NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
EDWIN I. HATCH, UNIT I	0 5 0 0 0 3 2 1	1 OF 0 3

TITLE (4)

Reactor Scrams (Engineered Safety Feature 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 NAMES	DOCKET NUMBER(S)
0 1	1 6	8 5	8 5	0 1 0		0 2	1 4	8 5		0 5 0 0 0
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OPERATING MODE (9)	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
1	20.402(b)	20.405(c)	X 50.73(a)(2)(iv)	73.71(b)
POWER LEVEL (10) 0 6 1	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
	AREA CODE
T. L. Elton, Acting Superintendent of Regulatory Compliance	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	
N/A											

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 01/16/85, at approximately 0955 CST, with the reactor mode switch in the run position and reactor power at 1496 MWt (approximately 61% power), Unit 1 received a reactor scram on a reactor vessel low water level signal.

This event was the result of a vital AC power supply trip, which caused both "A" and "B" Reactor feedpumps to runback. Consequently, a low water level condition resulted.

At approximately 1102 CST, while plant personnel were lowering reactor pressure, the reactor vessel water level increased to the high level setpoint and the reactor feedpumps tripped. Before plant personnel could reset the feedpumps, with reactor power at 0 MWt , the reactor level dropped to the low level setpoint. This caused an actuation of the Reactor Protection System logic (i.e., not a scram because all of the control rods were still inserted after the 0955 CST scram), and a Group 2 isolation. Plant personnel restored the reactor level to normal via the "A" reactor feedpump and the booster pumps as the vessel continued to depressurize.

No actual or potential safety consequences or implications resulted from these events. These events had no impact on any other Unit 1 system or on Unit 2. The health and safety of the public were not affected by these events. These are non-repetitive events; however, the last reactor scram is referenced in LER 50-321/1985-03.

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

This 30 day LER is required by 10CFR50.73(a)(2)(iv) due to the reactor scram and actuation of other Engineered Safety Features.

On 01/16/85, at approximately 0955 CST, with the reactor mode switch in the Run position and reactor power at 1496 MWt (approximately 61% power), Unit 1 scrammed due to a reactor low water level trip signal.

While Vital AC was being powered by its alternate power supply, the alternate power supply tripped, causing a runback of both reactor feedpumps. Reactor water level then decreased to approximately -100 inches (reference instrument zero). A group 2 isolation and a reactor scram were received due to the low water level signal. HPCI and RCIC auto-started and injected to recover reactor water level; at the same time, the "A" reactor feedpump was started manually. As the feedpump was recovered, HPCI and RCIC were secured. The high water level turbine trip was received immediately thereafter for the following: the HPCI, RCIC, and reactor feedpump turbines. Reactor pressure remained steady at approximately 980 PSIG and was controlled with the EHC system (the MSIV's were not closed, and the turbine had been manually tripped). Standby gas treatment initiated on low water level; no other ECCS systems started nor were needed.

After the group 2 isolation was reset (at approximately 1005 CST), the RWCU outboard isolation valve (G31-F004) was opened to reestablish RWCU system flow in order to accurately determine the bottom head temperature (done to determine if a reactor recirculation pump could be restarted). Plant personnel then noted that the water entering the vessel had cooled down the bottom head temperature such that the differential temperature between the steam dome and the bottom head was greater than the 145^{o} F limit of Tech. Specs. section 3.6.E. Thus, the reactor recirculation pumps could not be started. Plant personnel then decided to depressurize the vessel (by lowering the EHC setpoint) to get the temperature within the limit so that a reactor recirculation pump could be started.

At approximately 1102 CST, while plant personnel were lowering reactor pressure, the reactor vessel water level increased to the high level setpoint and the reactor feedpumps tripped. Before plant personnel could reset the feedpumps, with reactor power at 0 MWt , the reactor level dropped to the low level setpoint. This caused an actuation of the Reactor Protection System logic (i.e., not a scram because all of the control rods were still inserted after the 0955 CST scram), and a Group 2 isolation. Plant personnel restored the reactor level to normal via the "A" reactor feedpump and the booster pumps as the vessel continued to depressurize.

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

At the time of the initial scram, plant personnel noted that several other loads (i.e., in addition to Vital AC) were lost from the 600 volt bus "1C". These included the following: the "1A" RBCCW pump, the "1A" air compressor, the main turbine motor suction pump and turning gear oil pump, and an electric fire pump lock out relay on the "E" 4160 volt bus was energized. Upon a subsequent investigation, plant personnel determined that undervoltage relay ETR1A would have caused the loss of all those components, had it actuated. It is believed that the actuation of this relay was the most likely cause of the scram.

Other problems experienced during the scrams were as follows:

1. At approximately 1030 CST, RCIC was declared inoperable due to its trip and throttle valve not engaging the reset circuitry and not opening.

2. At approximately 1320 CST, HPCI was declared inoperable due to its turbine stop valve binding in the mid-position; this binding was the cause of erratic HPCI flow which was experienced during the initial scram.

3. When both HPCI and RCIC were determined to be inoperable, plant personnel depressurized the reactor with the objective of decreasing reactor pressure below 113 psig as required by Tech. Specs. sections 3.5.D.7 and 3.5.E.3.

Plant personnel investigated the RCIC problem, and determined that RCIC's trip and throttle valve was in the continuous trip position due to the centrifugal trip weight's spring becoming loose. The spring apparently became loose during operation, thus causing damage to other parts of the trip/reset mechanism. The trip and throttle valve's trip/reset mechanism was repaired by replacing the spring, spring seat, weight, tappet guide, tappet and ball assembly, and the compression spring and emergency tappet nut. RCIC was functionally tested satisfactorily per the "RCIC PUMP OPERABILITY" procedure (HNP-1-3405), and returned to service on 1/18/85.

Plant personnel investigated the HPCI problem, and determined that its turbine stop valve had stuck in the mid-position due to its having a galled stem. The valve was repaired by polishing the inside of the valve and the valve's stem. The valve was then functionally tested satisfactorily, and HPCI was returned to service on 01/17/85.

No actual or potential safety consequences or implications resulted from these events. These events had no impact on any other Unit 1 system or on Unit 2. The health and safety of the public were not affected by these events. These are non-repetitive events; however, the last reactor scram is referenced in LER 50-321/1985-03.

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Georgia Power

Edwin I. Hatch Nuclear Plant

February 14, 1985
GM-85-153

PLANT E. I. HATCH
Licensee Event Report
Docket No. 50-321

United States Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

Attached is Licensee Event Report No. 50-321/1985-010. This report is required by 10CFR 50.73(a)(2)(iv).

H. C. Nix
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HCN/TLE/vlz

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