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-1917
002192

November 6, 1991

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

PLANT E. I. HATCH - UNIT 1
NRC DOCKET 50-321
OPERATING LICENSE DPR-57
LICENSEE EVENT REPORT
SAFETY RELIEF VALVE SETPOINTS REMAIN
WITHIN IN-SERVICE TESTING TOLERANCE REQUIREMENTS

Gentlemen:

Georgia Power Company is submitting the enclosed voluntary Licensee Event Report (LER) due to the potential industry interest in the event. This event occurred at Plant Hatch - Unit 1.

Sincerely,

J. T. Beckham, Jr.

SRB/CLT/et

Enclosure: LER 50-321/1991-022

cc: (See next page.)

IE22
1/1

Georgia Power

cc: Georgia Power Company
Mr. H. L. Sumner, General Manager - Nuclear Plant
NORMS

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

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

002192

700775

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 1	05000321	1 OF 4

TITLE (4)
SAFETY RELIEF VALVE SETPOINTS REMAIN WITHIN IN-SERVICE TESTING TOLERANCE REQUIREMENTS

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)
										05000
10	09	91	91	022	00	11	06	91		05000

OPERATING MODE (9): 5
POWER LEVEL: 000

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR (11)

20.402(b)	20.405(c)	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)
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	AREA CODE	TELEPHONE NUMBER
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

SUPPLEMENTAL REPORT EXPECTED (14)

		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (16)

On 10/9/91, at 1430 CDT, Unit 1 was in its thirteenth refueling outage with the mode switch in Refuel and fuel removed from the reactor vessel. All eleven Main Steam System (EIIS Code SB) Safety Relief Valves (SRVs) had been sent offsite for testing of their mechanical lift setpoints as required by American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, IWV-3512. At that time, engineering personnel were notified by the offsite testing agency that tests showed the mechanical lift setpoints for all eleven SRVs were within the Section XI tolerance requirement of ± 3 percent.

This report is being submitted voluntarily due to the potential industry interest in view of the ongoing Boiling Water Reactor Owners Group (BWROG) activities to address SRV mechanical lift setpoint drift. The cause of the setpoint drift is corrosion-induced bonding of the SRV pilot valve disc and seat. In this event, the SRV's all lifted within their setpoint tolerance of ± 3 percent; however, this report is being submitted to apprise the industry of the results.

TEXT

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor
Energy Industry Identification System codes are identified in the text as (EIIS Code XX).

DESCRIPTION OF EVENT

On 10/9/91, at 1430 CDT, Unit 1 was in its thirteenth refueling outage with the mode switch in Refuel and fuel removed from the reactor vessel. All eleven Main Steam System (EIIS Code SB) Safety Relief Valves (SRVs) had previously been sent offsite for testing of their mechanical lift setpoints as required by American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, IWV-3512. On 10/9/91 engineering personnel were notified by the offsite testing agency that tests showed the mechanical lift setpoints for all eleven SRVs were within the in-service testing (IST) tolerance requirement of ± 3 percent. The test results were as follows:

PLANT HATCH MPL	PILOT CARTRIDGE S/N	NAMEPLATE SET PRESSURE (PSIG)	INITIAL LIFT PRESSURE (PSIG)	VARIANCE FROM NAMEPLATE PRESSURE (%)
1B21-F013A	1009	1080	1075	- 0.5
1B21-F013B	306	1100	1107	+ 0.6
1B21-F013C	313	1100	1097	- 0.3
1B21-F013D	302	1090	1079	- 1.0
1B21-F013E	1005	1080	1067	- 1.2
1B21-F013F	1187	1090	1103	+ 1.2
1B21-F013G	1010	1080	1086	+ 0.6
1B21-F013H	1007	1090	1090	0
1B21-F013J	1190	1100	1117	+ 1.5
1B21-F013K	1006	1090	1077	- 1.2
1B21-F013L	1011	1080	1104	+ 2.2

This report is being submitted voluntarily due to the potential industry interest in view of the ongoing Boiling Water Reactor Owners Group (BWROG) activities to address the issue of SRV mechanical lift setpoint drift. The cause of the setpoint drift is corrosion-induced bonding of the SRV pilot valve disc and seat. The bonding phenomenon is random and has in the past resulted in some of the Plant Hatch SRVs exceeding the tolerance requirement of ± 3 percent during Section XI testing.

