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U.S. Nuclear Regulatory Commission
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PLANT HATCH - UNIT 2
NRC DOCKET 50-366
OPERATING LICENSE NPF-5
LICENSEE EVENT REPORT
COMPONENT FAILURE AND INADEQUATE
DESIGN CAUSE GROUP I ISOLATION AND SCRAM

Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(iv) and 10 CFR 50.73(a)(2)(v), Georgia Power Company is submitting the enclosed, revised, Licensee Event Report (LER) concerning the unanticipated actuation of some Engineered Safety Features (ESFs) and a condition that prevented an ESF from fully performing its safety function. This event occurred in January of 1990 at Plant Hatch - Unit 2.

The revision is being submitted following a more detailed root cause investigation of the ESF component failure, per our commitment in LER 50-366/1990-001, Rev. 0.

Sincerely,

W. G. Hairston, III

JJP/ct

Enclosure: LER 50-366/1990-001 Rev 1

c: (See next page.)

IE22
1/1

c: Georgia Power Company
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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)
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TITLE (4)
COMPONENT FAILURE AND INADEQUATE DESIGN CAUSE GROUP I ISOLATION AND SCRAM

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 1	1 2	9 0	9 0	— 0 0 1	— 0 1	0 6	0 8	9 0		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 1 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)	X 50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	X 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 (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
X	D L	R T V	H 0 3 7	N					
X	B G	R L Y	G 0 8 0	Y					

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 fifteen single-space typewritten lines) (16)

On 1/12/90 at approximately 1610 CST, Unit 2 was in the Run mode at an approximate power of 2436 CMWT (approximately 100% of rated thermal power). At that time, the reactor scrammed because the Main Steamline Isolation Valves (MSIVs) were less than 90% open. The MSIVs had isolated on a Group 1 Primary Containment Isolation System (PCIS) signal which resulted from a false low condenser vacuum signal. The High Pressure Coolant Injection (HPCI) system automatically initiated and injected on low reactor water level as required. Following water level recovery, HPCI injection valve 2E41-F006 closed automatically on high water level; however, it could not be re-opened when Operations personnel subsequently attempted to start HPCI manually. The Reactor Core Isolation Cooling system and two Control Rod Drive system pumps were used to control water level following the failure of valve 2E41-F006 to open.

The root causes of the scram are component failure and the configuration of the condenser vacuum sensing lines and instruments. The disc of root isolation valve 2N61-F588D separated from its stem isolating the common sensing line for vacuum switches 2B21-N056C and D. Consequently, these switches then sensed a low condenser vacuum and, because they input to the A and B trip systems respectively of the isolation logic, the MSIVs isolated. The cause of valve 2E41-F006 failing to open is component failure. The heater strip of a thermal overload relay in the valve motor's local starter failed causing an open circuit to the motor. The failure of the heater strip appears to be an isolated event.

Corrective actions for this event included replacing the root isolation valves with a new type of valve, replacing the thermal overload relays in the local starter, and reconfiguring the tubing for the vacuum switches.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
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TEXT (If more space is required, use additional NRC Form 366A's) (17)

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 1/12/90 at approximately 1610 CST, Unit 2 was in the Run mode at an approximate power of 2436 CMWT (approximately 100% of rated thermal power). At that time, the reactor scrammed because the Main Steamline Isolation Valves (MSIVs, EIIS Code SB) were less than 90% open. The MSIVs had isolated on a Group 1 Primary Containment Isolation System (PCIS, EIIS Code JM) signal which resulted from a sensed low condenser vacuum. Vacuum switches 2B21-N056C and D, which provide input to the A and B trip systems of the PCIS Group 1 isolation logic, respectively, sensed low condenser vacuum after root isolation valve 2N61-F588D failed and isolated the switches' common sensing line. During normal scram recovery activities, the High Pressure Coolant Injection (HPCI, EIIS Code BG) system automatically initiated and injected on low reactor water level as required. Following water level recovery, HPCI injection valve 2E41-F006 closed automatically on high water level; however, it could not be re-opened when Operations personnel subsequently attempted to start HPCI manually. The Reactor Core Isolation Cooling (RCIC, EIIS Code BN) system and two Control Rod Drive (CRD, EIIS Code AA) system pumps were used to control water level following the failure of valve 2E41-F006 to open.

