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-2954
004114

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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
ELECTRO-HYDRAULIC SYSTEM LEAK
PROMPTS MANUAL SCRAM WITH SCHEDULED
SHUTDOWN IN PROGRESS

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 an Electro-Hydraulic Control System leak which prompted a manual scram during a scheduled shutdown.

Sincerely,

J. T. Beckham, Jr.

J. T. Beckham, Jr.

OCV/cr

Enclosure: LER 50-366/1992-015

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

NRC Form 366 (6-03) U.S. NUCLEAR REGULATORY COMMISSION APPROVED OMB NO. 3150-0104 EXPIRES: 4/30/92

LICENSEE EVENT REPORT (LER)

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TITLE (4)
ELECTRO-HYDRAULIC CONTROL SYSTEM LEAK PROMPTS MANUAL SCRAM WITH SCHEDULED SHUTDOWN IN PROGRESS

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 9	1 6	9 2	9 2	0 1 5	0 0	1 0	1 4	9 2		0 5 0 0 0

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR (11)			
POWER LEVEL	0 2 0	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)	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
B	T G	P S P	G 0 8 2	Y E S						

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (16)

On 9/16/92, at 0242 CDT, Unit 2 was in the Run mode at 487 CMWT (20 percent of rated thermal power), and in the process of shutting down for the tenth refueling outage. Reactor vessel pressure was being controlled by the Main Turbine Bypass valves (BPVs). At that time, a manual scram was initiated as a conservative action due to a leak discovered in the Electro-Hydraulic system control fluid piping that could have affected operation of the BPVs. Earlier, at 0220 CDT, a leak in the EHC piping between the BPVs and a system hydraulic fluid actuator was found during a walkdown of the Condenser Bay area. The EHC reservoir was then checked and found to be 4.5 inches below normal. Due to a concern that the BPVs could drift closed on low EHC system pressure and cause a pressure transient in the reactor vessel, a decision was made to initiate a manual scram. Consequently, at 0242 CDT, a manual scram was initiated. All control rods fully inserted as designed. Reactor water level decreased to a minimum of 27.5 inches above instrument zero (186 inches above the top of the active fuel) subsequent to the scram. At this level, no Engineered Safety Feature systems were required to actuate. Reactor water level was restored by the 2B Reactor Feedwater pump within one minute of the scram. Prior to the event, reactor pressure was approximately 920 psig and remained constant throughout the event, as pressure was controlled by the BPVs.

The cause of the event was through-wall cracking in a high pressure EHC system pipe.

Corrective actions include repairing the cracked portion of the pipe prior to restart from the current refueling outage.

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 9/16/92, at 0242 CDT, Unit 2 was in the Run mode at 487 CMWT (20 percent of rated thermal power). A controlled shutdown to enter the tenth refueling outage was in progress. Reactor vessel pressure was being controlled by the Main Turbine Bypass system (EIIS Code TA) valves (BPVs). At that time, a manual scram was initiated due to a leak being discovered in the Electro-Hydraulic (EIIS Code TG) control system fluid piping. Earlier, at 0220 CDT, the leak was found during a walkdown of the Condenser Bay area. The leak was located in EHC piping between the BPV hydraulic actuators and a system hydraulic fluid accumulator. The Main Turbine had been previously taken off line and turbine overspeed testing was in progress at the time of the discovery. Shortly after discovering the leak, the overspeed test was terminated and the Main Turbine was tripped. Further investigation of the condition showed that the leak could not be isolated and the oil level in the EHC reservoir was reaching its low level alarm point. Due to a concern that the BPVs could drift closed on low EHC system pressure and result in a pressure transient in the reactor vessel, a decision was made to initiate a manual scram.

Consequently, at 0242 CDT, a manual scram was initiated. All control rods fully inserted as designed. Reactor water level decreased to a minimum of 27.5 inches above instrument zero (186 inches above the top of the active fuel) subsequent to the scram. Normal water level is 37 inches above instrument zero. At the level reached in this transient, no Engineered Safety Feature systems were required to actuate. Reactor water level was restored by the 2B Reactor Feedwater system (EIIS Code SK) pump within one minute of the scram. Prior to the event, reactor pressure was approximately 920 psig and remained constant throughout the event, as pressure was controlled by the BPVs.

CAUSE OF EVENT

The cause of the event was through-wall cracking of an EHC system high pressure hydraulic line located between a system hydraulic fluid accumulator and the BPV hydraulic actuators. It appears that a less than adequate socket weld created a stress riser in the pipe wall adjacent to the weld. This pipe is normally subjected to a low amplitude, high frequency vibration, which is acceptable in the absence of such stress risers. However, this vibration in combination with the stress riser apparently resulted in fatigue failure of the pipe wall. The resulting crack caused a loss of approximately 67 gallons of hydraulic fluid. Due to the magnitude of the leak, the continued functioning of the BPVs was in question. Therefore, to preclude the possibility of the BPVs drifting closed and of the ensuing pressure transient in the reactor vessel, a manual scram was initiated.

