NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

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 1 0

TITLE (4)
CIRCUIT BREAKER FAILS CAUSING POWER FAILURE RESULTING IN 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)
0 8	0 3	8 7	8 7	— 0 0 9	— 0 1	0 1	1 8	8 8		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 9 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)	X 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)	Tech. Spec. 3.5.1
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)
Raymond D. Baker, Nuclear Licensing Manager - Hatch	4 0 4 5 2 6 - 7 0 1 6

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	E E	5 2	H 1 4 1	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 8/3/87 at approximately 1152 CDT, Unit 2 was in the run mode at an approximate power level of 2193 MWt (approximately 90 percent of rated thermal power). At that time, Vital AC (EIIS Code EE) power was lost. This resulted in a decrease in the reactor feedwater pumps flow and a decrease in reactor water level. The reactor water level decreased to the Reactor Protection System (RPS EIIS Code JC) actuation setpoint and a reactor scram occurred.

The root cause of this event is electrical equipment failure. Specifically, circuit breaker CB-4 would open under unduly low force conditions. It was concluded after field testing and consultation with the manufacturer that the tripping mechanism was weak.

Corrective actions for this event included: 1) installing jumpers and removing equipment from service, 2) designing and installing barrier boxes, 3) verifying trip instrumentation and level transmitters in calibration, 4) venting instrument lines and transmitters, 5) performing evaluations of air entrainment and spiking in instrument lines, 6) initiating procedure revisions, and 7) verifying certain other systems do not have low suction trips.

IE22 1/1

A. REQUIREMENT FOR REPORT

This report is required per 10 CFR 50.73 (a) (2) (iv), because an unplanned actuation of the Reactor Protection System (EIIS Code JC) and the following Engineered Safety Features (ESFs) occurred: 1) Primary Containment Isolation System (PCIS EIIS Code JM) Groups 2 and 5, 2) Unit 2 Standby Gas Treatment System (SGTS EIIS Code BH), 3) High Pressure Coolant Injection (HPCI EIIS Code BJ).

This report meets the additional reporting requirements of the Unit 2 Technical Specifications Section 3.5.1, Action C.

B. UNIT(s) STATUS AT TIME OF EVENT

Unit 2 was in the run mode at an approximate power level of 2193 MWt (approximately 90 percent of rated thermal power).

Vital AC (EIIS Code EE) power was being supplied from the alternate power supply source (600V Bus 2C) because the Vital AC inverter (normal supply) was out of service for trouble shooting/repair. This configuration is an approved plant configuration. However, in this configuration, there is no backup power supply to the Vital AC system if the alternate power supply is lost.

At 1018 CDT, the HPCI system had auto-transfered from its normal source of suction (the Condensate Storage Tank - CST EIIS Code KA) to its alternate source of suction (the torus).

C. DESCRIPTION OF EVENT

On 8/3/87 at approximately 1152 CDT, Unit 2 was in steady state operation when the alternate Vital AC power was lost. The loss of this source of electrical power caused a partial PCIS Group 5 isolation for the Reactor Water Cleanup (RWCU EIIS Code CE) system. Additionally, the loss of electrical power caused a loss of control power to the reactor feedpump (RFP) control logic (EIIS Code SD). Per design, the reactor feedwater pumps decreased in speed. Reactor water level decreased.

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

At 1153 CDT (all times are from the Safety Parameter Display System [SPDS EIIS Code IQ]), the reactor water level reached the RPS scram setpoint and the PCIS Group 2 isolation setpoint (approximately 10 inches above instrument zero). A reactor scram and Group 2 isolation occurred approximately 20 seconds after power was lost.

As reactor water level decreased to approximately 47 inches below instrument zero, the following protective actuations occurred: 1) full PCIS Group 5 isolation, 2) Unit 2 SGTS automatic start, 3) HPCI initiation, and 4) Reactor Core Isolation Cooling (RCIC EIIS Code BN) initiation.

While RCIC auto-started and injected as anticipated, HPCI did not operate as expected. The HPCI system received a valid auto-initiation signal, injected, and a normal pump turbine startup occurred. As HPCI approached rated flow conditions, the turbine tripped on a low pump suction pressure signal. The trip signal cleared and the turbine trip reset automatically, per design.