The root causes of the scram are component failure and the configuration of the condenser vacuum sensing lines and instruments. The disc of root isolation valve 2N61-F588D separated from its stem isolating the common sensing line for vacuum switches 2B21-N056C and D. These switches then sensed a low condenser vacuum and, because they input to the A and B trip systems respectively of the isolation logic, the MSIVs isolated. The cause of valve 2E41-F006 failing to open is component failure. The heater strip of a thermal overload relay in the valve motor's local starter failed causing an open circuit to the motor. The failure of the heater strip appears to be an isolated event.

Corrective actions for this event included replacing the root isolation valves with a new type of valve, replacing the thermal overload relays in the local starter, and reconfiguring the tubing for the vacuum switches, retaining the acceptability of the design relative to single failure criterion and reducing the potential for unnecessary full Group 1 isolation logic actuations.

DESCRIPTION OF EVENT

On 1/12/90 at approximately 1508 CST, Unit 2 was in the Run mode at an approximate power of 2436 CMWT (approximately 100% of rated thermal power). At that time, a Group 1 PCIS signal was received in the B isolation logic trip system. Operations and Instrument and Controls (I&C) personnel investigated the unexpected one-half Group 1 isolation signal. They discovered relay 2A71-K68D in the PCIS logic was de-energized. This relay de-energizes when vacuum switch 2B21-N056D trips on low condenser vacuum. It appeared at the time that relay 2A71-K68D had de-energized due to a failure of vacuum switch 2B21-N056D; chart recorder 2N21-R602 showed no indication of actual loss of condenser vacuum.

At approximately 1610 CST, before I&C technicians had gone to examine vacuum switch 2B21-N056D, the A isolation logic trip system actuated when vacuum switch 2B21-N056C tripped and a full Group 1 isolation signal resulted. The MSIVs began to close in response to the full Group 1 isolation signal as designed. When the MSIVs closed to less than 90% open, a full Reactor Protection System (RPS, EIIS Code JC) trip signal was generated as designed and the reactor scrammed.

With the MSIVs fully closed, the reactor was isolated from the condenser (EIIS Code SQ) and reactor pressure began to increase. Safety Relief Valves (SRVs, EIIS Code JE) 2B21-F013A, D, E, and H lifted to relieve pressure as designed. This action, in conjunction with the high pressure portion of the Low Low Set (LLS) arming logic having been fulfilled, armed LLS logic and LLS SRVs 2B21-F013B, F, and G lifted to control reactor pressure in the LLS mode. Reactor pressure peaked at approximately 1117 psig and LLS maintained pressure between 850 psig and 990 psig thereafter as designed.

As the SRVs actuated to relieve reactor pressure, water inventory was lost to the Suppression Pool (EIIS Code BT) as expected. Reactor water level decreased to approximately minus 40 inches relative to instrument zero (to approximately 10.4 feet above top of active fuel). PCIS Group 2 and 5 isolation signals were received and all Group 2 and 5 Primary Containment Isolation Valves closed as designed. Additionally, both Recirculation Pumps (EIIS Code AD) tripped, HPCI automatically started and injected to the vessel (RCIC had been started manually by Operations personnel prior to reactor water level reaching RCIC's automatic injection point), and both trains of the Standby Gas Treatment (EIIS Code BH) system started.

With HPCI and RCIC injecting water into the reactor, water level increased to the high reactor water level setpoint (approximately 52 inches above instrument zero) and HPCI and RCIC tripped per design. Reactor Feedwater Pumps A and B (EIIS Code SJ) also received a trip signal although they had stopped injecting water into the reactor vessel earlier in the event; with the MSIVs closed, no steam was available to drive their turbines. As the LLS SRVs continued to cycle to maintain reactor pressure between 850 psig and 990 psig, water level again began to decrease. Operations personnel attempted to start HPCI manually to control reactor water level; however, HPCI injection valve 2E41-F006 could not be opened from the Main Control Room using its Remote Manual Switch. Operations personnel proceeded to use RCIC and both CRD pumps to recover and maintain reactor water level. HPCI was used in the Full Flow Test Mode to assist in controlling reactor pressure.

