Georgia Power Company
40 Inverness Center Parkway
Post Office Box 1295
Birmingham, Alabama 35201
Telephone 205 877-7279

J. T. Beckham, Jr.
Vice President—Nuclear
Hatch Project

Georgia Power
the southern electric system

HL-1531
001343

March 15, 1991

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
COMPONENT FAILURE IN GENERATOR EXCITER
CAUSES TURBINE TRIP AND REACTOR SCRAM

Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(i) and (iv), Georgia Power Company is submitting the enclosed Licensee Event Report (LER) concerning a condition that was prohibited by the plant Technical Specifications and the unanticipated actuation of some Engineered Safety Features (ESFs). This event occurred at Plant Hatch - Unit 2.

Sincerely,

J. T. Beckham, Jr.

SWR/ct

Enclosure: LER 50-366/1991-004

c: (See next page.)

IE22
1/1

Georgia Power

c: <u>Georgia Power Company</u>
Mr. H. L. Sumner, General Manager - Nuclear Plant
Mr. J. D. Heidt, Manager Engineering and Licensing - Hatch
NORMS

<u>U.S. Nuclear Regulatory Commission, Washington, D.C.</u>
Mr. K. Jabbour, Licensing Project Manager - Hatch

<u>U.S. Nuclear Regulatory Commission, Region II</u>
Mr. S. D. Ebneter, Regional Administrator
Mr. L. D. Wert, Senior Resident Inspector - Hatch

001343

NRC Form 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104 EXPIRES: 4/30/92

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 8

TITLE (4)
COMPONENT FAILURE IN GENERATOR EXCITER CAUSES TURBINE TRIP AND REACTOR SCRAM

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQ NUM	REV	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
										0 5 0 0 0
0 2	1 4	9 1	9 1	0 0 4	0 0	0 3	1 5	9 1		0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL: 9 8

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR (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)
[ ] 20.405(a)(1)(iii)	[X] 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	912	367-7851

COMPLETE ONE LINE FOR EACH FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORT TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORT TO NPRDS	
X	J E	R V	T 0 2 0	Y E S		X	E L	R G	G 0 8 0	Y E S	

SUPPLEMENTAL REPORT EXPECTED (14)

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

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

ABSTRACT (16)

On 2/14/91, at approximately 1854 CST, Unit 2 was in the Run mode at an approximate power level of 2388 CMWT (approximately 97.5% rated thermal power). At that time, a voltage regulator card in the exciter system for the main generator failed, resulting in main generator and main turbine trips. The unit scrammed on turbine stop valve closure. Reactor water level decreased to approximately 2 inches below instrument zero and was restored to normal by the Reactor Feedwater Pumps and the Reactor Core Isolation Cooling system. Reactor vessel steam dome pressure briefly peaked at approximately 1116 psig and the Bypass valves successfully controlled pressure. No Safety Relief Valves opened. The pressure experienced in the main steam lines was transitory and apparently less than reactor vessel steam dome pressure. However, using steam dome pressure, the SRVs were possibly not in compliance with their +1% Technical Specifications tolerance requirement. The valves were removed and tested in accordance with code requirements. The actual drift experienced was within the bounds of a previous analysis demonstrating adequate overpressure protection.

The root cause of the scram was component failure. The cause of the SRV setpoint drift was corrosion-induced bonding of the surface between the pilot valve disc and seat.

Corrective actions for this event included replacing the failed component in the generator exciter system and removing and refurbishing the SRVs. Georgia Power Company (GPC) will ntinue to participate in the BWR Owners' Group action plan to resolve setpoint arift.

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

I. SCRAM DUE TO COMPONENT FAILURE

On 2/14/91, at approximately 1854 CST, Unit 2 was in the Run mode at an approximate power level of 2388 CMWT (approximately 97.5% rated thermal power). At that time, a voltage regulator card in the exciter system for the main generator failed, resulting in main generator and main turbine trips. The unit scrammed on turbine stop valve closure. Reactor water level decreased to approximately 2 inches below instrument zero and was restored to normal by the Reactor Feedwater Pumps (RFPs, EIIS Code SJ) and the Reactor Core Isolation Cooling system (RCIC, EIIS Code BN). Reactor vessel steam dome pressure briefly peaked at approximately 1116 psig and the Bypass valves (BPVs, EIIS Code SO) successfully controlled pressure.

The root cause of the scram was component failure.

Corrective actions included replacing the failed component in the generator exciter system.