Since the initiation signal was still present, the HPCI turbine began, again, to ramp up to rated speed. However, the pump discharge gate valve, 2E41-F006, had received an auto-close signal from the first turbine trip and was continuing to close when the turbine reset itself. The discharge valve logic is such that once the valve has begun to stroke closed, it can not be stopped in mid-position and re-opened. The valve must continue to travel until the close stroke is completed. The valve then closed completely, momentarily stopping flow to the reactor vessel. The valve immediately reopened (since the initiating signal was still present), and HPCI delivered rated flow to the reactor vessel (second injection, for this event).

The lowest reactor water level reached in this event was 77.1 inches below instrument zero. Reactor water level was stabilized and recovered by both HPCI and RCIC. Maintenance personnel locally restored Vital AC power by re-closing the circuit breaker which had tripped.

At 1157 CDT, operations personnel secured the HPCI and RCIC systems since, with the re-establishment of alternate Vital AC power, they were able to use the "A" RFP to bring reactor water level to approximately 56 inches above instrument zero. When reactor water level reached this point, the "A" RFP tripped on high reactor water level. This RFP was manually restarted by operations personnel and used in conjunction with the "B" Control Rod Drive pump (EIIS Code AA) to maintain reactor level.

During this event, reactor pressure varied between approximately 990 and 635 psig. Operations personnel were able to control reactor pressure using the main turbine bypass valves (EIIS Code SO). Additionally, reactor pressure was also affected by the steam flow to HPCI, RCIC and the reactor feedwater pump turbines. No safety relief valves actuated during this event nor were any safety relief valve actuations required.

At 1302 CDT the scram signal was reset and the plant was in a stable configuration.

D. CAUSE OF EVENT

Plant personnel (engineering and maintenance) investigated this event by interviewing involved personnel, reviewing operating information, reviewing vendor manuals and calibrating/checking plant equipment.

The root cause of the alternate Vital AC power loss is equipment failure. Specifically, plant maintenance personnel found that circuit breaker CB-4 (which is located between the Vital AC bus and its alternate power supply [600/120 VAC transformer]) had tripped in the open position.

Physical inspections of the circuit breaker noted that the control switch would trip when a force of 0.8 to 2.7 pounds was applied to it. Based on conversations with the breaker manufacturer, this breaker should require a force of 7 pounds to manually cause a trip. Also a small force applied to the back of the breaker casing could cause it to trip. The amount of force that would cause the breaker to open was so small that even minor vibrations or jarring could have caused the breaker to open.

At the time that the event occurred, plant electricians were working in the panel and in an electrical cubicle adjacent to the panel where CB-4 was mounted. When the electricians were questioned about their work activities, they stated that they were exercising all reasonable precautions to avoid inadvertent breaker operations. Additionally, they stated that they were not aware that the circuit breaker had opened.

Overcurrent testing of CB-4 using the multi-amp tester in the maintenance shop revealed no malfunctions of the breaker. Current measurements were taken with a clamp on ammeter on both legs of the Vital AC bus after power was restored. This test was to determine if an overcurrent condition existed that would trip the breaker. No overcurrent condition was found.

Based on the above information, plant personnel concluded: 1) an overcurrent trip of the breaker was not likely, 2) the circuit breaker trip mechanism is very sensitive to force or impact and the tripping mechanism is weak, and 3) the breaker is located where it is subject to inadvertent contact by personnel working in the area.

The anomalous operation of the HPCI system was also reviewed as part of the investigation of this event. Plant personnel concluded that the anomalous HPCI operation was the result of an air pocket in the HPCI pump suction instrument line. When maintenance personnel vented the instrument line, significant amounts of air were found. Several possibilities exist which could have caused the air to become entrapped, such as valve maintenance or the introduction of air during instrument calibration.

As previously stated, it was noted that HPCI had previously auto-transfered from its normal source of suction (the CST), to its alternate source of suction (the torus). This occurred at approximately 1018 CDT and was the result of a sensed high water level in the torus. This auto-transfer is part of the normal design.

