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 1	0 5 0 0 0 3 2 1	1 OF 1 2

TITLE (4)
SPURIOUS GROUND FAULT TRIPS MAIN TURBINE AND GENERATOR 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 2	2 6	8 8	8 8	— 0 0 3	— 0 0	0 3	2 8	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)	1 0 0		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)
J. D. Heidt, Nuclear Licensing Manager - Hatch	4 0 4 5 2 6 - 4 3 5 0

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	J C	P T R	3 6 9	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 2/26/88 at approximately 0839 CST, Unit 1 was in the run mode at an approximate power level of 2436 MWt (approximately 100 percent of rated thermal power). At that time, a main generator field ground detection relay (EIIS Code EL) tripped. This resulted in a turbine trip which caused a scram signal to be inserted into the Reactor Protection System (RPS EIIS Code JC) logic. This was an unanticipated actuation of the RPS.

The root cause of this event (actuation of the ground fault detection relay) could not be conclusively determined. Plant personnel performed a detailed investigation. However, all equipment that was checked, functioned correctly. It is concluded the ground relay spuriously actuated.

Corrective actions for this event included: 1) cleaning, testing and inspecting electrical equipment, 2) inspecting and testing the field ground relay, 3) investigating and replacing pressure transmitters, 4) reviewing other utilities' operating experiences, and 5) scheduling evaluation of future actions to prevent the possibility of recurrence.

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FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
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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 an unplanned actuation of the Reactor Protection System (RPS EIIS Code JC) occurred. Additionally, the Low Low Set (LLS EIIS Code JE) logic actuated during this event. The LLS is an Engineered Safety Feature (ESF).

B. UNIT(s) STATUS AT TIME OF EVENT

1. Power Level/Operating Mode

Unit 1 was in steady state operation at an approximate power of 2436 MWt (100 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 the event.

C. DESCRIPTION OF EVENT

1. Event

On 2/26/88 at approximately 0839 CST, a main generator field ground detection relay (1N71-K731, EIIS Code EL) tripped. This caused a main generator trip which, in turn, resulted in the closure of the main Turbine Stop Valves (TSV EIIS Code TA).

The closure of the TSV is a scram signal input to the Reactor Protection System (EIIS Code JC). A full reactor scram occurred. As required by the plant's Emergency Operating Procedures (EOPs), licensed plant operations personnel inserted a manual scram signal into the RPS circuitry after the automatic scram signal occurred. Also, as a result of the pressure spike that occurred when the TSVs closed, a high reactor pressure signal was inserted into the RPS logic.

As a result of the TSV closure, reactor pressure increased and caused the voids in the core to collapse. This was an anticipated event. The collapse of the voids resulted in a decrease in sensed reactor water level. The lowest level reached in this event was approximately 15 inches above instrument zero (+15) (approximately 179 inches above the Top of Active Fuel - TAF).

The Reactor Feed Pumps (RFPs EIIS Code SJ) sensed the decrease in the reactor water level and automatically increased their injection flow rate. At 0840 CST, plant operations personnel manually tripped the A RFP. Reactor water level continued to increase because the B RFP was still injecting into the reactor vessel. At 0841 CST, reactor water level reached +58 inches and the B RFP tripped off automatically, per design, on an high reactor water level signal. The maximum water level reached in the event was approximately +70 inches.

After the B RFP tripped, reactor water level began to decrease. At approximately 0850 CST, licensed plant operations personnel manually restarted the B RFP. Plant operations personnel were able to maintain reactor water level within its normal range (+32 inches to +42 inches) by using the B RFP.

As a result of the closure of the TSV at the start of the transient, reactor pressure increased to approximately 1091 psig. The increase in the reactor pressure caused Safety Relief Valves (SRVs EIIS Code JE) A, F, G, H and L to lift. The C SRV (which should have opened concurrently with the other SRVs), opened approximately 3 seconds later when reactor pressure was decreasing to approximately 1062 psig. At this pressure, the G SRV closed.