II. SAFETY RELIEF VALVE (SRV) SETPOINT DRIFT

During the pressure transient following the scram, no Safety Relief Valves (SRVs, EIIS Code RV) opened. The pressure experienced in the main steam lines was transitory and apparently less than reactor vessel steam dome pressure. However, using steam dome pressure, the SRVs were possibly not in compliance with their $\pm 1\%$ Technical Specifications tolerance requirement. The valves were removed and tested in accordance with code requirements. The actual drift experienced was within the bounds of a previous analysis demonstrating adequate overpressure protection.

The cause of the SRV setpoint drift was corrosion-induced bonding of the surface between the pilot valve disc and seat.

Corrective actions included removing and refurbishing the SRVs. Georgia Power Company (GPC) will continue to participate in the BWR Owners' Group action plan to resolve setpoint drift.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (5) YEAR	SEQ NUM	REV	PAGE (3)
PLANT HATCH, UNIT 2	05000366	91	004	00	3 OF 3

TEXT

DESCRIPTION OF EVENT

I. SCRAM DUE TO COMPONENT FAILURE

On 2/14/91, at approximately 1854 CST, Unit 2 was in normal operation near full power when a voltage regulator card in the exciter system for the main generator failed. This resulted in an increase in generator field excitation which was detected by the main generator differential current auxiliary lockout relay, per design. The trip of this relay resulted in main generator and main turbine trips per the protective logic design. The turbine trip occurred at greater than 30% rated thermal power, causing a Reactor Protection System (RPS, EIIS Code JC) actuation and a scram per design. The reactor recirculation pumps (EIIS Code AD) also tripped due to the turbine trip. Immediately after the scram, reactor water level decreased due to collapsing voids, and a Group 2 Primary Containment Isolation System (PCIS, EIIS Code JM) isolation was received on low reactor water level. All Group 2 Primary Containment Isolation Valves (PCIVs) closed as required.

Following the closure of the turbine stop valves the BPVs opened to control pressure. Reactor vessel steam dome pressure peaked at approximately 1116 psig about 7 seconds following the scram, and decreased steadily thereafter. The BPVs continued to control reactor pressure throughout the transient.

Reactor water level was initially restored to normal and maintained by the RFPs. However, during the first minutes after the trip, a pressure regulator malfunction in the turbine steam seal system permitted steam seal pressure to decrease. This resulted in the intrusion of noncondensible gases into the main condenser, allowing condenser vacuum to slowly decrease. By approximately 1859 CST, condenser vacuum reached approximately 22.5 inches of mercury, and the RFPs tripped per design. Licensed operators then manually started the RCIC system to control reactor water level. When the operators restored flow to the steam seal system by bypassing the malfunctioning regulator valve, the RFP turbines were reset. Control of reactor water level with the RFPs was resumed by approximately 1914 CST. The minimum reactor water level observed during the transient was approximately 2 inches below instrument zero, which is about 162 inches above the top of active fuel. By approximately 1919 CST, with reactor water level being controlled in a band normal for the shutdown condition, the scram signal was reset.

II. SRV SETPOINT DRIFT

During the pressure transient following the scram, no SRVs lifted. Based upon the conservative assumption that the pressure experienced by the SRVs was equivalent to reactor vessel steam dome pressure, the SRVs were possibly not in compliance with the $\pm 1\%$ tolerance allowed by Unit 2 Technical

TEXT

Specifications Section 3.4.2.1. Therefore, the SRVs were removed from the main steam lines and sent to an independent laboratory for testing in accordance with the ASME Boiler and Pressure Vessel Code, Section XI. A tabulation of the as-received mechanical lift setpoints for the SRVs is shown below:

MPL Number	Serial Number	Nominal Lift Pressure (psig)	Initial Lift Pressure (psig)	% Over Nominal Lift Pressure
2B21-F013A	315	1100	1185	7.7
2B21-F013B	312	1090	1146	5.1
2B21-F013C	1004	1090	1144	5.0
2B21-F013D	1001	1100	1122	2.0
2B21-F013E	303	1110	1219	9.8
2B21-F013F	310	1090	1152	5.7
2B21-F013G	314	1090	1100	1.0
2B21-F013H	308	1110	*1323	*19.2
2B21-F013K	1008	1100	1150	4.5
2B21-F013L	307	1110	1217	9.6
2B21-F013M	301	1100	1151	4.6

* This SRV did not lift during the normal scope of as-received testing in accordance with Section XI. Maximum pressure applied was 1323 psig before testing was terminated.

The average level of SRV setpoint drift experienced in this event is on the high end of utility data on SRV drift collected by the Boiling Water Reactor Owners' Group (BWROG) over the last several years. However, on an individual valve basis this magnitude of drift has been seen before and is therefore not entirely unexpected. The referenced SRV data has been compiled by the BWROG in its ongoing efforts to address the issue of setpoint drift by eliminating corrosion-induced bonding as a contributor. Finally, it should be noted that one of the SRVs in this event did demonstrate a lift setpoint during the testing which was below the peak reactor vessel steam dome pressure of 1116 psig. This appears to confirm that the SRVs did not experience sufficient steam line pressure to lift during the pressure transient following the scram.

