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FACIL:50-244 Robert Emmet Ginna Nuclear Plant, Unit 1, Rochester G 05000244

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SUBJECT: LER 90-007-00:on 900510,inadvertent closure of A steam generator regulating valve causes reactor trip.

W/9 ltr.

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June 11, 1990

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Subject: LER 90-007, Inadvertent Closure of "A" Steam Generator Main Feedwater Regulating Valve Due To Controller Malfunction Causes A Reactor Trip On Low Steam Generator Level.
R. E. Ginna Nuclear Power Plant
Docket No. 50-244

In accordance with 10CFR50.73, License Event Report System, Item (a)(2)(iv), which requires a report of "any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)," the attached License Event Report, LER 90-007, is hereby submitted.

This event has in no way affected the public's health and safety.

Very truly yours,

Robert C. Mecredy

Robert C. Mecredy
Division Manager
Nuclear Production

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Region I
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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): R.E. Ginna Nuclear Power Plant

DOCKET NUMBER (2): 0 5 0 0 0 2 4 4 | PAGE (3): 1 OF 0 7

TITLE (4): Inadvertent Closure of "A" Steam Generator Main Feedwater Regulating Valve Due to Controller Malfunction Causes a Reactor Trip on Low Steam Generator Level

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 5	1 0	9 0	9 0	— 0 0 7 —	0 0	0 6	1 1	9 0		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 8 8

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: Wesley H. Backus
Technical Assistant to the Operations Manager

TELEPHONE NUMBER: AREA CODE 3 1 5 | 5 2 4 - 4 4 4 6

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
B	J B	L C	F 1 8 0	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES (If yes, complete EXPECTED SUBMISSION DATE) [X] NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On May 10, 1990, at 0219 EDST, with the reactor at approximately 88% full power, a reactor trip occurred from "A" Steam Generator (S/G) Low Level coincident with "A" S/G Feed Flow/Steam Flow mismatch.

The two reactor trip breakers opened as required and all shutdown and control rods inserted as designed.

The reactor trip was due to a malfunctioning "A" S/G Main Feedwater Regulating Valve Control System.

The underlying cause of the malfunctioning "A" S/G main feedwater regulating valve control system was attributed to the shorting of two high gain AC amplifier transistors in the flow controller, due to the transistor cans touching each other.

Immediate corrective action was to stabilize the plant in hot shutdown.

Subsequent action was to physically separate the two transistor cans so that they could not touch each other.

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

I. PRE-EVENT PLANT CONDITIONS

Unit startup was in progress per operating procedure O-1.2 (Plant From Hot Shutdown to Full Load) from the annual refueling and maintenance outage. Reactor power was approximately 88% full power. Plant operators were swapping All Volatile Treatment (AVT) full flow demineralizer beds on the secondary plant. The AVT full flow demineralizer bed swapping routinely places a moderate transient on the condensate and feedwater system.

II. DESCRIPTION OF EVENT

A. DATES AND APPROXIMATE TIMES OF MAJOR OCCURRENCES:

- May 10, 1990, 0219 EDST: Event date and time
- May 10, 1990, 0219 EDST: Event discovery date and time
- May 10, 1990, 0219 EDST: Control Room Operators verify both reactor trip breakers open and all control and shutdown rods inserted.
- May 10, 1990, 0253 EDST: Plant stabilized at Hot Shutdown.

B. EVENT:

On May 10, 1990, at 0219 EDST, with the reactor at approximately 88% full power, a reactor trip occurred due to low level in the "A" Steam Generator (i.e., Steam Generator level $\leq$ 30%) coincident with Steam Flow-Feed Flow (SF/FF) mismatch (i.e., Steam Flow $\geq$ 0.8E6 lbm/hr more than feedwater flow).

The Control Room Operators performed the applicable actions of Emergency Operating Procedures, E-0 (Reactor Trip Or Safety Injection) and ES-0.1 (Reactor Trip Response), and stabilized the plant. Both reactor trip breakers opened as required and all control and shutdown rods were verified inserted.

The Control Room Operators notified higher supervision and the Nuclear Regulatory Commission (NRC) of the Reactor Protection System activation from the "A" Steam Generator (S/G) Low Level coincident with SF/FF mismatch.

C. INOPERABLE STRUCTURES, COMPONENTS, OR SYSTEMS THAT CONTRIBUTED TO THE EVENT:

None.

