Georgia Power Company
333 Piedmont Avenue
Atlanta, Georgia 30308
Telephone 404 526-3195

Mailing Address:
40 Inverness Center Parkway
Post Office Box 1295
Birmingham, Alabama 35201
Telephone 205 868-5581

the southern electric system

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

HL-1103
000543

May 14, 1990

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
PERSONNEL ERROR RESULTS IN
INADEQUATE TECHNICAL SPECIFICATIONS SURVEILLANCE

Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(i), Georgia Power Company is submitting the enclosed Licensee Event Report (LER) concerning a miswired thermocouple which resulted in an inadequate Technical Specifications surveillance. This event occurred at Plant Hatch - Unit 1.

Sincerely,

W. G. Hairston, III

SR/ct

Enclosure: LER 50-321/1990-006

c: (See next page.)

9005220173 900514
PDR ADOCK 05000321
S PDC

IE22
1/1

c: Georgia Power Company
Mr. H. C. Nix, General Manager - Nuclear Plant
Mr. J. D. Heidt, Manager Engineering and Licensing - 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
Mr. S. D. Ebneter, Regional Administrator
Senior Resident Inspector - Hatch

000543

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)
PERSONNEL ERROR RESULTS IN INADEQUATE TECHNICAL SPECIFICATIONS SURVEILLANCE

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 4	1 8	9 0	9 0	0 0 6	0 0	0 5	1 4	9 0		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 5
POWER LEVEL (10): 0 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)	[ ] 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)	[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	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

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 4/18/90, at approximately 1430 CDT, Unit 1 was in the Refuel mode at an approximate power level of 0 CMWt (approximately 0% of rated thermal power). At that time, plant engineering personnel discovered that a wiring error existed in a junction box (EIIS Code JBX) leading to a strip chart recorder (EIIS Code IM) used for measuring the reactor vessel bottom head drain temperature. This temperature reading is used to comply with Unit 1 Technical Specifications section 3/4.6.E which establishes a maximum limit on the temperature differential between the upper and lower regions of the reactor vessel prior to starting a recirculation pump (EIIS Code AD). This requirement could not be met with the wiring error. Subsequent investigation determined no violation of reactor vessel thermal stress protection occurred as a result of this event.

The causes of the event are personnel error and a less than adequate functional test following maintenance. Specifically, the error occurred during the 1987 Unit 1 refueling outage when work was performed on a junction box common to various reactor vessel thermocouples (EIIS Code TE). Two thermocouples were incorrectly wired during that work and the functional test which followed failed to detect the error.

Corrective action for this event included correcting the wiring error, counselling the maintenance personnel responsible for performing the work in 1987, and verifying that no violation of thermal stress limits had occurred. In addition, the event will be discussed in a memorandum addressed to maintenance scheduling supervisors from the Manager, Outages and Planning, stressing the importance of adequate post-maintenance functional testing.

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 4/18/90, at approximately 1430 CDT, Unit 1 was in the Refuel mode at an approximate power level of 0 CMWt (approximately 0% of rated thermal power). At that time, plant engineering personnel discovered that a wiring error existed in a junction box (EIIS Code JBX) leading to a strip chart recorder (EIIS Code IM) used for measuring the reactor vessel bottom head drain temperature. This temperature reading is used to comply with Unit 1 Technical Specifications section 3/4.6.E which establishes a maximum limit on the temperature differential between the upper and lower regions of the reactor vessel upon starting a recirculation pump (EIIS Code AD). This requirement could not be met with the wiring error. Subsequent investigation determined no violation of reactor vessel thermal stress protection occurred as a result of this event.

The causes of the event are personnel error and a less than adequate functional test following maintenance. Specifically, the error occurred during the 1987 Unit 1 refueling outage when work was performed on a junction box common to various reactor vessel thermocouples (EIIS Code TE). Two thermocouples were incorrectly wired during that work and the functional test which followed failed to detect the error.

Corrective action for this event included correcting the wiring error, counselling the maintenance personnel responsible for performing the work in 1987, and verifying that no violation of thermal stress limits had occurred. In addition, the event will be discussed in a memorandum addressed to maintenance scheduling supervisors from the Manager, Outages and Planning, stressing the importance of adequate post-maintenance functional testing.

DESCRIPTION OF EVENT

On 4/18/90 plant engineering personnel working in electrical junction box 1B21-J002, discovered that conductors from the thermocouple located on the vessel bottom head drain, 1G31-N042, were not connected as indicated on design drawings. Further investigation revealed that the display point on recorder 1B21-R606 which was labeled "Vessel Bottom Head Drain" was actually connected to a different thermocouple, 1B21-N030A2, which measures the temperature of the reactor vessel flange. The thermocouple located on the vessel bottom head drain was not connected to the recorder. The specific location of the wiring error was in common junction box 1B21-J002.

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	9 0	— 0 0 6 —	0 0	0 3 OF 0 5

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

The temperature of the reactor coolant at the bottom of the vessel is compared with that of the coolant in the upper regions of the vessel prior to startup of the recirculation pumps. There is a maximum limit on the temperature differential which can exist to avoid potentially excessive thermal stresses on the vessel during recirculation pump startup. (Based on the determination that the wiring error occurred in 1987, review of available, conservative vessel temperatures later confirmed that no violation of thermal stress limits occurred.)

