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Georgia Power
the southern electric system

Docket No. 50-366 April 15, 1994 HL-4560

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

Edwin I. Hatch Nuclear Plant - Unit 2
Licensee Event Report
Fuse Actuation Results in Engineered Safety Features
Actuation and Interruption in Shutdown Cooling Flow

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 a fuse actuation which resulted in Engineered Safety Features actuations and an interruption of shutdown cooling flow. This event occurred at Plant Hatch - Unit 2.

Sincerely,

J T Beckham Jr

J. T. Beckham, Jr.

OCV/cr

Enclosure: LER 50-366/1994-003

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
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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)
Edwin I. Hatch Nuclea: Plant - Unit 2	05000366	1 OF 8

TITLE (4)
Fuse Actuation Results in ESF Actuation and Interruption in Shutdown Cooling Flow

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 NAME	DOCKET NUMBER(S)
03	17	94	94	— 003 —	00	04	15	94		05000
										05000

OPERATING MODE (9)	4
POWER LEVEL (10)	000

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)	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	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

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 3/17/94, at 1131 EST, Unit 2 was in the Cold Shutdown mode, one day into a scheduled refueling outage. Reactor coolant temperature was 168 degrees Fahrenheit. The "B" loop of the Residual Heat Removal (RHR) System was in operation in the Shutdown Cooling (SDC) mode. At that time, a non-licensed engineer was tracing the routing of a control wire in a Control Room panel when he observed an arc in the panel and heard several relays actuate. He notified the licensed Shift Supervisor. Subsequent walkdown of the control room panels revealed that the loop "B" SDC discharge valve, 2E11-F015B, was closed and SDC flow was zero. An operator attempted to open the valve with the control switch but was unsuccessful. Shift personnel then entered procedure 34AB-E11-001-2S, "Loss of Shutdown Cooling," and began, with engineers' assistance, to investigate the cause of the valve isolation. At 1202 EST, the cause of the event had not been determined, and it was decided to place the "A" loop of RHR in the SDC mode. By 1250 EST, SDC flow had been restored via the "A" loop of the RHR system. Bulk average reactor coolant temperature did not exceed 212 degrees Fahrenheit throughout the event. However, investigation showed that the reactor vessel pressurized to approximately 9 psig during the event. The cause of the event was inadvertent grounding of a Primary Containment Isolation System logic circuit resulting in a fuse actuation and isolation of the SDC discharge valve. Corrective actions include replacing the fuse, repairing a wire termination, revising procedures, changing an alarm setpoint and training personnel.

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 3/17/94, at 1131 EST, Unit 2 was in the Cold Shutdown mode, one day into a scheduled refueling outage. Reactor coolant temperature was 168 degrees Fahrenheit as measured at the inlet to the Reactor Water Cleanup System (RWCU, EIIS Code CE). The "B" loop of the Residual Heat Removal System (RHR, EIIS Code BO) was in operation in the Shutdown Cooling (SDC) mode. The "A" loop of RHR was aligned to the Low Pressure Coolant Injection (LPCI) mode. The "A" and "B" loops of Core Spray were operable. The 2A, 1B, and 2C Emergency Diesel Generators (EDG, EIIS Code EK) were operable. The reactor vessel was vented to the Unit 2 Drywell via the reactor head vent line. The Unit 2 Drywell head had been removed; thus, Unit 2 Primary Containment was communicating with Unit 1 Secondary Containment. (The two units share a common refueling floor which is part of Unit 1 Secondary Containment.) Unit 1 Secondary Containment was intact but the Unit 2 Secondary Containment (Reactor Building) was not intact. All Drywell penetrations which allow communication between Unit 2 Primary Containment and Unit 2 Secondary Containment were isolated.

Prior to the event, it had been determined that several control circuit relay coils needed replacement. In support of the replacement effort, a nonlicensed engineer was tasked with physically verifying the wiring configuration of the relays. At 1131 EST, the engineer was tracing the routing of a control wire in Main Control Room panel 2H11-P623 when he observed an arc in the panel and heard several relays actuate. After being notified of the arc, the Shift Supervisor inspected the panel and directed the licensed operators to walk down the control panels, noting any unusual conditions that may have resulted from the arcing incident. At approximately 1140 EST, an operator walking down the panels discovered that the loop "B" SDC discharge valve, 2E11-F015B, was closed and that SDC flow was zero. The operator notified the Shift Supervisor. No plant conditions existed that would necessitate closure of the valve; therefore, the operator attempted to open the valve with the control switch. The valve cycled open and then automatically closed indicating that a Primary Containment Isolation System (PCIS, EIIS Code JM) signal was in effect on the valve control logic. This valve is a dual function valve providing Primary Containment isolation capability as well as injection capability for the SDC and the Low Pressure Coolant Injection (LPCI) modes of the RHR system. The operator then reset the PCIS signal and attempted to open the valve. Again, the valve cycled open and then automatically closed, indicating that an invalid PCIS signal was sealed in. The operating RHR pump was then secured and the discharge throttle valve, 2E11-F017B, was closed.

