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 1	0 5 0 0 0 3 2 1	1 OF 0 4

TITLE (4)
Unplanned Reactor 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 8	0 3	8 4	8 4	— 0 1 5	— 0 0	0 8	2 9	8 4		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
POWER LEVEL (10)	0 9 5	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
T. L. Elton, Acting Supt. of Regulatory Compliance	AREA CODE 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	E A	X F M R	G 0 8 0	Y							
X	S B	R V	T 0 2 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 08/03/84 at approximately 2146 CDT, with the reactor mode switch in the run position and reactor power at 2307 MWT (approximately 95% power), Unit 1 received a reactor scram on Turbine Control Valve (TCV) fast closure subsequent to a generator load rejection. The generator load rejection occurred when a differential overcurrent relay actuated due to an internal fault (when the insulation between the #2 and #3 windings failed) in Unit auxiliary transformer 1B. The differential overcurrent relay tripped on auxiliary lockout relay which opened the generator output breakers.

No actual or potential safety consequences or implications resulted from this event. This event had no impact on any other Unit 1 system or on Unit 2. The health and safety of the public were not affected by this non-repetitive event.

The cause of these events is component failure.

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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) because of the reactor scram and Engineered Safety Feature Actuation.

On 08/03/84 at approximately 2146 CDT, with the reactor mode switch in the run position and reactor power at 2307 MWT (approximately 95% power), Unit 1 received a reactor scram on Turbine Control Valve (TCV) fast closure subsequent to a generator load rejection. The generator load rejection occurred when a differential overcurrent relay actuated due to an internal fault (when the insulation between the #2 and #3 windings failed) in Unit auxiliary transformer 1B. The differential overcurrent relay tripped on auxiliary lockout relay which opened the generator output breakers.

The transient proceeded smoothly. Reactor water level dropped to +11" (reference instrument zero) and was then recovered adequately by the reactor feedpumps. As level increased above +37", the operator tripped the "A" feedpump and lowered the feedwater controller setpoint in an unsuccessful effort to prevent the high level trip. The "B" feedpump subsequently tripped on +54" water level. The HPCI and RCIC turbines which were in their normal standby configuration also received a trip on high water level. When the high level condition was reset, the "A" feedpump was started and placed in automatic to maintain reactor water level. +11" was the lowest level recorded, hence no ECCS systems auto-initiated. Neither was there a need to manually start any ECCS or RCIC for level control.

Reactor water level did reach the level of a group 2 and 5 isolation. Both of these occurred. The isolations were reset within approximately 5 minutes after the scram, and reactor water cleanup was placed back in service. The drywell sump inboard and outboard isolation valves were reopened. Both trains of standby gas treatment auto-started as a result of the low level.

The local and control room reactor pressure instruments indicated that the peak reactor pressure during the transient was between 1100 and 1107 psig. Nine (of 11 total) SRV's lifted on high reactor pressure; however, "A" SRV (1080 psig setpoint) and "B" SRV (1100 psig setpoint) did not open. The 1% tolerance (Ref. Tech. Specs. section 2.2A.1.b) on the setpoint of these SRV's is $\pm$ 10.8 psig and $\pm$ 11 psig, respectively. The investigation revealed that pressure did not peak high enough to require "B" SRV to lift, but "A" SRV was declared inoperable for safety relief valve operation.

Since a generator load rejection initiated the reactor scram, the recirculation pump trip breakers functioned to trip the reactor recirculation pumps. The breakers were quickly reset, however, and the reactor recirculation pumps were restarted within approximately 8 minutes after the scram.

No actual or potential safety consequences or implications resulted from this event. This event had no impact on any other Unit 1 system or on Unit 2. The health and safety of the public were not affected by this non-repetitive event.

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

The cause of these events is component failure.

On further investigation, a short was found between the Phase 2 and Phase 3 windings of the 1B unit auxiliary transformer. Until it is repaired, this transformer will be isolated from the rest of the electrical system. In the interim, the 1A and 1B 4160V busses will be fed by the startup transformers.

The failure of the "A" SRV valve to operate is assumed to be due to set point drift. During the next cold shutdown, the "A" SRV top works will be replaced. The top works of the "A" SRV will be bench tested to verify set pressure. "A" SRV is considered to be inoperable for the pressure relief function only (Reference Tech. Specs. section 3.6.H.1). Unit 1 remained in hot shutdown until the subsequent startup. The "RELIEF VALVE OPERABILITY" procedure (HNP-1-3901) was performed satisfactorily on the "A" SRV to ensure the valve would operate for the ADS function.

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

IDENTIFICATION OF EACH FAILED COMPONENT

MASTER PARTS LIST NUMBER	COMPONENTS	MANUFACTURER	MODEL NUMBER
S11-S003	XFMR	General Electric	Type QA/FA/FOA-T 15/20/25 MVA
B21-F013A	RV	Target Rock	7567F (2 Stage)

Georgia Power Company
Post Office Box 439
Baxley, Georgia 31513
Telephone 912 367-7781
912 537-9444

Georgia Power

Edwin I. Hatch Nuclear Plant

August 29, 1984
GM-84-722

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. 50-321/1984-015. This report is required by 10CFR 50.73(a)(2)(iv).

for H. C. Nix
General Manager

HCN/TLE/vlt

xc: R. J. Kelly
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