The G SRV is one of the four SRVs that is assigned a Low Low Set (LLS) function. The other valves are the A, C, and H SRVs. The LLS SRVs are to remain open for longer periods of time and allow a more complete depressurization of the reactor vessel than the non-LLS SRVs.

As such, the G SRV should have remained opened until reactor pressure was approximately 887 psig. The C SRV (which is also a LLS SRV) closed at approximately 901 psig. It should have remained open until the reactor pressure was approximately 897 psig.

However, even though the C and G LLS SRVs closed prematurely, the remaining SRVs functioned correctly to relieve reactor pressure. At approximately 0840 CST, the A, C, and H SRVs closed.

Following the initial pressure transient, reactor pressure was controlled by operations personnel using the main Turbine Bypass Valves (TBV EIIS Code SO).

At approximately 0851 CST, the plant was stable and plant operations personnel reset the scram signal. No high pressure emergency core cooling systems were needed to maintain reactor water level in this event.

At approximately 0910 CST, plant operations personnel informed the NRC of the reactor scram, as required by 10 CFR 50.72.

At approximately 1100 CST, an event review team was established, in accordance with the plant' administrative guidelines, to investigate the cause of the reactor scram. Other plant personnel were also performing investigations (relative to the anomalous functioning of the LLS SRVs).

2. Dates/Times

Date	Time (CST)	Description
2/26/88	0839	The main generator tripped on ground fault detection signal. The TSVs closed and caused an automatic RPS actuation (scram).

Date	Time (CST)	Description
2/26/88	0839	Plant operations personnel inserted a manual scram signal into the RPS logic as required by the plant's EOPs.

Also, as a result of the pressure spike that occurred when the TSVs closed, a high reactor pressure signal was inserted into the RPS logic.

The following SRVs operated (all pressures from the Safety Parameter Display System [SPDS EIIS Code IQ]):

SRV #	Open Pressure (psig)	Close Pressure (psig)
L	1080	1027
F	1091	1027
A	1091	later
G	1091	1062
H	1091	later
C	1062	later

0840 Operations personnel manually tripped the A RFP.

The following SRVs closed:

SRV	Close Pressure
C	901
A	866
H	847

0841 Reactor water level reached approximately +58 inches and the B RFP tripped automatically, per design.

0850 Operations personnel manually start the B RFP to maintain reactor water level.

Date	Time (CST)	Description
2/26/88	0851	The plant was in a stable condition and plant operations personnel reset the scram signal.
	0910	Plant operations personnel notified the NRC of the reactor scram per the requirements of 10 CFR 50.72.
	1100	An event review team was established in accordance with plant administrative guidelines to investigate the event. Other plant personnel were also performing investigations (relative to the anomalous functioning of the LLS SRVs).

3. Other Systems Affected

The only systems affected by this event were the RPS and the LLS. The RPS functions to initiate protective actions to prevent damage to the principal safety barriers. The LLS system has no other secondary function other than ensuring that sufficient pressure reductions occur to prevent excessive SRV actuations and subsequent loadings on the SRV discharge piping and torus shell.

4. Method of Discovery

The scram occurred when the ground fault detection circuitry on the main generator actuated.

5. Operator Actions

Licensed Operations personnel performed the following action:

1. Responded to the automatic scram in accordance with emergency operating procedures and ensured that the plant was in a stable configuration.

2. Made applicable reports per the requirements of 10 CFR 50.72.

Non-licensed engineering and maintenance personnel performed the following actions:

1. Investigated the cause of the generator ground fault detection trip.

2. Investigated the cause of the anomalous SRV LLS operation.

6. Auto/Manual Safety System Response

The only safety systems that actuated in this event were the RPS, SRVs and LLS. All of these systems automatically responded to the event.

D. CAUSE OF EVENT

1. Immediate Cause

The immediate cause of the scram was actuation of the main generator field ground fault detection relay (1N71-K731) and the subsequent turbine trip.

The immediate cause of the anomalous LLS SRV operation was: 1) the failure of the LLS logic to arm for the G SRV, and 2) slow response of the logic for the C LLS SRV.

2. Root/Intermediate Cause

The root cause of the actuation of the generator field ground fault detection relay could not be conclusively identified.