Shift personnel then entered procedure 34AB-E11-001-2S, "Loss of Shutdown Cooling," and began, with engineers' assistance, to investigate the cause of the PCIS signal. Procedure 34AB-E11-001-2S requires increased monitoring (i.e., every 15 minutes) of the reactor coolant temperature and the reactor vessel pressure, as well as other plant parameters. The procedure also requires that reactor water level be raised to greater than 53 inches above instrument zero, if SDC flow is not restored immediately, in order to induce natural circulation in the reactor vessel. Prior to the event, the reactor water level was 37 inches above instrument zero (195 inches above the top of the active fuel) and was therefore raised to 57 inches above instrument zero.

Procedure 34AB-E11-001-2S requires that if reactor coolant temperature approaches 212 degrees Fahrenheit an alternate means of SDC should be established. The "A" loop of RHR was operable and capable of being aligned to the SDC mode if necessary. However, since temperature as measured at the RWCU system inlet was essentially constant at 168 degrees Fahrenheit, placing the "A" loop of RHR in the SDC mode was not immediately pursued and activities focused on monitoring plant parameters and identifying and correcting the cause for the invalid PCIS signal.

At approximately 1202 EST, licensed personnel saw reactor coolant temperature as measured at the RWCU inlet begin increasing. At this point, the cause of the event had not been determined and it was decided to place the "A" loop of RHR in operation in the SDC mode. By 1250 EST, reactor coolant temperature as measured at the RWCU inlet had increased to 185 degrees Fahrenheit and SDC flow had been restored via the "A" loop of the RHR system. Forced circulation was thus restored and the higher temperature coolant in and above the reactor core region was being moved into the annulus region. Consequently, RWCU inlet temperature initially increased, reaching a maximum of 194.8 degrees at 1255 EST, before it began to decrease.

By 1325 EST, a blown fuse in a Group 2 PCIS initiation circuit had been found and replaced. Subsequently, the PCIS signal was reset and the 2E11-F015B valve was cycled to ensure its operability.

An Event Review Team was established to fully investigate the event. In reviewing strip charts and process computer printouts after the event, the team determined that the reactor pressure had reached a maximum of approximately 9 psig during the interruption in SDC flow. Pressure was greater than atmospheric for approximately 3 hours. Because of the apparent pressurization of the reactor vessel, there was concern that the plant had gone from the cold shutdown condition, to hot shutdown. Analysis of the event by General Electric, however, showed that the bulk average temperature was less than 212 degrees Fahrenheit throughout the event, indicating that the Hot Shutdown mode had not been entered. (By definition, the plant is in the hot shutdown mode when the average reactor coolant temperature is above 212 degrees Fahrenheit).

CAUSE OF EVENT

The cause of the interruption in SDC flow was inadvertent grounding of a PCIS initiation circuit. S[illegible]ally, an engineer, in attempting to identify control wiring, slightly moved a bundle of control w[illegible] a control panel to view a wire label. One of the wires in the bundle that was landed to a nearby relay terminal contacted a metal cable raceway when the bundle was moved, grounding the circuit. This wire is a multi-strand wire and is attached to the relay with a compression fitting. One of the strands of the wire had been separated from the other strands apparently during installation and was not held by the compression fitting. When the bundle was moved, the wire moved enough to bring the strand into contact with an adjacent metal raceway, grounding the associated circuit. The grounding incident caused the circuit fuse to actuate and the circuit to de-energize. The wire [illegible] part of a PCIS initiation logic circuit which is of a fail-safe design in that it de-energizes to [illegible]te an actuation. Consequently, when the fuse actuated, a PCIS signal was generated and several outboard PCIS valves including valve 2E11-F015B automatically closed.

Pressurization of the reactor vessel was caused by several factors. First, the reactor had been shutdown approximately 34 hours before the event; thus, decay heat load was relatively high. Therefore, with the interruption in SDC flow, localized boiling occurred in the reactor core and steam was generated and emitted into the steam dome area of the reactor vessel. The reactor head vent line was open at the time of the event. However, the 1/2 inch line was not of sufficient capacity to totally vent off the steam being generated, resulting in slight pressurization of the reactor vessel.