At approximately 1704 CST, the unit was in a stable condition in the Hot Shutdown mode. Reactor water level was being maintained at the normal level of approximately 37 inches above instrument zero. Reactor pressure was being maintained at approximately 800 psig.

CAUSE OF THE EVENT

The root causes of the scram are component failure and the configuration of the condenser vacuum sensing lines and instruments. The disc in root isolation valve 2N61-F588D separated from its stem as a result of excessive wear due to vibration in the vacuum sensing line from the condenser. The keeper ring and stem shoulder had worn to the point where the disc fell off the stem. When the disc fell off the stem, it isolated the vacuum sensing line which feeds vacuum switches 2B21-N056C and D. Over an unknown period of time, the isolated line lost vacuum, probably from small inleakages through various sources. The line lost vacuum to the point where, at approximately 1508 CST, switch 2B21-N056D tripped. Due to slight differences in actual trip setpoints (the as-found trip setpoints of the two switches were within procedural tolerances, but slightly different), the line had to further lose vacuum to actuate the second vacuum switch; therefore, switch 2B21-N056C did not trip until approximately 1610 CST.

The configuration of the condenser vacuum sensing lines and instruments relative to plant reliability considerations resulted in an unnecessary full Group 1 isolation and the subsequent scram. Two vacuum switches were fed by one of two sensing lines off the condenser; two more vacuum switches were fed by the other sensing line. Each vacuum switch, in turn, provided input to one of four channels in the Group 1 isolation logic. These four channels (A1, A2, B1, and B2) are divided into two trip systems (A and B). One of the two channels in each system must actuate to trip the system and both systems must trip to generate a full Group 1 isolation signal and cause the MSIVs to close. Vacuum switch 2B21-N056D provided input to the B2 channel (the B trip system) and vacuum switch 2B21-N056C provided input to the A2 channel (the A trip system). With the two switches fed by the same sensing line providing inputs to channels in different trip systems, a single failure can cause false trips in both systems and a full Group 1 isolation signal. This design was implemented to meet single failure criterion should one of the two sensing lines fail in the ability to sense low condenser vacuum; however, it also resulted in an increased potential for unnecessary full isolation logic actuations. It should be noted that the low condenser vacuum (pressure) switches for the turbine (EIIS Code TA) trip logic are arranged the same way as the switches for the Group 1 isolation logic. The trip logics also are the same. Because a turbine trip will result in a reactor scram, the design of the turbine trip switches and logic is also inadequate.

The cause of HPCI injection valve 2E41-F006 failing to open is component failure. The heater strip of a thermal overload relay in the valve motor's local starter failed. The heater strip (a metallic strip which is part of the thermal overload relay) failed as attempts were made to open the valve. This created an open circuit to the motor. No current could reach the motor; therefore, it could not be energized to move the valve. The design configuration was reviewed and found to be in accordance with the applicable regulatory guidance and industry standards.

Investigation into the cause of the failure of the heater strip by Hatch's Architect/Engineer, Southern Company Services (SCS), concluded this was an isolated event. An exhaustive review of the design, operating conditions, and maintenance history of the injection valve and the injection valve starter, thermal overload relay, motor, and operator was performed. No conclusive evidence was found of any event or combination of circumstances which could have caused the heater strip damage; therefore, it was concluded that an independent component failure of the heater strip had occurred.