CAUSE OF THE EVENT

I. SCRAM DUE TO COMPONENT FAILURE

The root cause of the scram is component failure. Specifically, a voltage regulator card in the main generator field excitation system failed. This led to an increase in excitation resulting in a trip of the differential current auxiliary lockout relay. This relay is sensitive to, among other things, the relationship between exciter voltage and generator frequency, and thus tripped due to over excitation. As a result the main turbine tripped and the reactor scrammed on turbine stop valve closure.

TEXT

II. SRV SETPOINT DRIFT

The cause of the SRV setpoint drift is corrosion-induced bonding between the SRV pilot valve disc and seat. Georgia Power Company (GPC)is participating in the BWROG action plan to resolve the SRV setpoint drift issue which has been concurred with by the NRC.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

I. SCRAM DUE TO COMPONENT FAILURE

This report is required per 10 CFR 50.73(a)(2)(iv) because an unplanned actuation of the RPS and Engineered Safety Features (ESFs) occurred. Specifically, the RPS actuated per design on turbine stop valve closure when the main turbine tripped. The RPS again actuated on low reactor water level when collapsing voids caused water level to decrease below the scram setpoint. The Group 2 PCIS, an ESF, received an isolation signal on low reactor water level. The Group 2 PCIVs closed per design.

The RPS automatically initiates a reactor scram to ensure the radioactive materials barriers, such as fuel cladding and the pressure system boundary, are maintained and to mitigate the consequences of transients and accidents. Closure of the turbine stop valves, such as occurs on a main turbine trip, can result in the addition of positive reactivity to the core as the resultant reactor pressure increase collapses voids. Therefore, turbine stop valve closure initiates a scram prior to high neutron flux or high reactor pressure signals to provide a satisfactory margin to core thermal-hydraulic limits. The high-pressure scram, in conjunction with the pressure relief system, is adequate to preclude overpressurizing the pressure system boundary; however, the turbine stop valve closure scram provides additional margin.

In this event, the turbine stop valves closed on a main turbine trip. The RPS actuated on turbine stop valve closure, per design. Reactor water level decreased as expected due to void collapse. The RFPs responded to limit the drop in water level and restore level to its normal range. At no time was water level less than 162 inches above the top of the active fuel. Reactor water level was maintained with both the RFPs and RCIC; no Emergency Core Cooling Systems were needed to recover and/or maintain water level.

II. SRV SETPOINT DRIFT

This report also is required per 10 CFR 50.73(a)(2)(i) because a condition prohibited by the plant's Technical Specifications potentially existed. Specifically, based on the conservative assumption that the pressure experienced by the SRVs was equivalent to reactor vessel steam dome pressure, some of the SRVs were potentially not in compliance with the +1% tolerance requirement of Unit 2 Technical Specifications Section 3.4.2.1. The apparent SRV setpoint drifts were in the range of 0.5 to 2.4%.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (5) YEAR	SEQ NUM	REV	PAGE (3)
PLANT HATCH, UNIT 2	05000366	91	004	00	6 OF 8

TEXT

The purpose of the SRVs is to provide overpressurization protection for the reactor vessel and attached reactor coolant system piping. Eleven SRVs are located in the main steam lines. The SRVs are manufactured in accordance with the requirements of ASME Boiler and Pressure Vessel Code Section III for pilot-operated valves. There are three sets of valves: four valves with a nominal lift setpoint of 1090 psig, four valves with a nominal lift setpoint of 1100 psig, and three valves with a nominal lift setpoint of 1110 psig. The size of the SRVs, in conjunction with their nominal lift setpoints, is intended to limit the most severe pressure transient to +110% of the reactor vessel design pressure of 1250 psig, or a maximum of 1375 psig.

Reactor vessel steam dome pressure only reached a peak pressure of 1116 psig due to the opening of the BPVs. However, based on the conservative assumption noted earlier, it appeared that for 8 of the 11 SRVs (those with setpoints at 1090 and 1100 psig) their tolerances were outside the 1% Technical Specifications requirement. Therefore, testing, in accordance with Section XI of the Code, was conducted to determine the actual setpoint drift exhibited by all the SRVs and to assure compliance with the 1% tolerance requirement for restart of Unit 2.

Ten of the eleven SRVs exhibited setpoint drift in excess of the +1% Technical Specifications tolerance requirement. Nine exhibited drift in excess of the +3% tolerance specified in Section XI. However, the as-received setpoint for 1 SRV was below the peak reactor vessel steam dome pressure. This indicates the pressure actually experienced by the SRVs was probably less than dome pressure. The differences in pressure between the reactor vessel and the main steam lines are expected and have been demonstrated in a previous analysis of the turbine trip event for Plant Hatch.

