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J. T. Beckham, Jr.
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Georgia Power
the southern electric system

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Docket No. 50-366

HL-5090

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Edwin I. Hatch Nuclear Plant - Unit 2
Licensee Event Report
Component Failure Results in
Engineered Safety Feature System Actuation

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 the actuation of certain primary containment isolation valves which resulted from component failure.

Sincerely,

J T Beckham Jr

J. T. Beckham, Jr.

JKB/ld

Enclosure: LER 50-366/1995-010

cc: *Georgia Power Company*
Mr. H. L. Sumner, Nuclear Plant General Manager
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. B. L. Holbrook, Senior Resident Inspector - Hatch

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NRC FORM 366 (5-92)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4)
Component Failure Results in Engineered Safety Feature System Actuation

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 NAME	DOCKET NUMBER(S)
12	06	95	95	010	00	01	02	96		05000
									FACILITY NAME	05000

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)	100			
	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 (include area code)
Steven B. Tipps , Nuclear Safety & Compliance Manager, Hatch	AREA CODE 912 367-7851

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	KD	FI	T172	Yes					

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 15 single-space typewritten lines) (16)

On 12/6/95 at 0840 EST, Unit 2 was in the Run mode at a power level of 2558 CMWT (100% rated thermal power). At that time, personnel were placing the hydrogen water chemistry (HWC) system into service when a partial Group 1 Primary Containment Isolation System (PCIS) isolation signal on Main Steam Line (MSL) high radiation was received. Primary Containment Isolation Valves 2B31-F019 and 2B31-F020 closed, and the steam packing exhauster tripped per design. Personnel tripped the HWC system, and MSL radiation levels decreased to pre-event levels. Personnel reset the isolation signal, reopened valves 2B31-F019 and 2B31-F020, and returned the steam packing exhauster to service by 0845 EST.

This event was caused by component failure. An internal soldered connection in hydrogen flow rate monitor 2P73-N025 failed, causing the flow rate output signal to go to zero. The flow rate monitor provides feedback to the controller for the hydrogen flow control valve. Thus, the controller continued to increase the injection rate, resulting in a higher hydrogen flow rate and higher MSL radiation levels than expected. Levels exceeded the trip setpoints, and an isolation signal was generated per design.

Corrective actions for this event included replacing the flow rate monitor and verifying the flow rate monitor and the flow control valve controller were functioning properly.

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 12/6/95, Unit 2 was in the Run mode at a power level of 2558 CMWT (100% rated thermal power). Plant personnel were placing the hydrogen water chemistry (HWC) system (EIIS Code KD) into service following completion of the Fall 1995 refueling outage. The initial hydrogen flow rate was to be 5 scfm. The Main Steam Line (MSL) radiation monitor (EIIS Code IL) trip setpoints were based upon the radiation levels expected for a flow rate of 5 scfm. The normal flow rate is approximately 40 scfm; however, an initial flow rate of 5 scfm was being used as a part of planned incore stress corrosion monitoring testing.

At 0840 EST, a partial Group 1 Primary Containment Isolation System (PCIS, EIIS Code JM) isolation signal on MSL high radiation was received as personnel increased the controller signal for the hydrogen flow control valve from 0 scfm to 5 scfm per system operating procedure 34SO-P73-001-2S, "Hydrogen & Oxygen Injection and Control for HWC." Primary Containment Isolation Valves (PCIVs, EIIS Code JM) 2B31-F019 and 2B31-F020 closed, and the steam packing exhauster (EIIS Code TC) tripped per design. No other actuations occurred or were required to occur from this isolation signal.

Personnel tripped the HWC system, and MSL radiation levels decreased to below the high radiation trip setpoints within 4 minutes of receipt of the isolation signal. Operations personnel reset the partial Group 1 PCIS isolation signal. PCIVs 2B31-F019 and 2B31-F020 were reopened, and the steam packing exhauster was returned to service by 0845 EST.

CAUSE OF EVENT

This event was caused by component failure. A soldered connection internal to the hydrogen flow rate monitor failed, causing the flow rate output signal to go to zero.

