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): PLANT HATCH, UNIT 2

DOCKET NUMBER (2): 0 5 0 0 0 3 6 6

PAGE (3): 1 OF 0 7

TITLE (4): DESIGN DEFICIENCY CAUSING BLOWN FUSE, LOSS OF FEEDWATER, AND SUBSEQUENT REACTOR SCRAM

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 NAMES	DOCKET NUMBER(S)
08	05	88	88	020	00	09	06	88		0 5 0 0 0
										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: Steven B. Tipps, Manager Nuclear Safety and Compliance, Hatch

TELEPHONE NUMBER: AREA CODE 912 367-7851

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
B	SJ	FU	M175	N					
X	JC	PDT	R369	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 08/05/88, at approximately 1546CDT, Unit 2 was in the run mode at an approximate power level of 2424 MWt (approximately 100 percent of rated thermal power). When reinstalling a repaired Feedwater Minimum Flow Controller , a fuse blew, causing minimum flow valves on the Condensate, Condensate Booster and Reactor Feed Pumps to fail open. This resulted in a loss of suction and subsequent tripping of the Condensate Booster Pumps and Reactor Feed Pumps.

The rapid decrease in reactor water level resulted in a reactor scram, a Primary Containment Isolation system valve Group 2 and partial Group 5 isolation. Reactor water level was restored using the 'A' Reactor Feed Pump, the High Pressure Coolant Injection System and the Reactor Core Isolation Cooling System.

The event was caused by a design deficiency. An electrical circuit containing twelve instruments, including four Condensate and Feedwater system controllers, had only one in-line fuse. As a result, an in-line fuse (2N21-F7) failed which resulted in a loss of suction for and the tripping of the Condensate Booster Pumps and the Reactor Feed Pumps. The fuse was replaced and a design review of the Feedwater Control System was initiated to identify and change any similar deficient conditions.

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FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
PLANT HATCH, UNIT 2	0 5 0 0 0 3 6 6	8 8	— 0 2 0 —	0 0	0 2 OF 0 7

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

A. REQUIREMENT FOR REPORT

This report is required per 10 CFR 50.73 (a)(2)(iv) because unplanned actuations of the Reactor Protection System (RPS EIIS CODE JC) and Engineered Safety Features (ESF) occurred. Specifically, RPS was initiated automatically on low reactor water level. The ESFs which actuated were the Primary Containment Isolation System (PCIS EIIS Code JM) valve Group 2 and Group 5 (partial isolation since one of the two valves failed to close) and the High Pressure Coolant Injection System (HPCI EIIS CODE BJ).

B. UNIT(s) STATUS AT TIME OF EVENT

1. Power Level/Operating Mode:

Unit 2 was in a steady state operation at an approximate power level of 2424 MWt (approximately 100 percent of rated thermal power). The reactor mode switch was in the Run position.

2. Inoperable Equipment:

The Reactor Feedwater Minimum Flow Controller (2N21-R384G) had been removed from control room panel 2H11-P662, on the morning of 8/5/88, for repair per Maintenance Work Order (MWO) #2-88-3280.

C. DESCRIPTION OF EVENT

1. Event

On 8/5/88, at approximately 1546 CDT, two Instrument and Control (I & C) technicians requested and received approval to return a repaired Feedwater Minimum Flow Controller (2N21-R384G) to service in control room panel 2H11-P622. In the process of reinstallation, a short occurred in the controller connection, which blew the circuit in-line fuse 2N21-F7. The failure of this fuse caused four controllers to become inoperable, resulting in the minimum flow valves for the Condensate Pumps (2N21-C001A & 2N21-C001B), the Condensate Booster Pumps (2N21-C002A & 2N21-C002B), and the Reactor Feed Pumps (2N21-C005A and 2N21-C005B) (EIIS-Code SJ) failing open.

The 'failing open' of the minimum flow valves for the Condensate Pumps reduced feedwater flow which resulted in inadequate suction pressure at the Condensate Booster Pumps. The Condensate Booster Pumps (2N21-C002A and 2N21-C002B) tripped on low suction pressure. This caused Condensate Booster Pump 2N21-C002C to auto start. One booster pump, however, was not sufficient to provide adequate flow to the Reactor Feed Pumps (RFP EIIS Code SJ). Both Reactor Feed Pumps (2N21-C005A & 5B) tripped on loss of suction pressure.

Due to the loss of the Reactor Feed Pumps, the reactor water level began to decrease. At 1546 CDT, the reactor automatically scrammed on "Reactor Low Water Level" and a PCIS valve Group 2 isolation occurred. As the water level continued to decrease, the Main Turbine (MT EIIS code TA) was manually tripped by the licensed plant operator to limit inventory loss (removal of steam) from the reactor. The Recirculating Water Pumps (2B31-C001A & 1B), which had run back to the number #1 speed limiter (due to feedwater flow being less than 20%), tripped at approximately -30 inches in reactor water level. As water level continued to decrease, the High Pressure Coolant Injection (HPCI EIIS Code BJ) and the Reactor Core Isolation Cooling (RCIC EIIS code BN) Systems auto initiated (at the setpoint of -35.0 inches of water) and injected into the reactor. A partial Group 5 isolation was received . The Reactor Water Cleanup (RWCU EIIS Code CE) outboard isolation valve (2G31-F004) closed per design on Reactor Low Low level. The inboard isolation valve 2G31-F001, however, did not close as required. (The partial isolation was discovered when operations personnel reset the isolation as reactor level was recovered.)