TEXT

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This report is required pursuant to 10 CFR 50.73(a)(2)(iv) because an ESF system was manually initiated in order to preclude a reactor vessel pressure transient and the subsequent ESF actuations required to mitigate the high pressure condition. Specifically, with the EHC system leaking, the BPVs would have eventually drifted closed. Closing of the BPVs would have resulted in a reactor vessel pressure transient, further resulting in an automatic scram on high reactor pressure. In order to preclude this from happening, a manual scram was initiated.

The Main Turbine Bypass system is designed to dissipate up to 25 percent of the rated core thermal power during reactor heatup, while the turbine is being brought up to speed, during power operation when the reactor system steam generation exceeds the transient turbine steam requirements and limitations, and during cooldown of the reactor. The system consists of three sequentially operated regulating valves (i.e., the BPVs) mounted in a single valve manifold. The manifold and associated piping interconnect the Main Steam lines immediately upstream of the Turbine Stop Valves and the lines to the Main Condenser, providing a vent path for the Main Steam lines directly to the Main Condenser, bypassing the Main Turbine. The BPVs are signaled to open whenever the amount of steam generated by the reactor cannot be entirely absorbed by the Main Turbine either because of the operating condition of the plant or because of a transient condition in which the Main Turbine is not capable of dissipating all of the steam generated by the reactor.

In this event, a crack in a high pressure EHC system line jeopardized the operation of the BPVs. The leak was discovered during a walkdown in the Condenser Bay area and appropriate actions were taken before operation of the system was affected. Specifically, the Main Turbine was tripped and a manual scram of the reactor was initiated. All control rods fully inserted as designed. Following the scram, reactor water level decreased to a minimum of 186 inches above the top of the active fuel before being restored by the Reactor Feedwater system (normal water level is at 196 inches above the top of the active fuel). The reactor water level did not exceed any automatic ESF actuation setpoints during the level transient and, thus, no ESF systems were required to operate. Since the EHC system was still functional at this point, the BPVs operated as designed and controlled reactor pressure. Consequently, the reactor pressure which was at 920 psig prior to the event remained constant during the event. In summary, conservative actions were taken by licensed personnel to mitigate the consequences of the condition, and the plant functioned as designed resulting in the reactor being brought to a shutdown condition.

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LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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TEXT

Had the leak not been discovered, the EHC reservoir would have continued to decrease resulting in the low level annunciator alarming in the Main Control Room. Per the annunciator response procedure, the operator would have tripped the Main Turbine had the Turbine Control Valves began to drift closed. It is possible that the Main Turbine would have automatically tripped on low EHC pressure prior to this point. With the reactor at low power (less 30 percent of rated thermal power), the reactor scram signals on Turbine Stop Valve closure and on Turbine Control Valve fast closure are bypassed. Consequently, in this condition, when the Main Turbine tripped, a scram would not have initiated. It is possible that the BPVs at this point might have drifted closed. With the reactor in operation (at a relatively low power) and the BPVs closed, pressure in the reactor would have increased to the high pressure scram setpoint of 1054 psig, resulting in an automatic scram. The Safety Relief Valves (EIIS Code SB) would have cycled open to control reactor pressure.

Based upon the Final Safety Analysis Report, the peak pressure resulting from such an event would remain well within the design limit of 1375 psig for the reactor vessel bottom head. The peak cladding temperature would remain at a relatively low value and no significant reduction in the Minimum Critical Power Ratio would be experienced. Consequently, in a low power condition, had the leak gone undetected, the ensuing scram and transient would not have jeopardized nuclear safety.

Had the event occurred at a high power (greater than or equal to 30 percent of rated thermal power), the Main Turbine trip would have resulted in an automatic scram on Turbine Stop Valve closure. The Safety Relief valves would have functioned to control reactor pressure. The peak reactor pressure expected would be 1208 psig, well below the design limit of 1375 psig for the reactor vessel bottom head. The increase in the temperature of the fuel cladding would be minimal.

The most severe transient associated with BPV failure is the full-power generator load rejection. In such an event, the load rejection is sensed and a Turbine Control Valve fast closure (which occurs more rapidly than a Turbine Stop Valve closure) is initiated immediately in order to prevent an overspeed of the Main Turbine. Fast closure of the valves results in a simultaneous scram and a sudden pressure increase in the reactor pressure vessel. With the failure of the Main Turbine Bypass system, no steam bypass to the Main Condenser is available resulting in a more severe pressure transient. The Safety Relief Valves cycle open to mitigate the pressure transient. The peak pressure reached during the event would be 1207 psig at the reactor vessel bottom, significantly lower than the design limit of 1375 psig. The peak fuel cladding temperature would remain well within limits. Consequently, no fuel failure would be experienced.

Based on the above information, it is concluded that this event had no adverse impact on nuclear safety. This analysis envelopes all operating conditions.

CORRECTIVE ACTIONS

The cracked piping will be repaired prior to startup from the current refueling outage.

ADDITIONAL INFORMATION

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

No similar events have occurred in the previous two years in which a leak in the EHC system resulted in or necessitated a reactor scram.

Failed Component Information:

Plant Identification : 2N32-B001
Component Description : EHC System Piping
Manufacturer : General Electric Corporation
Manufacturer Code : G082
EIIS Component Code : PSP