Testing of exact-kind heater strips at Wyle Laboratories revealed the heater strip will fail completely in six to ten seconds when subjected to a 1000% (400 amps) overcurrent. However, review of Motor Actuator Characterizer (MAC) motor time/current traces for the injection valve's opening strokes taken since 2/9/88, including those taken after the heater element failure on 1/12/90, indicated the motor is experiencing normal operating currents. The traces showed the maximum current occurs at the start of the valve opening stroke and is approximately 250 amps. This peak current lasts for a fraction of a second and, after the valve is unseated, immediately drops to less than 40 amps for the remainder of the valve opening stroke. Based on laboratory testing, these currents are not sufficient to cause heater strip failure. Moreover, visual inspection of the motor following the 1/12/90 event did not reveal any evidence it had experienced an overcurrent of the duration and magnitude necessary to destroy the heater strip.

SCS also reviewed other thermal overload relay failures at Plant Hatch 13 potentially similar events. All but one of the potentially similar events involved failure modes indicative of an actual overcurrent condition. In each of these 12 events there were overload alarms and/or thermal damage to the motor, contactor, and/or circuit breaker in addition to heater strip failure. With regard to the one event (1985) which does not fit this pattern, there is not enough information available to determine if other damage accompanied the heater strip failure. Therefore, based on the available information, SCS concluded independent failure of heater strips as seen in the 1/12/90 failure is not a common failure mode.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This report is required per 10 CFR 50.73 (a)(2)(iv) because an unplanned actuation of the RPS and Engineered Safety Features (ESF) occurred. Specifically, the RPS was initiated automatically on MSIVs less than 90% open. The other ESFs which activated during this event were the PCIS valve Groups 1, 2, and 5; the HPCI System; LLS; and the Standby Gas Treatment System. This report also is required per 10 CFR 50.73 (a)(2)(v) because the HPCI system did not function as designed following initial recovery of reactor water level. The injection valve's motor control circuitry failed thereby preventing HPCI from being used for continued reactor water level control.

The RPS automatically initiates a reactor scram to ensure the radioactive materials barriers (such as fuel cladding and pressure system boundary) are maintained and to mitigate the consequences of transients and accidents. The MSIV closure scram is provided to limit the release of fission products from the nuclear system. Automatic closure of the MSIVs can be initiated as a result of various conditions. One of these is low condenser vacuum. Low condenser vacuum indicates a possible leak in the condenser. Closing the MSIVs prevents potential loss of reactor coolant and potential release of radioactive material from the nuclear system process barrier.

The MSIVs have position switches installed on the valves. These switches provide RPS trip signals. If the MSIVs were to close suddenly, this could cause a rapid pressure increase in the reactor vessel. This pressure increase would affect the reactor vessel (due to the pressure increase) and result in a positive reactivity insertion (due to void collapse). The MSIV closure scram anticipates the neutron flux scram and the high pressure scrams. In this event all of the three RPS scrams (MSIV closure, neutron flux and high pressure) were operable and the MSIV position scram functioned as designed to terminate power production prior to the other variables (pressure and neutron flux) exceeding their trip setpoints.

Following the scram, reactor water level was restored via the automatic initiation of HPCI and the manual initiation of RCIC (RCIC was initiated manually prior to its automatic initiation setpoint). The SRVs operated in their relief and, later, LLS modes to control reactor pressure. Consequently, reactor vessel pressure was maintained well below vessel design pressure and vessel level did not decrease below approximately 10.4 feet above the top of the active fuel.

The HPCI system is provided to assure that the reactor is adequately cooled to limit fuel-clad temperature in the event of a small break in the nuclear boiler system causing a loss of coolant which does not result in rapid depressurization of the reactor vessel. The Automatic Depressurization System (ADS, EIIS Code JE) is a backup for the HPCI system. Upon ADS initiation, the reactor is depressurized to a point where either the Low Pressure Coolant Injection (LPCI, EIIS Code BO) system or the Core Spray (CS, EIIS Code BM) system can operate to maintain adequate core cooling.

In this event, the HPCI system was rendered inoperable following successful automatic initiation when its injection valve failed in the closed position when a manual re-start of HPCI was attempted. The LPCI pumps and their associated equipment, ADS, and both loops of CS were operable. Based upon the Unit 2 Final Safety Analysis Report (FSAR), either loop of the CS system or the LPCI system can supply sufficient cooling to the reactor for any rupture of the nuclear safety boundary up to and including the Design Basis Accident (DBA).

