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)	DOCKET NUMBER (2)	PAGE (3)
PLANT HATCH, UNIT 1	0 5 0 0 0 3 2 1	1 OF 0 5

TITLE (4): LOOSE GEAR FASTENER IN FLOW CONTROLLER RESULTS IN INOPERABILITY OF HPCI

EVENT DATE (5) Month	Day	Year	LER NUMBER (6) Year	Sequential Number	Revision Number	REPORT DATE (7) Month	Day	Year	OTHER FACILITIES INVOLVED (8) Facility Names	Docket Number(s)
0 8	2 6	8 8	8 8	— 0 1 2 —	0 0	0 9	2 0	8 8		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)	[ ] 50.73(a)(2)(iv)	[ ] 73.71(b)
[ ] 20.405(a)(1)(i)	[ ] 50.36(c)(1)	[X] 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	Number
Steven B. Tipps, Manager Nuclear Safety and Compliance, Hatch	9 1 2	3 6 7 - 7 8 5 1

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	B J	F F C	G 0 8 0	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 08/26/88, at approximately 1515 CDT, Unit 1 was in the run mode at an approximate power level of 2436 CMWT (approximately 100 percent of rated thermal power). At that time, the High Pressure Coolant Injection (HPCI, EIIS Code BJ) system was declared inoperable due to a malfunctioning system flow controller. The controller was subsequently repaired and returned to service. At approximately 2120 CDT, the HPCI system was declared operable following successful completion of procedure 34SV-E41-002-1S, HPCI Pump Operability.

The cause of the flow controller malfunction was a loose fastener on an intermediate gear in the gearing mechanism of the controller. When the fastener became loose, the intermediate gear became unmeshed with the adjacent gearing. This caused a break in the gear train preventing adjustment of the controller setting.

Corrective actions include placement of the gear into the gear train and tightening of the fastener. To prevent recurrence, procedures will be revised to require a periodic check of the gear train and fasteners.

IE22
1/1

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

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

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

On 08/26/88, at approximately 1515 CDT, Unit 1 was in the run mode at an approximate power level of 2436 CMWT (approximately 100 percent of rated thermal power). At that time, the High Pressure Coolant Injection (HPCI, EIIS Code BJ) system was declared inoperable due to a malfunctioning system flow controller. The controller was subsequently repaired and returned to service. At approximately 2120 CDT, the HPCI system was declared operable following successful completion of procedure 34SV-E41-002-1S, HPCI Pump Operability.

Description of Event

On 08/26/88, at approximately 1515 CDT, routine surveillance testing of the HPCI pump in accordance with procedure 34SV-E41-002-1S had been satisfactorily completed. At that time, a licensed plant operator was restoring the system to the standby mode and was unable to adjust the setting of the flow controller to the system design flowrate as required by the procedure. Inability to restore the controller setting to the system design flowrate would have prevented the HPCI pump from achieving the design flowrate during an automatic initiation of the system. The HPCI pump was subsequently declared inoperable at 1515 CDT.

At the time of the event, the Residual Heat Removal/Low Pressure Coolant Injection (RHR/LPCI, EIIS Code BO) system 'B' pump was tagged out of service for preventive maintenance. As required by Technical Specification Section 3.5.D.3, when the HPCI system and a loop of the RHR system LPCI mode are inoperable, "an orderly shutdown shall be initiated and the reactor vessel pressure shall be reduced to 150 psig or less within 24 hours." Consequently, at 1515 CDT, procedure 34GO-OPS-013-1S, Normal Plant Shutdown, was initiated.

At approximately 1715 CDT, preventive maintenance on the LPCI 'B' pump was complete and the pump was declared operable following satisfactory completion of procedure 34SV-E11-001-1S, Residual Heat Removal Pump Operability. At approximately 1730 CDT, Unit 1 was in full compliance with the Limiting Condition of Operation of Technical Specification Section 3.5.D.2 and normal plant shutdown activities were terminated.

Following the discovery of the flow controller malfunction, maintenance personnel began investigating the problem. At approximately 1925 CDT, the controller had been repaired and returned to service and operability testing of the HPCI system per 34SV-E41-002-1S was in progress. On 08/26/88, at approximately 2120 CDT, following successful completion of procedure 34SV-E41-002-1S, the HPCI system was declared operable.