The hydrogen flow rate monitor provides a feedback signal to the controller for the hydrogen flow control valve. Because the flow output from the monitor was zero, the feedback signal to the controller indicated no hydrogen flow. Thus, the controller continued to drive the flow control valve open in an attempt to increase the hydrogen flow rate. This resulted in the flow control valve opening too far and a hydrogen flow rate of approximately 17 to 20 scfm. The normal flow rate is approximately 40 scfm; however, an initial flow rate of 5 scfm was being used as part of planned incore stress corrosion monitoring testing. Since the MSL radiation monitor trip setpoints had been set for the radiation levels expected for a hydrogen flow rate of 5 scfm, actual radiation levels exceeded the trip setpoints, and a partial Group 1 PCIS isolation signal on MSL high radiation was generated per design.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This report is required by 10 CFR 50.73(a)(2)(iv) because of the unplanned actuation of the partial Group 1 PCIS on MSL high radiation. Specifically, Group 1 PCIVs 2B31-F019 and 2B31-F020 automatically closed per design when the actual hydrogen flow rate was greater than planned, resulting in higher than expected MSL radiation levels. The Group 1 PCIS is an engineered safety feature system.

The HWC system mitigates reactor water chemical conditions associated with intergranular stress corrosion cracking (IGSCC). Boiling water reactors use high purity water as the primary coolant in the direct cycle production of steam. Because of radiolytic decomposition, reactor water contains a minute steady-state concentration of dissolved oxygen, which is sufficient to initiate IGSCC of sensitized stainless steel. The HWC system injects hydrogen into the feedwater at the suctions of the condensate booster pumps (EIIS Code SD) to mitigate IGSSC in the recirculation piping and the reactor vessel. The injected hydrogen reduces the dissolved oxygen and lowers the radiolytic production of the hydrogen and oxygen within the vessel.

Operation of the HWC system substantially increases the carryover of Nitrogen-16 from the reactor to the steam system (EIIS Code SB). When the HWC system is in operation, the MSL radiation setpoints are required to be increased because of the increase in MSL radiation due to Nitrogen-16 carryover.

In this event, the actual hydrogen flow rate exceeded the planned flow rate of 5 scfm. Because of the higher hydrogen flow, MSL radiation levels increased due to increased Nitrogen-16 carryover. Because the MSL radiation monitor trip setpoints were set for a lower hydrogen flow rate and a corresponding lower radiation level, a partial Group 1 PCIS isolation signal on MSL high radiation was achieved. All automatic actions occurred as designed. Because the steady-state hydrogen flow rate typically is 35-45 scfm, a flow rate of 17-20 scfm did not result in abnormally high radiation levels and, therefore, did not adversely affect plant personnel or the health and safety of the public.

Based upon this assessment, it is concluded that this event had no adverse impact on safety. This assessment is applicable to all power levels.

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

CORRECTIVE ACTIONS

The hydrogen flow rate monitor was replaced, and the new monitor and flow control valve controller were verified to be functioning properly. The HWC system was successfully placed into service on 12/8/95 at 1101 EST at an initial hydrogen flow rate of 5 scfm.

ADDITIONAL INFORMATION

No systems other than those mentioned in this report were involved in this event.

Failed Component Information:

MPL Number: 2P73-N025
Manufacturer: Thermal Instruments
Model Number: 600DL
Type: Flow Meter
Manufacturer Code: T172

EIIS System Code: KD
Reportable to NPRDS: Yes
Root Cause Code: X
EIIS Component Code: FI

Within the last 2 years, one similar event in which an unplanned Group 1 PCIS actuation was the result of component failure has occurred. This event was reported in Licensee Event Report 50-366/1994-001, dated 2/2/94. In the previous event, a failed relay coil resulted in actuation of Group 1 PCIS logic on loss of power and isolation of some Group 1 PCIVs per design. Corrective actions for the previous event would not have prevented this event, because the failed components, their locations and functions, and their failure mechanisms were completely different and unrelated. Consequently, none of the corrective actions for the previous event could have affected the hydrogen flow monitor.