Reactor Water Level reached its lowest level of -65 inches of water (99.44 inches above top of active fuel) and water level began to rise. By 1547, one Feedwater Pump (2N21-C005A) had been reset and was running.

Water Level continued to rise and had reached a water level of +40 inches by 1549 CDT. At this point, HPCI was secured to limit water level increase. At approximately 1550 CDT, as the licensed plant operator was preparing to manually trip RCIC, both RCIC and the 'A' Reactor Feed Pump tripped per design on High Reactor Water Level (+51.5 inches and +56.5 inches, respectively). The partial Group 5 isolation was reset to place the Reactor Water Cleanup System back in service to reduce reactor water level to the normal operating level. By 1616 CDT, water level was stabilized and the 'A' Reactor Feed Pump was used to maintain level in the normal range of +32 inches to +42 inches.

The one hour notification on the reactor scram was made to the NRC at 1640 CDT. The four hour notification of the failure of 2G31-F001 to isolate was made to the NRC at 1928 CDT. These notifications were made per 10 CFR 50.72 reporting requirements.

2. Other Systems Affected:

The RPS and HPCI systems activated, as well as a PCIS valve Group 2 and a partial valve Group 5 isolation. These systems have no secondary functions.

3. Method of Discovery:

The reactor scram and associated ESF actuations was discovered by licensed operations personnel by observation of control room indications.

4. Operator Actions:

Operations personnel responded to the reactor scram in accordance with Emergency Operating Procedures.

5. Auto/Manual System Response:

An automatic RPS actuation occurred and scrammed the reactor. A PCIS valve Group 2 isolation occurred with all Group 2 valves closing per design on low reactor water level. A 'partial' Group 5 isolation was received with only valve 2G31-F004 activating. HPCI and RCIC systems auto initiated per design on low low reactor water level .

D. CAUSE OF EVENT

1. Immediate Cause:

The immediate cause for the loss of feedwater was determined to be the failure of fuse 2N21-F7 during the installation of minimum flow controller 2N21-F384G per MWO 2-88-3280. The fuse failure caused the minimum flow valves of the Condensate, Condensate Booster, and Reactor Feed Pumps to fail open. This action resulted in a loss of suction pressure to the Condensate Booster and Feedwater Pumps resulting in their subsequent trip.

The immediate cause of valve 2G31-F001 failing to isolate was due to the water level transmitter, 2B21-N081B, reacting sluggishly.

2. Root/Intermediate Cause

The root cause for this event is a design deficiency which "ganged" four minimum flow controllers with eight other instruments in a circuit with only one in-line fuse for protection. No fused connectors were provided in case of inadvertent arcing/shorting of the circuit.

The root cause of transmitter 2321-N081B reacting sluggishly could not be conclusively determined. This Rosemount transmitter is being shipped to Rosemount for testing with results expected in September, 1988. It is thought that the transmitter had a failed diaphragm seal which prevented a timely isolation of valve 2G31-F001.

E. ANALYSIS OF EVENT

The RPS provides timely protection against the onset and consequences of conditions that could threaten the integrities of the 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.

In this event, the decrease in vessel level was a direct result of the tripping of the Condensate Booster and Feedwater pumps. Operations personnel reacted quickly to manually trip the Main Turbine to reduce removal of steam from the reactor vessel. The RPS functioned per design. Operations personnel restored reactor water level 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 operations (and energy generation) and restored monitored plant parameters (such as reactor water level) to their nominal values.

The PCIS valve Group 5 isolation on low low reactor water level provides protection against release to the environment of radioactive materials. This is accomplished by the complete isolation of system lines that penetrate the containment. Only one line of the RWCU system penetrates the containment. It contains two isolation valves and only one of these valves is required to ensure the isolation capability. In this event, one of the two isolation valves, 2G31-F004, closed accomplishing the isolation of the RWCU system.

Based on the above information, it is concluded that this event had no adverse impact on nuclear safety. Since this event occurred at a high reactor power level, it is felt this event would not have been worse under other conditions.

F. CORRECTIVE ACTIONS

The corrective actions for this event included:

1. Replacing the failed fuse (2N21-F7) in panel 2H11-P662.

2. Replace the water level transmitter, 2B21-N081B.

3. Initiating a design review of the feedwater control circuitry to identify similar "ganged" circuits. The review will be completed by October 31, 1988. Design Change Requests to correct any similar identified design deficiencies will be developed based on the results of this review.

G. ADDITIONAL INFORMATION

1. FAILED COMPONENT(s) IDENTIFICATION

a. MPL (Plant Index Identifier): Panel 2H11-P662, Fuse #F-7
Manufacturer: McGraw-Edison Co. Root Cause Code: B
Model Number: FRN-R3 Component Code : FU
Type: Fusetron Manufacturer Code: M175
EIIS: SJ
Reportable to NPRDS: No

b. MPL (Plant Index Identifier: 2B21-N081B
Manufacturer: Rosemount, Inc. Root Cause X
Model Number: 1154DP5 System Code: JC
Type: 1154 Component Code: PDT
EIIS: JC
Reportable to NPRDS: Yes

2. PREVIOUS SIMILAR EVENTS

No similar events have occurred at Plant Hatch.

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September 6, 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
DESIGN DEFICIENCY CAUSING BLOWN FUSE,
LOSS OF FEEDWATER, AND SUBSEQUENT 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

DPR/ct

Enclosure: LER 50-366/1988-020

c: (see next page)

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Georgia Power

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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