A review of the SPDS output (from torus narrow range instruments) indicated that the auto-transfer occurred because of a sensed increase in the torus water level. The sensed torus water level reached a maximum height of 153 inches and the torus high level suction transfer setpoint is 152 inches. The HPCI torus suction valves began opening at the height of the sensed increase in torus level.

The plant design is such that there is no automatic transfer function for the HPCI suction back to the CST once it is aligned to the torus. As such, once the high level transfer occurred, the HPCI suction pathway remained aligned to the torus until the scram event occurred.

The SPDS review did not reveal any pump or valve operation which could be attributed as being the cause of a sensed torus water level swell. However, the review did show RCIC suction was transferred to the torus at 1005 CDT on 8/3/87, then realigned to the CST at 1028 CDT. The RCIC pump suction high torus level transfer switch is on the same instrument leg as one of the HPCI high torus level transfer switches, as well as torus level instruments 2T48-N010B and 2T48-N021B. Both 2T48-N010B and 2T48-N021B recorded a rise in level at 1018.

It is postulated that the RCIC suction transfer at 1005 was due to testing RCIC torus level switch 2E51-N062B per procedure 57SV-SUV-015-2S. When level switch 2E51-N062B was valved back into service, a spike in the instrument leg was experienced. This caused HPCI to transfer its suction to the torus at 1018 CDT on 8/3/87 on a sensed high water level condition. The RCIC suction was re-aligned to the CST at 1028 per procedure.

The transfer in suction sources from the CST to the torus is annunciated in the main control room to alert shift personnel of the off normal automatic suction pathway alignment. Operations personnel on shift noted that the HPCI Suppression Chamber High Level annunciator alarmed prior to the scram. However, they attributed it to I&C personnel performing procedure 57SV-SUV-015-2S, (HPCI/RCIC Pump Suction Source Instrument Functional Test and Calibration), which was in progress at that time.

This procedure checks both the torus and CST level switches for RCIC, but only checks the CST level switches on HPCI since the HPCI torus level switches are a different model. HPCI suction had previously been transferred to the torus during the CST level switch calibration, then realigned back to the CST per procedure.

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.

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.

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, it is concluded that this event had no adverse impact on nuclear plant safety.

F. CORRECTIVE ACTIONS

The corrective actions for this event included:

1. The Maintenance Department installed jumpers around CB-4 to allow removal of the breaker with the Vital AC bus energized. A subsequent investigation was conducted to determine the cause of the circuit breaker trip. The results of this investigation are presented in other sections of this LER. Additionally, a limited review for other like kind molded case circuit breakers was performed. No other circuit breakers of this type were found in the plant.

A replacement breaker for CB-4 was received on site. However, when it was inspected, it was found to be defective and was returned to the vendor. At the present time, the jumpers are still installed around breaker CB-4. Plant engineering personnel are evaluating leaving the circuit breaker out of the circuit.

2. Engineering has already issued non-safety related Design Change Request EM # 87-H-2-E-151 to design and install barrier boxes over the circuit breakers on the Vital AC inverter cabinets. These will prevent inadvertent contact with the breakers. Maintenance personnel have fabricated and installed the boxes.

3. The HPCI torus level transmitter which initiates the suction auto transfer function has since been calibrated and found to be in tolerance.

4. Since an air pocket was postulated to be the cause of the HPCI trip, plant personnel investigated the instrument lines and instruments. The HPCI and RCIC pump suction pressure transmitters on both units have been checked for air accumulation in the instrument lines. No air was found in the RCIC lines. However, air was found in the Unit 1 HPCI pump suction instrument and instrument line.

5. The plant Engineering Department completed an investigation concerning the root cause and required corrective actions to preclude any future entrapment of air into the HPCI instrument lines. The investigation concluded that the configuration of the instrument tubing, for the low pump suction pressure trip instrumentation on the Unit 1 and Unit 2 HPCI systems, can allow air to be entrapped in the instrument lines any time that the pump suction piping is drained and then refilled.