NRC Form 366A (6-89) U.S. NUCLEAR REGULATORY COMMISSION APPROVED OMB NO 3150-0104 EXPIRES: 4/30/92

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (5) YEAR	SEQ NUM	REV	PAGE (3)
PLANT HATCH, UNIT 1	0 5 0 0 0 3 2 1	9 1	0 2 2	0 0	3 OF 4

TEXT

In this event none of the SRV drift magnitudes exceeded ± 2.2 percent. During the approximately eight months the SRVs had been in service, after being refurbished in February 1991, two automatic reactor scrams occurred on 8/9/91 and 9/11/91 in which ten of the eleven SRVs opened which may have contributed to the low drift magnitude experienced during the subject Section XI testing. These events were reported in LER 50-321/91-13, dated 9/3/91 and LER 50-321/91-17, dated 10/9/91. In both of the referenced events SRV 1B21-F013J was the one SRV that did not open. However, review of the reactor pressure transient peaks experienced following both scrams confirmed the SRV was not required to actuate.

CAUSE OF EVENT

This is a voluntary report which is being submitted because of potential industry interest in view of the ongoing BWROG activities regarding SRV setpoint drift. In this event, all eleven SRVs actuated within their Section XI tolerance requirement of ± 3 percent. Therefore, no causal analysis with respect to failures was required.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This report is being submitted voluntarily because the event may be of potential interest to the industry in view of the ongoing efforts of the BWROG to address the issue of SRV setpoint drift.

The purpose of the SRVs is to provide over-pressure protection for the reactor pressure vessel and associated reactor coolant system piping. There is a total of eleven SRVs located in the main steam lines between the reactor pressure vessel and the Main Steam Isolation Valves. The SRVs are manufactured by Target Rock Company in compliance with the requirements of ASME Section III (1968 with Winter 1968 addenda), Paragraph N911.4(a)(1) for pilot operated valves. There are three sets of valves: four valves are designed to open at 1080 psig, four at 1090 psig, and three are designed to open at 1100 psig. The size of the valves coupled with the designated lift pressures is intended to limit a vessel pressure transient to +110% of the reactor vessel design pressure of 1250 psig, or a maximum of 1375 psig.

During the testing, all eleven SRVs actuated within their Section XI tolerance requirement of ± 3 percent. The SRV drift magnitudes experienced in this event are well within the bounds of drift magnitudes considered in a previous plant specific analysis which demonstrated that Plant Hatch has sufficient margin for over-pressure protection.

Based on the above information, it is concluded that this event had no adverse impact on nuclear safety. This assessment applies to all power levels.

CORRECTIVE ACTIONS

Actions taken or planned for this event include:

- o Refurbishing the SRVs to bring lift pressures within a $\pm$ 1 percent tolerance.

- o GPC will continue to participate in the BWROG corrective action plan to resolve the SRV setpoint drift issue. The BWROG action plan which has been concurred with by the NRC consists of two parallel options. The primary BWROG option consists of controlling the chemistry of the local environment in the SRV valve cavity to mitigate corrosion. A catalyst design is being 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 in-service experience in selected SRVs at various BWRs to determine the effectiveness of the catalyst design. The second option, being developed in parallel, is a system of externally powered pressure switches to assure opening of the SRVs pneumatically when needed. This option is presently planned for implementation at Plant Hatch Unit Two during the Fall 1992 refueling outage and Unit One during the Spring 1993 refueling outage.

ADDITIONAL INFORMATION

No systems other than the Unit 1 SRVs were affected by this event.

There have been events within the last two years in which SRVs experienced setpoint drift in excess of the $\pm$ 3 percent Section XI tolerance requirement as reported in the following LERs:

50-321/90-005, dated 4/24/90
50-321/91-004, dated 3/14/91
50-366/89-007, Rev. 1, dated 2/7/91
50-366/91-004, dated 3/15/91
50-366/91-009, dated 5/8/91

There were no failed components associated with this event.