Based on the above information, it is concluded that this event had no adverse impact on nuclear plant safety. The above analysis is applicable to all reactor power levels.

CORRECTIVE ACTIONS

Root isolation valves 2N61-F588B and D (in the sensing lines for Group 1 isolation logic vacuum switches) and 2N61-F061 and F064 (in the sensing lines for turbine trip logic vacuum switches) were removed and replaced. They were replaced with gate valves that are not susceptible to the failure mode of valve 2N61-F588D. The gate valves' disc and stem are one piece; therefore, sensing line vibration can not cause the disc to rotate and wear any disc retaining parts as was the case with the failed valve. The valves also were installed upside down so any catastrophic failure of the disc/stem will not result in the disc falling into and isolating the sensing line. Additionally, the disc was removed from root isolation valves 2B21-R462A and B, and 2N61-F009 and F010 to prevent any sensing line vibration from causing the disc to separate and isolate its vacuum sensing line. Two isolation valves still remain in each sensing line, i.e., the new gate valve and the instrument isolation valve. (This latter valve is located on the instrument rack and is not subject to effects from sensing line vibration.) Design Change Request (DCR) 2H90-003 was developed and approved, in accordance with plant administrative controls, to allow for these changes to the valves in the condenser vacuum sensing piping. The new valves were installed and the discs removed in the other isolation valves on 1/14/90.

The vacuum sensing lines (3/8 inch stainless tubing) were reconfigured such that each of the four sensing lines off the condenser now has one Group 1 isolation logic vacuum switch and one turbine trip logic vacuum switch. The new arrangement is single failure proof and prevents spurious trips due to single failures. The sensing lines were reconfigured on 1/14/90. Design Change Request 2H90-003 also provided for reconfiguration of the vacuum sensing lines.

The thermal overload relays in the motor's local starter for HPCI injection valve 2E41-F006 were removed and replaced under a Maintenance Work Order. The valve was functionally tested using the MAC test equipment to ensure the torque switch settings were correct and the valve was functioning properly. As-found torque switch settings were acceptable and the valve stroked open and closed properly. HPCI was declared operable and returned to service at approximately 1950 CST on 1/14/90.

The Unit 1 vacuum sensing lines, root isolation valves, and Group 1 isolation and turbine trip logics were examined to determine if similar problems existed. It was found that the root isolation valves were not of the same type as those on Unit 2. However, the sensing line and logic arrangements were similar. Therefore, Design Change Request 1H90-009 has been generated to remove the root isolation valves from the Unit 1 vacuum sensing lines in order to preclude any type of valve failure from causing a false low vacuum signal to be generated. The Unit 1 design change will be implemented prior to startup from the current Unit 1 Refueling Outage.

ADDITIONAL INFORMATION

1. Previous Similar Events:

   There was one similar event in which the reactor scrammed due to MSIVs less than 90% open. This event was reported in LER 50-321/1988-009, dated 6/20/88. In that event, the MSIVs drifted closed from loss of air due to an incorrect instrument air system valve lineup. Corrective actions taken for that event would not have prevented the event described in this LER because the causes of the MSIVs closing are different.

2. Failed Components Identification:

   a. Master Parts List Number: 2N61-F588D
      Manufacturer: Hancock Root Cause Code: X
      Model Number: 5500W EIIS Component Code: RTV
      Type: Root Isolation Valve
      Manufacturer Code: H037
      EIIS System Code: DL
      Reportable to NPRDS: No

   b. Master Parts List Number: None
      Manufacturer: General Electric Root Cause Code: X
      Model Number: CR 124L028 EIIS Component Code: RLY
      Type: Thermal Overload Relay
      Manufacturer Code: G080
      EIIS System Code: BG
      Reportable to NPRDS: Yes

3. Other Affected Equipment:

   No systems other than the RPS, LLS, PCIS, and HPCI were affected by this event.