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Georgia Power
the southern electric system

HL-1545
001384

March 27, 1991

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 MAIN
TURBINE TRIP AND REACTOR SCRAM

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 unanticipated actuation of some Engineered Safety Features (ESFs). This event occurred at Plant Hatch - Unit 1.

Sincerely,

J. T. Beckham, Jr.

JJP/CT/cr

Enclosure: LER 50-321/1991-007

cc: (See next page.)

IE22
1/1

Georgia Power

cc: Georgia Power Company
Mr. H. L. Sumner, General Manager - Nuclear Plant
Mr. J. D. Heidt, Manager Engineering and Licensing - Hatch
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

001384

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 6

TITLE (4)
PERSONNEL ERROR RESULTS IN MAIN TURBINE TRIP AND REACTOR SCRAM

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
02	27	91	91	007	00	03	27	91		05000

OPERATING MODE (9): 1

POWER LEVEL: 100

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

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	MANUFAC-TURER	REPORT TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	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 2/27/91 at approximately 1247 CST, Unit 1 was in the Run mode at an approximate power level of 2436 CMWT (approximately 100% rated thermal power). At that time, both Recirculation Pumps tripped while plant Instrument and Controls (I&C) technicians were performing the quarterly surveillance on the turbine first stage pressure permissive input to the Reactor Protection System and the End-of-Cycle (EOC) Recirculation Pump trip logic. Loss of recirculation system flow caused a rapid increase in reactor water level and the Main Turbine and both Reactor Feedwater Pump Turbines tripped on high reactor water level. The Main Turbine trip resulted in a full scram per design. Reactor water level decreased such that the Group 2 Primary Containment Isolation System valves isolated. Reactor water level was recovered using the High Pressure Coolant Injection system and "A" Reactor Feedwater Pump. Reactor pressure was controlled with the Main Turbine Bypass Valves.

The cause of this event was personnel error. I&C technicians had inadvertently left in place a jumper following performance of a different surveillance on 2/22/91. This jumper, in conjunction with the surveillance on the turbine first stage pressure permissive instruments, was sufficient to actuate enough of the EOC Recirculation Pump trip logic to cause a trip of both pumps.

Corrective actions for this event include counseling the involved personnel, performing an inspection of both units' Main Control Room panels, and revising procedures.

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 2/27/91 at approximately 1247 CST, Unit 1 was in the Run mode at an approximate power level of 2436 CMWT (approximately 100% rated thermal power). At that time, both Reactor Recirculation Pumps (EIIS Code AD) tripped while plant Instrument and Controls (I&C) technicians were performing the quarterly surveillance on the turbine first stage pressure permissive input to the Reactor Protection System (RPS, EIIS Code JC) and the End-of-Cycle (EOC) Recirculation Pump trip logic (included in the RPS system logic). Loss of recirculation system flow caused a rapid decrease in reactor power and a rapid increase in reactor water level as a result of increased core voiding. Within five seconds of the trip of the Recirculation Pumps, the Main Turbine (EIIS Code TA) and both Reactor Feedwater Pump Turbines (EIIS Code SJ) tripped on high reactor water level. The Main Turbine trip resulted in automatic actuation of both channels of the RPS and a full scram per design. As expected following a scram and with both Reactor Feedwater Pumps tripped, reactor vessel water level decreased. Water level decreased to the actuation setpoint for the Group 2 Primary Containment Isolation System (EIIS Code JM) and the Group 2 Primary Containment Isolation Valves closed. Reactor water level was recovered from its minimum level of 23 inches below instrument zero (141 inches above the top of the active fuel) using the High Pressure Coolant Injection (HPCI, EIIS Code BJ) system and the "A" Reactor Feedwater Pump. Reactor pressure was controlled with the Main Turbine Bypass Valves (EIIS Code SO).

The cause of this event was personnel error. I&C technicians had inadvertently left in place a jumper following performance of a different surveillance on 2/22/91. This jumper, in conjunction with the surveillance on the turbine first stage pressure permissive instruments, was sufficient to actuate enough of the EOC Recirculation Pump trip logic to cause a trip of both pumps.

Corrective actions for this event include counseling the involved personnel, performing an inspection of both units' Main Control Room panels, and revising procedures.