The setpoint drift magnitudes ranged from 1.0 to 9.8%, with the exception of one valve, 2B21-F013H, which would not lift during the normal scope of Section XI testing. Special testing was performed to determine its actual drift magnitude. However, the scope of testing was limited by the desire to maximize the preservation of the corrosion on the pilot valve disc/seat assembly for metallurgical analysis. Therefore, the actual magnitude of the drift could not be determined. It is probable that this valve effectively would not have opened in its self-actuating mode. However, it is also likely it would have been possible to open the valve manually during an event where its actuation was warranted. Additionally, it should be noted that, based on BWROG SRV drift data, there have been other valves which would not open during the normal scope of Section XI testing. This is the first time additional testing has been done to determine the actual magnitude of the drift.

A previous plant-specific analysis performed for Plant Hatch by General Electric Company demonstrates that Plant Hatch has sufficient margin for

TEXT

overpressure protection and that analysis bounds the SRV drift magnitudes seen in this event. Specifically, the analysis evaluated the peak vessel pressure at various setpoint drifts up to 200-psi drift for the plant's most limiting pressurization event, the MSIV closure-flux scram event. If it was conservatively assumed that all 11 SRVs opened at a lift pressure +9% over the nominal lift pressure, the resulting pressure transient would be limited to approximately 1300 psig, which is less than the design limit of 1375 psig.

Based on the previous discussion, it is concluded this event had no adverse impact on nuclear safety. The analysis is conservative in that it assumes worst-case initial conditions and is therefore applicable to all power levels.

CORRECTIVE ACTION

I. SCRAM DUE TO COMPONENT FAILURE

The main generator exciter system was repaired.

The failed electronic voltage regulator card will be sent to General Electric Company for analysis of the failure mode.

II. SRV SETPOINT DRIFT

All eleven SRVs were removed, bench tested, refurbished, and reinstalled.

GPC will continue to be an active participant in the BWROG corrective action plan to resolve the SRV setpoint drift issue which has been concurred with by the NRC. The BWROG action plan consists of two parallel options. The primary BWROG option consists of controlling the local environment in the SRV valve cavity to mitigate corrosion. A catalyst design will be developed based on European experience that indicates that catalysts appear to reduce oxygen induced corrosion. It is expected that it will take about two cycles of inservice experience in selected SRVs at various BWRs to determine the effectiveness of the catalyst design. The BWROG backup option, being developed in parallel, is a safety-grade system of externally powered pressure switches to assure opening of the SRVs pneumatically when needed. This option will be available for implementation on a plant specific basis should the inservice experience with the catalyst indicate it does not resolve the issue.

As part of GPC's active participation in this BWROG program the pilot valve disc/seat assembly from the SRV which would not lift during normal Section XI testing has been supplied to General Electric for metallurgical examination of the corrosion. Significant results, if any, from this examination will be factored into the BWROG action plan. Additionally, GPC will review the results to determine the necessity for any further actions.

TEXT

ADDITIONAL INFORMATION

1. Other Affected Plant Systems:

   No plant systems other than those mentioned in this report were affected by this event.

2. Previous Similar Events:

   No events occurring in the past two years were identified in which a malfunction in generator excitation resulted in a reactor scram.

   Events reported in the past two years in which SRVs were found to have experienced mechanical lift setpoint drift were reported in the following LERs:

   50-321/1990-005, dated 04/24/90
   50-321/1991-004, dated 03/14/91
   50-366/1989-007 Rev 1, dated 2/7/91

   Corrective actions for those events included refurbishing the SRVs to bring the lift pressures within a $\pm 1\%$ tolerance and continuing to participate in BWROG efforts to resolve the issue of SRV setpoint drift. Those actions would not have prevented this event because the causes and corrective actions for SRV setpoint drift have not yet been fully resolved by the industry.

3. Failed Component Information:

Master Parts List Number:	1B21-F013A, B, C, E, F, H, K, L, M
Manufacturer:	Target Rock Company
Type:	Two Stage Safety Relief Valve
Model Number:	7567F
Manufacturer Code:	T020
EIIS System Code:	JE
EIIS Component Code:	RV
Root Cause Code:	B
Reportable to NPRDS:	Yes

Master Parts List Number:	2N51
Manufacturer:	General Electric
Type:	Voltage Regulator
Model Number:	44C238493-G01
Manufacturer Code:	G080
EIIS System Code:	EL
EIIS Component Code:	RG
Root Cause Code:	X
Reportable to NPRDS:	Yes