Plant Engineering and Maintenance personnel performed systematic troubleshooting in an effort to determine the cause of the relay actuation. This detailed investigation included the following:

1. Meggering the generator field while: 1) the main turbine/generator was turned by the turning gear, and 2) the main turbine/generator was turned at rated speed with no load. All readings were acceptable.

2. Meggering all generator excitation circuitry. All readings were satisfactory.

3. Calibrating the field ground detection relay, 1N71-K731. The relay was found to be within calibration limits.

4. Visually inspecting the rectifier, excitation cabinets, generator field bus work, and generator field instrumentation. The inspection did not reveal any gross abnormalities which could have caused a ground fault path.

5. Reviewing operating experience from another utility (Fitzpatrick Nuclear Plant LER 1987-012). This review indicated that the teflon cooling water tubes to the exciter rectifier banks could potentially have caused the event. The tubes were inspected. Several of these tubes were found to be dirty and had low resistance readings. However, the resistance readings were not sufficiently low so as to be the most probable cause of the event.

   The review of operating experience also indicated that the inside of the insulators (teflon tubes) may be coated with a copper oxide compound resulting from system corrosion and wear. The insulators at Plant Hatch were not disassembled for inspection because no replacement insulator tubes were available and the tubes are very fragile.

The root cause of the failure of the G SRV LLS logic to arm was the failure of pressure transmitter 1B21-N122D. The cause of the slow operation of the C SRV was the failure of its pressure transmitter, 1B21-N120C. At this time, these failures are believed to be unrelated. However, the transmitters are being returned to the manufacturer (Rosemount) for further analysis.

E. ANALYSIS OF EVENT

The turbine stop valve closure scram anticipates the pressure, neutron flux, and heat flux increases that could result from rapid closure of the turbine stop valves. Closure of the turbine stop valves with the reactor at power can result in a significant addition of positive reactivity to the core as the reactor pressure rise collapses steam voids. The turbine stop valve closure scram initiates a scram earlier than either the neutron monitoring system or the reactor high pressure scrams. With a scram trip setting of less than or equal to 10 percent of valve closure from full open, the scram limits the fuel thermal overpower to well within acceptable limits.

Although either the reactor high neutron flux or high pressure scram, in conjunction with the pressure relief system, is adequate to preclude overpressurizing the nuclear system, the turbine stop valve closure position scram provides additional margin to the reactor pressure limit.

The LLS relief logic system is designed to mitigate the thrust loads on the Safety Relief Valve Discharge Lines (SRVDLs) and the resulting loads on the torus shell from subsequent SRV actuations during small and intermediate break Loss of Coolant Accidents (LOCAs). This can be accomplished by extending the time between SRV actuations such that the actuation times are long enough to allow the SRVDL water leg to return to its normal level.

The water leg in the SRVDL will rise due to the condensation of the blown down steam in the SRVDL after the SRV is closed. If no further SRV actuations occur, the water leg will then decrease as water flows, under gravity, back to the suppression pool (torus). The LLS system ensures that subsequent SRV actuations occur after the water leg in the SRVDL stabilizes, at its normal level, by increasing the blowdown range and decreasing the closing and opening setpoints for the four LLS SRVs.

The LLS design involves four non-ADS SRVs. The LLS control logic operates the four valves through arming and actuation. The arming function requires concurrent signals of any SRV operating and a high reactor vessel pressure. Once the LLS control logic is armed, it then causes the four LLS valves to actuate. The LLS valves actuate at pressures lower than their respective relief settings and remain open longer (allowing a greater blowdown over a pressure range of approximately 140 psi).

The failures of reactor vessel pressure transmitters 1B21-N122D and 1B21-N120 C caused: 1) the LLS logic of SRV G to not arm, and 2) the sluggish operation of the C SRV LLS logic. Since the LLS logic of the G SRV did not arm, the G SRV responded (closed) per its normal setpoints and not the LLS logic setpoints. Since the C SRV logic was operating sluggishly, the C SRV closed slightly prematurely.