DESCRIPTION OF EVENT

On 2/27/91 at approximately 1145 CST, I&C technicians began performance of surveillance procedure 57SV-C71-003-1S, "Turbine First Stage Pressure Permissive FT&C." This procedure, performed quarterly to comply with the surveillance requirements of Unit 1 Technical Specifications Table 4.1-1, item 12, tests the function and setpoint of pressure switches 1C71-N003A through D. These pressure switches sense Main Turbine first stage pressure. When the plant is at less than or equal to 30% rated thermal power, as measured by these pressure

TEXT

switches, the logic is bypassed for the Turbine Control Valve fast closure and Turbine Stop Valve closure scrams, and the EOC Recirculation Pump trip. Below 30% rated thermal power these trips are not required.

Testing of pressure switches 1C71-N003A through C was successfully performed. However, at approximately 1247 CST, during testing of the last pressure switch, 1C71-N003D, both Recirculation Pumps unexpectedly and automatically tripped. As expected, the loss of recirculation system flow caused a rapid decrease in reactor power and a rapid increase in reactor water level as a result of increased core voiding. Within five seconds of the trip of the Recirculation Pumps, the Main Turbine and both Reactor Feedwater Pump Turbines tripped as reactor water level reached the high level trip setpoint of 56 inches above instrument zero.

The reactor scrammed due to Turbine Stop Valve closure. As expected following a scram and with both Reactor Feedwater Pumps tripped, reactor vessel water level decreased. Water level decreased to about 12.5 inches above instrument zero, the actuation setpoint for the Group 2 Primary Containment Isolation System and the Group 2 Primary Containment Isolation Valves closed per design.

Reactor water level continued to decrease to its minimum level during this event of 23 inches below instrument zero. At approximately 1248 CST, Operations personnel manually started the HPCI system (its automatic initiation setpoint is 35 inches below instrument zero) to recover water level. At approximately 1251 CST, the Reactor Feedwater Pumps were reset and Operations personnel began feedwater injection with the "A" pump. The HPCI system was secured. By 1255 CST, reactor water level was returned to its normal range and was maintained with the Reactor Feedwater Pump.

Reactor pressure reached its peak during the event of 1031 psig immediately following the scram. The Main Turbine Bypass Valves opened to decrease and control reactor pressure per design. No Safety Relief Valves opened nor were any required to open.

CAUSE OF THE EVENT

The cause of this event was personnel error. Several days prior to this event, on 2/22/91, surveillance procedure 34SV-C71-005-1S, "Turbine Control Valve Fast Closure Instrument Functional Test," was performed. At that point, I&C technicians inadvertently left in place a test jumper used to simulate a Turbine Control Valve fast closure signal in the "B" Recirculation Pump EOC trip logic system. The procedural steps requiring the test jumper to be removed at the end of the test and its removal to be verified independently were therefore not successfully completed. Then on 2/27/91, while performing the functional test on pressure switch 1C71-N003D per procedure 57SV-C71-[illegible]-1S, the other Turbine Control Valve fast closure signal in the same portion of the EOC

Recirculation Pump trip logic system was also simulated. When pressure switch 1C71-N003D was pressurized, per procedure to check its setpoint, the bypass of the EOC Recirculation Pump trip logic was removed. The sensing of two Turbine Control Valve fast closure signals in the same trip system, together with the removal of the bypass, was sufficient to actuate the EOC Recirculation Pump trip. Both Recirculation Pumps tripped per design.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This report is required per 10 CFR 50.73(a)(2)(iv) because unplanned actuations of the RPS and an Engineered Safety Feature occurred. Specifically, the RPS actuated on Turbine Stop Valve closure when the Main Turbine tripped on high reactor water level resulting from a trip of both Recirculation Pumps. Subsequent to the scram, the Group 2 Primary Containment Isolation System, an Engineered Safety Feature, received an isolation signal on low reactor water level.

The RPS automatically initiates a reactor scram to ensure the radioactive materials barriers, such as fuel cladding and the pressure system boundary, are maintained and to mitigate the consequences of transients and accidents. Closure of the Turbine Stop Valves, such as occurs on a main turbine trip, can result in the addition of positive reactivity to the core as the resultant reactor pressure increase collapses voids. Therefore, Turbine Stop Valve closure initiates a scram prior to high neutron flux or high reactor pressure signals to provide a satisfactory margin to core thermal-hydraulic safety limits. The high-pressure scram, in conjunction with the pressure relief system, is adequate to preclude overpressurizing the pressure system boundary; however, the Turbine Stop Valve closure scram provides additional margin.