Cause of Event

The immediate cause of the event was a loose fastener in the gear train of the HPCI system flow controller. The loose fastener allowed an intermediate gear in the gear train to become unmeshed from adjacent gearing. The break in the gear train prevented adjustment of the controller setting. Thus, the controller could not be adjusted to the required system design flowrate and placed in the automatic mode.

The root cause of the fastener loosening could not be conclusively determined. No previous similar controller failures have occurred at Plant Hatch, and a review of the Nuclear Plant Reliability Data System (NPRDS) records also indicates no history of similar failures at other plants. The mechanism for holding the fastener in place was a star washer for which no preventative maintenance was required by the controller's vender manual to preclude fastener loosening. A review of the maintenance history for the controller did not reveal any probable causes for the fastener loosening. This appears to be an isolated incident as a result of normal operation of the controller.

Reportability Analysis and Safety Assessment

This report is required per 10 CFR 50.73 (a)(2)(v) because a component failure occurred in the HPCI system which would have prevented the system from functioning as designed to mitigate the consequences of an accident. The failure would have prevented the HPCI system from reaching design flow upon automatic initiation of the system.

The HPCI system is provided to assure that the reactor is adequately cooled to limit fuel-clad temperature in the event of a small break in the nuclear system and a loss of coolant which does not result in rapid depressurization of the reactor vessel. The Automatic Depressurization System (ADS, EIIS Code JE) is a backup for the HPCI system. Upon ADS initiation, the reactor is depressurized to a point where either the LPCI system or the Core Spray (CS, EIIS CODE BM) system can operate to maintain adequate core cooling.

In the event addressed in this report, the HPCI system and one of the four LPCI pumps were inoperable. Three of the four LPCI pumps and their associated equipment, ADS, and both loops of CS were operable. Based upon the Unit 1 Final Safety Analysis Report (FSAR), either loop of the CS system or the LPCI system, consisting of two operable pumps in one RHR loop and at least one operable pump in the other RHR loop, can supply sufficient cooling to the reactor for any rupture of the nuclear safety boundary up to and including the Design Basis Accident (DBA). Based on this information, it is concluded that this event had no adverse impact on nuclear plant safety.

Since the reactor was operating at approximately 100 percent power at the time of the event, the event would not have been more severe had it occurred under other operating conditions.

Corrective Action

Procedures 57CP-CAL-044-1/2S, GE Type 547-01 Self Synchronizing M/A Transfer Station, will be permanently revised by 01/15/89 to require a check of the controller gearing and gearing fasteners during calibration of the controller. Also, procedures 52PM-E41-003-1/2S, HPCI System Maintenance, will be permanently revised by 01/15/89 to require a check of the controller gearing and gearing fastener on a twelve month frequency. In the interim, the controllers of this type will be visually inspected, by the Maintenance Department, during the upcoming Unit One refueling outage.

Additional Information

No systems other than the HPCI system were directly affected by the component failure.

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

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

The component failure is considered to be an isolated incident in that no previous component failures were caused by a loose fastener in the component gear train. However, a HPCI flow controller malfunction was reported in LER 50-366/1987-017, dated 12/15/87, in which the HPCI flow controller failed to properly control HPCI system flowrate in the automatic mode. Corrective actions for the event included replacing an amplifier card and repairing a defective solder joint in a support module for the controller. These corrective actions would not have prevented the event addressed in this report since they did not involve the gearing of the flow controller.

Failed Component Identification:

MPL (Plant Index Identifier): 1E41-R612
Manufacturer: General Electric Company
Model Number: 547-01
Type: Self-synchronizing Basic Control Station
EIIS Code: FFC

Georgia Power Company
333 Piedmont Avenue
Atlanta, Georgia 30308
Telephone 404 526-6526

Mailing Address:
Post Office Box 4545
Atlanta, Georgia 30302

W. G. Hairston, III
Senior Vice President
Nuclear Operations

the southern electric system

HL-78
0191t
X7GJ17-H310

September 20, 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
LOOSE GEAR FASTENER IN FLOW CONTROLLER
RESULTS IN INOPERABILITY OF HPCI

Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(v), Georgia Power Company is submitting the enclosed Licensee Event Report (LER) concerning a condition that could have prevented an Engineered Safety Feature (ESF) from fully performing its safety function. This event occurred at Plant Hatch - Unit 1.

Sincerely,

W. G. Hairston, III

W. G. Hairston, III

SB/ct

Enclosure: LER 50-321/1988-012

c: (see next page)

IE22
1/1

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

0191t