Investigation into the cause of the wiring error included researching the maintenance history of all equipment related to the affected recorder and researching the data output history of the affected recorder. It was co[illegible]d that the wiring error occurred during the 1987 Unit 1 refueling outage when maintenance, which involved lifting and replacing conductors in junction box 1B[illegible]J02, was performed on a vessel flange thermocouple, 1B21-N030A2.

CAUSE OF THE EVENT

The causes of the event are personnel error and a less than adequate functional test following maintenance. Specifically, the error occurred during the 1987 Unit 1 refueling outage when work was performed on a junction box common to various reactor vessel thermocouples (EIIS Code TE). Two thermocouples were incorrectly wired during that work and the functional test which followed failed to detect the error.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This event is reportable per 10 CFR 50.73(a)(2)(i)(B) because an event occurred in which Unit 1 entered a condition prohibited by the plant's Technical Specifications. Specifically, a surveillance required by Technical Specification section 3/4.6.E could not be adequately performed prior to starting recirculation pumps due to a miswired recorder.

The purpose of recorder 1B21-R606 is to continuously record the temperatures of various points on the reactor vessel, thus assuring the availability of information used to protect the reactor pressure vessel from thermal stresses which could be induced by starting a recirculation pump when excessive thermal stratification of the reactor coolant exists. Such stratification may form when the recirculation pumps have been deenergized, and flow into the vessel bottom head area has continued via the control rod drives (CRDs, EIIS Code AA). The introduction of cooler CRD water into the lower vessel region in the absence of driving or mixing flow from the recirculation pumps permits cooler CRD water to accumulate in the bottom of the vessel. Should the stratification permit the temperature differential between the upper and lower regions of the vessel to exceed 145°F, the thermal stresses on the vessel and vessel internals resulting from starting a recirculation pump could possibly exceed those allowed by the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III. Therefore, Unit 1 Technical Specifications section 3/4.6.E requires operators to verify that the temperature differential between the steam dome and vessel bottom head drain is less than 145°F prior to starting a recirculation pump and to make a permanent record of this verification.

In the event addressed in this report it was discovered that the above mentioned surveillance requirement had not been adequately met due to an instrumentation wiring error. The control room recorder point labeled "Vessel Bottom Head Drain" was actually connected to and reading vessel flange temperature.

In order to assess the possible consequences of the event, the architect/engineer (A/E) was consulted for information on other sources of vessel bottom head temperature. The A/E made the following two observations:

1) The effects of reactor coolant stratification as described in the basis for Unit 1 Technical Specification section 3/4.6 occur only when a recirculation pump has been tripped and restarted and the reactor is not in a Cold Shutdown condition (known as a "hot start"). It is not possible to have coolant stratification leading to a temperature differential of greater than 145°F when the reactor is in Cold Shutdown.

2) A thermocouple reading bottom head skin temperature is available and is considered to be a conservative (i.e., lower than actual) estimate of reactor coolant temperature in the vessel bottom head region. This is because the outer surface of the reactor is likely to be cooler than the inside due to the radiation of thermal energy away from the outer surface. Thus, comparing vessel bottom head skin temperature to the steam dome temperature results in a greater temperature differential than actually exists between the coolant in the bottom head and the steam dome.

Using these assumptions, personnel from the Engineering Support Department researched all recirculation system "hot starts" which have occurred since the 1987 refueling outage. In no case was the differential between the steam dome and the vessel bottom head skin temperature found to exceed 145°F. Therefore, even though the required surveillance was not adequately performed, thermal stress protection limits for the reactor vessel for recirculation system "hot starts" were not violated.

Based on the above analysis, it is concluded that the subject event had no adverse impact on nuclear safety. Because the "hot start" of the reactor recirculation system is assumed to be the most severe recovery possible from thermal stratification, the analysis is applicable to all recirculation pump start conditions.

CORRECTIVE ACTIONS

Corrective actions for this event included:

1) Correcting the wiring error and performing a functional test of the circuit which positively verified that the affected control room recorder actually measures the indicated parameter. This action is complete.

2) Counselling the maintenance personnel involved in performing the work in 1987 on junction box 1B21-J002.

3) Verifying that no reactor recirculation system "hot starts" which have occurred since the introduction of the wiring error in 1987 have been performed under conditions wherein a differential of greater than 145°F existed between the steam dome and the vessel bottom head skin temperatures. This action has been completed. "Hot starts" occurring since that time have shown no evidence of violation of thermal stress limits.

4) Issuing a memorandum from the Manager, Outages and Planning describing the event and stressing the need for adequate post-maintenance functional tests as a preventive measure to preclude such events from recurring. This action will be complete by 5/31/90.

ADDITIONAL INFORMATION

1. Previous Similar Events:

An event was reported in LER 50-321/1989-007, dated 6/16/89, in which the installation of a miswired radiation monitor followed by an inadequate post-installation functional test led to a Technical Specifications required instrument being inoperable. Corrective actions for that event included notifying an equipment vendor of his wiring error, rewiring the equipment correctly, calibrating affected equipment, counseling an individual, and including the event in the Engineering Continuing Training program. The event which is the subject of this report occurred prior to the event reported in LER 50-321/1989-007. Therefore, corrective actions for that event would not have been prevented this event.

2. Failed Components Identification:

No failed components contributed to this event.

3. Other Affected Equipment:

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