An additional factor contributing to the event was that licensed personnel believed that with the reactor head vent open, pressurization of the reactor vessel was not possible for the conditions existing at the time of the event. Procedure 34AB-E11-001-2S also contained wording that implied that pressurization was not possible with the head vent open. However, a review of design documentation for the vent line showed that the purpose for the vent was to provide a path for noncondensibles when flooding up the vessel for hydrostatic testing, not to prevent pressurization due to localized boiling.

Procedure 34GO-OPS-015-2S, "Maintaining Cold Shutdown," was less than adequate in that it specified inappropriate instrumentation for monitoring reactor pressure in low pressure conditions. Specifically, the procedure directed licensed personnel to use reactor pressure indicators 2C32-

R605A, B, or C or the Process Computer for monitoring reactor pressure in the Cold Shutdown mode. The scale of the pressure indicators is 0 - 1200 psig and the smallest graduation is 20 psig on a spacing of 3/16-inch. Consequently, a pressure increase of the magnitude occurring in this event would be difficult if not impossible to discern on these indicators. The Process Computer, however, provides a digital output that is accurate at low pressures. In this event, licensed personnel, given the option to use the indicators or the Process Computer, used the indicators and, therefore, did not see the rise in reactor pressure during the event.

A Process Computer alarm for low core flow that had recently been installed did not annunciate during the event. The alarm was designed to annunciate when jet pump flow decreased below the level expected with SDC in service, providing an indirect indication for an interruption of SDC flow. However, when the interruption in SDC flow occurred, the alarm did not annunciate. It was later determined that indicated jet pump flow did not decrease with the interruption in SDC flow. This was because the jet pump flow instrumentation is calibrated under high temperature conditions. Without density compensation integrated into the instrumentation, indicated flow increases as temperature of the coolant decreases even when actual flow remains constant. At the time of the event, the reactor coolant temperature was substantially less than that during normal operation. Therefore, due to the higher density of the lower temperature coolant, the indicated flow of the jet pumps did not decrease sufficiently to trip the alarm. Had the alarm functioned in this event, licensed personnel would have been made aware of the interruption in SDC flow approximately eight minutes earlier; the eight minute delay was, however, inconsequential in this event.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This report is required pursuant to 10 CFR 50.73(a)(2)(iv) because an unplanned Engineered Safety Feature (ESF) actuation occurred resulting in the automatic closure of Group 2 PCIS valves. Additionally, the event resulted in an interruption of SDC flow which further resulted in relatively low pressurization of the reactor pressure vessel. This aspect of the event is not reportable, however, is being included in the report since it was a consequence of the ESF actuation and may be of industry interest.

The Primary Containment Isolation System provides automatic isolation capability of Primary Containment penetrations to preclude the release of radioactive material and the loss of reactor coolant inventory in the unlikely event of an accident. In this event, inadvertent grounding of a control circuit resulted in a loss of power to a PCIS initiation circuit which is of a fail-safe design in that, upon a loss of power to the circuit, an isolation signal is generated. Consequently, a partial

PCIS isolation occurred resulting in several PCIS valves automatically closing. With the unit in an outage, many of the valves were already closed prior to the event. Had the event occurred coincident with an accident, the valves would still have closed as required, isolating the associated Primary Containment penetration.

One of the PCIS valves that closed as a result of the fuse actuation was valve 2E11-F015B. As stated previously, this valve is a dual function valve providing Primary Containment isolation in the closed position and SDC/LPCI injection capability in the open position. Consequently, when the valve automatically closed in this event, SDC flow was interrupted. The purpose of the SDC mode of the RHR system is to provide adequate cooling to the reactor core while the reactor is shutdown in order to reduce the reactor coolant temperature to and/or maintain it below 212 degrees Fahrenheit. In this event, SDC flow was interrupted for approximately one hour and twenty minutes. At the time of the event several backup systems, each of which are capable of maintaining adequate reactor core cooling, were available. The "A" loop of RHR-SDC was available and was ultimately used to re-establish cooling flow to the reactor core. Had the "A" loop of RHR not been available, either of the "A" or "B" loops of Core Spray (CS, EIIS Code BM), which were both available, could have been used for alternate shutdown cooling. In this mode of operation, the RHR System is aligned to the Suppression Pool Cooling mode. When reactor pressure reaches approximately 50 psig, a Safety Relief Valve (SRV, EIIS Code SB) would be opened providing a flow path from the reactor vessel to the Suppression Pool. A Core Spray pump would then be started taking suction from the Suppression Pool. Reactor water level would then be raised to the level of the Main Steam lines and flow would be established through the SRV to the Suppression Pool.

