: BN

2. Previous Similar Events:

One previous event similar to the one described in this LER has been noted. This event was reported by LER 50-366/1987-008 (dated 5/22/87).

This LER described an event where the reactor scrammed and a HPCI injection occurred because a condensate booster pump tripped. The pump trip was caused by a hard electrical ground condition in the pump's motor.

Corrective actions for this event included replacing the motor, sending the grounded motor to a repair facility for investigation and repair, reviewing maintenance history on the failed motor, and performing maintenance on the main feed pump turbine trip circuitry.

These corrective actions would not have prevented the event described by LER 50-366/1988-017 because the referenced corrective actions dealt with electrical components rather than procedural problems with placing the pumps into service.

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Nuclear Operations Department

Georgia Power
the southern electric system

SL-4865
0343I
X7GJ17-H310

June 27, 1988

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

PLANT HATCH - UNIT 2
NRC DOCKET 50-366
OPERATING LICENSE NPF-5
LICENSEE EVENT REPORT
DEFICIENT PROCEDURE CAUSES LOSS OF
FEEDWATER RESULTING IN REACTOR SCRAM

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 the unanticipated actuation of some Engineered Safety Features (ESFs). The event occurred at Plant Hatch - Unit 2.

Sincerely,

W. G. Hairston III

W. G. Hairston, III
Senior Vice President

BF/ct

Enclosure: LER 50-366/1988-017

c: (see next page)

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Georgia Power

U. S. Nuclear Regulatory Commission
June 27, 1988
Page Two

c: Georgia Power Company
Mr. J. T. Beckham, Jr., Vice President - Plant Hatch
Mr. L. T. Gucwa, Manager Nuclear Safety and Licensing
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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