   To eliminate air entrapment in the tubing, these instrument lines should be vented through the pressure instruments. This should occur after filling the process piping, but prior to returning the system to service.

To prevent recurrence of the air entrainment into the instrument lines, plant procedures will be revised. Specifically, procedure 34SO-E41-001-1S, (High Pressure Coolant Injection (HPCI) System), and 34SO-E41-001-2S, (High Pressure Coolant Injection (HPCI) System), will be revised to require the HPCI pump suction instrument tubing to be vented through instruments 1E41-N019 and 1E41-N053 (for Unit 1) and 2E41-N019 and 2E41-N053 (for Unit 2). The venting will be required, as part of the procedures, any time the HPCI pump suction piping is drained and then refilled. It is anticipated that these procedures will be revised by approximately 3/1/88.

Plant engineering personnel determined that air entrapment into the Unit 1 and Unit 2 RCIC instrument lines is not a problem because the instrument tubing is sloped such that it is self venting.

Plant Instrument and Control (I&C) personnel investigated the spiking in the instrument leg of both the HPCI and RCIC high torus level transfer switches and the torus water level instruments. They concluded that the spiking is a rare occurrence and that no additional actions are required at this time.

6. Plant personnel verified that the Core Spray (CS EIIS Code BM) and the Low Pressure Coolant Injection (LPCI EIIS Code BO) mode of the Residual Heat Removal system do not have a low pressure pump suction trip. Therefore, these systems would not be susceptible to the condition noted on the HPCI system.

G. ADDITIONAL INFORMATION

1. FAILED COMPONENT(s) IDENTIFICATION

Plant Index Identifier: Not listed
Manufacturer: Heinemann
Model Number: CAT CJ2-G3-U
Type: Molded case circuit breaker 125 Amp magnetic only
EIIS: EE

2. PREVIOUS SIMILAR EVENTS

As of the date of this LER, there have been a total of 36 HPCI injections into the reactor vessel. In this event, there were 2 automatic injections.

There have been three previous events similar to the ones described in this LER. These events were reported by LERs 50-321/1985-010 Rev 1 (dated 1/16/85), 50-321/1987-011 (dated 7/23/87) and 50-366/1987-006 (dated 7/26/87).

These LERs describe events where Vital AC power was lost and a reactor scram occurred. In the case of the first LER, this event was caused by the tripping of an undervoltage relay. The other two LERs involved events where inverter failures caused the loss of Vital AC power.

The corrective actions for these events included: 1) resetting the scram and repairing/replacing some plant components, 2) refurbishing inverters, 3) adding temporary cooling to the inverter rooms, 4) developing permanent design changes to the inverter room ventilation, 5) developing a preventative maintenance procedure, 6) performing engineering evaluations for inverter replacement, and 7) replacing failed parts and scheduling follow up inspections.

The corrective actions for these events would not have prevented the event described by LER 50-366/1987-009 because the causes of the loss of the Vital AC were different. In the first case, Vital AC was lost because an undervoltage relay actuated. In the other cases, equipment failed due to excessive room temperatures.

Additionally, the corrective actions for LER 50-321/1987-011 and 50-366/1987-006 have not yet been completed. The events occurred within 12 days of the time of the event described in LER 50-366/1987-009.

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X7GJ17-H310

January 18, 1988

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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PLANT HATCH - UNIT 2
NRC DOCKET 50-366
OPERATING LICENSE NPF-5
LICENSEE EVENT REPORT
CIRCUIT BREAKER FAILS CAUSING POWER
FAILURE 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, revised, Licensee Event Report (LER) concerning an unanticipated actuation of an Engineered Safety Feature (ESF). The event occurred in August of 1987 at Plant Hatch - Unit 2.

Sincerely,

for L. T. Gucwa

LGB/lc

Enclosure: LER 50-366/1987-009 Rev 1

c: (see next page)

IE22
1/1

Georgia Power

c: Georgia Power Company
Mr. J. P. O'Reilly, Sr. Vice President - Nuclear Operations
Mr. J. T. Beckham, Jr., Vice President - Plant Hatch
GO-NORMS

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

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