During the event, the interruption in SDC flow resulted in heatup of the reactor coolant. An analysis performed by General Electric showed that the bulk average temperature of the reactor coolant did not exceed 212 degrees Fahrenheit. Therefore, based on the Technical Specification criteria for entering the Hot Shutdown mode, a mode change to Hot Shutdown did not occur as a result of this event. However, the reactor vessel did pressurize to approximately 9 psig during the event. The General Electric Evaluation also determined that the noted pressure increase was credible.

Had the Hot Shutdown mode been reached during the event, the requisite conditions of Primary Containment and Secondary Containment would not have existed. Specifically, Unit 2 Primary Containment was not intact. The Drywell head had been removed prior to the event and Unit 2 Primary Containment was communicating with Unit 1 Secondary Containment. (The two units share a common refueling floor which is part of Unit 1 Secondary Containment.) The penetrations between Unit 2 Primary Containment and Unit 2 Secondary Containment were isolated; therefore, Unit 2 Primary Containment was not in communication with Unit 2 Secondary Containment. Unit 2 Secondary Containment was not intact in that the Reactor Building equipment door was open to support outage activities. Had the Hot Shutdown mode been reached, Unit 2 Secondary Containment could have been established quickly by closing the equipment door. Unit 2 Primary Containment could not have been established expeditiously due to the Drywell head being removed. However, due to the fact that no gross fuel failures existed, any radioactive releases into Unit 1 Secondary Containment would be minimal and would be processed by the Standby Gas Treatment system (SGT, EIIS Code BH) and released via an elevated release path. Consequently, 10 CFR 100 limits would not have been exceeded.

Based on the above information, it was concluded that this event had no adverse impact on nuclear safety. This assessment applies to all operating conditions.

CORRECTIVE ACTIONS

The fuse was replaced and the operability of the 2E11-F015B valve was verified.

The wire involved in the grounding incident was reterminated with all of the strands landed in the compression fitting.

The Process Computer alarm setpoint problem has been corrected.

Procedure 34GO-OPS-015-2S will be revised to specify the Process Computer as the primary means for monitoring reactor pressure in low pressure conditions. Other operations procedures will be reviewed for this problem and revised as necessary. This action will be completed by 7/29/94.

Procedure 34AB-E11-001-2S will be revised to state that the potential exists for pressurization of the reactor vessel upon interruption in SDC flow even if the reactor head vent is open. Other operations procedures will be reviewed for this problem and revised as necessary. This action will be completed by 7/29/94.

An engineering evaluation will be completed by 6/30/94 to determine if the reactor head vent line size should be increased.

The outage planning and scheduling philosophy has been revised to require that both loops of RHR be available until the reactor cavity is flooded.

Operations licensed shift personnel have been trained on this event with special emphasis on the potential for pressurizing the reactor vessel even with the head vent open.

ADDITIONAL INFORMATION

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

One event has occurred in the previous two years in which SDC flow was interrupted. The event was reported in LER 50-321/93-004, dated 5/14/93. In that event, a PCIS logic circuit fuse actuated when the circuit was inadvertently grounded during a wiring modification. The condition went unnoticed for approximately one and a half hours. Corrective actions for the event included increasing the frequency of checking critical plant parameters, including SDC flow, during Cold Shutdown and issuing a plantwide directive informing plant personnel of the possible consequences of grounding incidents and the need to aggressively investigate incidents to determine their effect on the plant. These corrective actions were not intended to prevent future interruptions in SDC flow. No reasonable actions can be taken to completely eliminate the potential for such events. These corrective actions were intended to prevent prolonged interruptions in SDC flow by helping to ensure that licensed personnel became aware of such interruptions as soon as possible. One of the corrective actions was instrumental in this manner. Specifically, the individual seeing the arcing in this event knew of the potential consequences of such arcing events because of previous training and notified the shift supervisor of the event almost immediately after it had occurred.

The frequency of SDC flow checks had been increased from once every four hours to once every hour as a result of the previous event. It had been performed thirty minutes prior to the event and was not scheduled to be performed for another thirty minutes when the event occurred. Had the interruption in SDC flow gone unnoticed for another thirty minutes, placing the "A" loop of SDC in service would most likely not have been delayed since the decision to do so was based on RWCU inlet temperature increasing which did not occur until approximately thirty minutes after the interruption occurred.

No failed components contributed to this event.