Turbine Control Valve fast closure and Turbine Stop Valve closure initiate the EOC Recirculation Pump trip logic whenever the Main Turbine first stage pressure is above that which corresponds to 30% rated thermal power. The EOC Recirculation Pump trip logic is arranged such that redundant signals of either Turbine Control Valve fast closure or Turbine Stop Valve closure in one trip system, in conjunction with being above 30% rated thermal power are necessary to trip the Recirculation Pumps. The logic's function is to trip the Recirculation Pumps in response to a Main Turbine trip. By tripping the Recirculation Pumps early in the event the severity of the Main Turbine trip is reduced. The rapid reduction in core flow reduces void collapse during pressurization events resulting in the addition of less positive reactivity to the core. As with the scram on Turbine Stop Valve closure, the EOC Recirculation Pump trip provides a satisfactory margin to core thermal-hydraulic safety limits.

The Primary Containment Isolation System provides timely protection against the onset and consequences of events involving the potential release of radioactive materials from the fuel and nuclear system process barriers by isolating appropriate lines which penetrate the primary containment. A Group 2 Primary

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Containment Isolation System isolation initiated by a low reactor water level condition prevents the escape of radioactive materials from the primary containment through process lines which may have been breached. Additionally, isolation of these process lines conserves reactor coolant inventory if a breach of one of these lines caused the low water level condition.

In the event described in this report, the Recirculation Pumps tripped when two Turbine Control Valve fast closure signals were simulated in the "B" EOC Recirculation Pump trip logic system and the logic bypass for reactor power below 30% was removed. The pump trip caused a reactor water level increase which tripped the Main Turbine. This, in turn, caused a reactor scram. As water level decreased following the scram, a Group 2 Primary Containment Isolation System isolation signal was received on low reactor water level and the Group 2 Primary Containment Isolation Valves closed. All systems functioned per design, given the signals simulated during the performance of the surveillance procedure and by the jumper inadvertently left in place, and the resulting transient. Reactor water level was recovered with the HPCI system and the "A" Reactor Feedwater Pump and maintained at its normal level with the "A" pump. Water level never decreased below 141 inches above the top of the active fuel. Reactor pressure was controlled by the Main Turbine Bypass Valves. The Bypass Valves limited the peak reactor pressure to 1031 psig, only 31 psig above the pre-event pressure and well below the Safety Relief Valves' lift pressures of 1080 psig, 1090 psig, and 1100 psig.

The jumper inadvertently left in place for five days following performance of procedure 34SV-C71-005-1S on 2/22/91 left the "B" EOC Recirculation Pump trip logic system in a partially tripped condition. Consequently, only one additional Turbine Control Valve fast closure signal, not the normal redundant signals, was needed to actuate this logic system and trip both Recirculation Pumps. The logic, therefore, was in a conservative condition during those five days. It would have functioned to trip the Recirculation Pumps had an actual Main Turbine trip occurred during that period.

Based on the above analysis, it is concluded this event had no adverse impact on nuclear safety. This analysis is applicable to all power levels.

CORRECTIVE ACTION

The involved I&C technicians were counseled.

Both units' Main Control Room panels were inspected to determine if any other problems of this nature existed. Two links in the Main Control Room Environmental Control system actuation logic were found open in a Unit 1 panel. These links had been left open following the performance of another surveillance procedure. Similar to the jumper left in the EOC Recirculation Pump trip logic, these open links left the Main Control Room Environmental Control system actuation logic in a partially tripped condition. Consequently, instead of the normal two actuation signals needed to actuate the system, only one was needed. This logic, too, was in a conservative condition with the links open. The personnel who left open the links contrary to procedural requirements were counseled.

TEXT

Procedures 34SV-C71-005-1S and 57SV-C71-003-1S will be revised to include additional checks to ensure the EOC Recirculation Pump trip logic has been returned to normal status following testing. Specifically, steps will be added to ensure lights indicating the status of the Turbine Stop Valve closure and Turbine Control Valve fast closure relays in the trip logic are lit before and after performance of the respective tests. These indicating lights, four per EOC Recirculation Pump trip system, extinguish when their corresponding valve closure signals are received. Therefore, checking that these lights are lit will provide assurance the logic is in its normal condition both before performing and after completing any testing. Similar changes will be made to the corresponding Unit 2 surveillance procedures. These revisions will be effective by 4/30/91.

ADDITIONAL INFORMATION

1. Other System Affected:

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

2. Failed Components Identification:

   No failed components caused or resulted from this event.

3. Previous Similar Events:

   Previous similar events in the last two years in which the reactor scrammed as a result of a Main Turbine trip were reported in the following LERs:

   LER 50-321/1991-004, dated
   LER 50-321/1991-001, dated 02/11/91
   LER 50-321/1990-020, dated 10/26/90
   LER 50-366/1991-004, dated

   Corrective actions for these events would not have prevented this event because the causes of the Main Turbine trips were different. The previous events were caused by component failure whereas this event was caused by personnel error.