The possibility of two of the four LLS valves failing to properly actuate has already been analyzed as part of the Unit 1 Final Safety Analysis Report (FSAR) event scenarios. The analysis assumed, as an event scenario, that a small break LOCA occurred concurrent with an early isolation of the reactor vessel (due to a Loss of Off Site Power [LOSP] condition).

In this scenario (assuming that two of the LLS SRVs become inoperable in the lowered setpoint relief mode after the initial blowdown), the remaining two LLS valves can reduce reactor pressure sufficiently such that no non-LLS SRVs will re-actuate at their pressure setpoints. Pressure is controlled during the remainder of the event by one of the two remaining LLS SRVs.

Based on the above information, it is concluded that this event had no adverse impact on nuclear plant safety. Additionally, since this event occurred at approximately 100 percent of rated thermal power, it is not believed that the consequences of the event would have been more severe at other power conditions.

F. CORRECTIVE ACTIONS

Corrective actions for this event included:

1. Cleaning, testing, and inspecting electrical equipment associated with the generator excitation system.

2. Inspecting and testing generator field ground detection relay, 1N71-K731.

3. Investigating the operation of LLS pressure transmitters 1B21-N122D and 1B21-N120C. The transmitters were replaced by plant personnel on 2/27/88. The failed transmitters were returned to the vendor for further analysis.

4. Reviewing other utilities' operating experiences.

5. Having plant engineering personnel evaluate the following actions for possible future implementation:

   a. Removing the generator field ground trip function from the turbine generator unit and replacing it with an alarm function.

   b. Installing a generator ground fault recorder to determine the locations of any future generator faults.

   c. Reviewing preventive maintenance procedures to determine any areas where preventive maintenance may be improved. The improvements would be designed to prevent recurrence of these types of events.

   d. Removing and inspecting the rectifier bank teflon cooling water tubes on both Units 1 and 2. This equipment would either be cleaned and re-installed or replaced, depending on the inspection results.

G. ADDITIONAL INFORMATION

1. FAILED COMPONENT(s) IDENTIFICATION

MPL (Plant Index Identifier): 1B21-N122 D and 1B21-N120 C
Manufacturer: Rosemount
Model Number: 1153-GP8PAN0019
Type: Pressure Transmitter Type GP
EIIS: JC

2. PREVIOUS SIMILAR EVENTS

Two previous LERs reported events similar to the one reported in this LER. These LERs were: 50-321/1987-001 (dated 1/1/87) and 50-321/1987-002 (dated 1/15/87).

These LERs describe events where reactor scrams occurred as a result of grounding problems. In LER 50-321/1987-001, the event was the result of either a ground fault on a battery system (the main turbine electro-hydraulic control system receives electrical power from a station battery system) or from welding activities in the vicinity of the electro-hydraulic control system.

The event described in LER 50-321/1987-002 was caused by a trip of a ground fault detector. It was determined that a loose wire and a highly conductive film on the outside of the wire's insulation caused an electrical ground path which actuated the ground fault detector.

The corrective actions for these events included: 1) performing an engineering evaluation of the event, 2) testing the main turbine backup overspeed circuitry, 3) testing the main turbine trip circuitry, 4) cleaning and tightening generator components, 5) meggering the main generator field, and 6) formulating recommendations for additional preventative maintenance.

The corrective actions for these events would not have prevented the event described by LER 50-321/1988-003 because no conclusive cause for LER 50-321/1988-003 could be determined. The testing that was performed (meggering, calibrating, and visually inspecting), did not detect any problem that would have caused the main generator to trip.

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March 28, 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
SPURIOUS GROUND FAULT TRIPS MAIN TURBINE
AND GENERATOR 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 an unanticipated actuation of some Engineered Safety Features (ESFs). This event occurred at Plant Hatch - Unit 1.

Sincerely,

L. T. Gucwa

LGB/lc

Enclosure: LER 50-321/1988-003

c: (see next page)

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Georgia Power

c: Georgia Power Company
Mr. J. T. Beckham, Jr., Vice President - Plant Hatch
GO-NORMS

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