D. OTHER SYSTEMS OR SECONDARY FUNCTIONS AFFECTED:

None.

E. METHOD OF DISCOVERY:

The event was immediately apparent due to alarms and indications in the Control Room.

F. OPERATOR ACTION:

After the reactor trip, the Control Room Operators performed the actions of Emergency Operating Procedures, E-0 (Reactor Trip Or Safety Injection) and ES-0.1 (Reactor Trip Response), and stabilized the plant.

G. SAFETY SYSTEM RESPONSES:

None.

III. CAUSE OF EVENT

A. IMMEDIATE CAUSE:

The reactor trip occurred due to "A" S/G Low Level ≤ 30% coincident with "A" S/G SF/FF mismatch ≥ 0.8E6 lbm/hr.

B. INTERMEDIATE CAUSE:

The "A" S/G Low Level coincident with "A" S/G SF/FF mismatch was due to the "A" S/G main feedwater regulating valve closing due to a controller malfunction.

C. ROOT CAUSE:

The underlying cause of the event was the FC-466A flow controller malfunction. This Foxboro model 62H controller's current signal output failed low, which was attributed to the metal cans of transistors Q3 and Q4 momentarily touching together. The collector of the transistor is electrically connected to the can for power and heat dissipation. The design of the circuit board positions these transistors in close proximity, on plastic stands. If disturbed during maintenance, the transistor can pivot on the collector, emitter and base leads and allow the metal cans to become close enough to touch. FC-466A has had maintenance done internally in the past, including capacitor and relay replacement and necessary resoldering. Vibration of the relay room during the swapping of the demineralizer beds could have provided the necessary force to cause the transistors to touch. Failure of the output current immediately causes the feedwater valve to close.

Subsequent to this failure, FC-466A exhibited a similar failure mode on June 9, 1990. This subsequent failure also caused closure of the "A" S/G main feedwater regulating valve, and resulted in a reactor trip. This event will be reported as LER 90-010.

IV. ANALYSIS OF EVENT

This event is reportable in accordance with 10CFR50.73, License Event Report System, Item (a)(2)(iv), which requires reporting of "any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)," in that the "A" S/G Low Level coincident with "A" S/G SF/FF mismatch reactor trip was an automatic actuation of the RPS.

An assessment was performed considering both the safety consequences and implications of this event with the following results and conclusions:

There were no operational or safety consequences and implications attributed to the "A" S/G Low Level coincident with "A" S/G SF/FF mismatch reactor trip because:

- o The two reactor trip breakers opened as required.
- o All control and shutdown rods inserted as designed.
- o The plant was stabilized in Hot Shutdown.

This transient was compared to the Loss of Normal Feedwater Flow Transient described in the Ginna Updated Final Safety Analysis Report (UFSAR). None of the assumptions of the UFSAR were violated during this event.

The response of the plant to this transient is bounded by the results of the UFSAR analysis. The analysis of this transient showed that the plant responded as expected to the Loss of Feedwater to the "A S/G.

During the entire event, the "A" and "B" S/Gs were always available as a heat sink due to sufficient auxiliary feedwater flow to both S/Gs and adequate steam release from both S/Gs.

Based on the above, it can be concluded that the public's health and safety was assured at all times.

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

V. CORRECTIVE ACTION

A. ACTION TAKEN TO RETURN AFFECTED SYSTEMS TO PRE-EVENT NORMAL STATUS:

- The Instrument and Control (I&C) Department did extensive troubleshooting on the "A" S/G Feedwater Control System and determined the apparent cause of the control malfunction. The "A" S/G Feedwater Control System was repaired and tested satisfactorily and returned to service.

- The high gain AC amplifier transistors Q3 and Q4 were physically separated to their original design position, which prevented them from being able to touch again.

B. ACTION TAKEN OR PLANNED TO PREVENT RECURRENCE:

- The similar controller on the "B" S/G (FC-476A) was inspected, and verified that transistors Q3 and Q4 were separated.

- The spare Foxboro model 62H controllers were inspected and verified that transistors Q3 and Q4 were separated.

- The industry will be notified via Nuclear Network, describing this failure.

- Procedures used for rework and inspection of these controllers will be changed by the end of August, 1990, to include concerns for maintaining component separation.

- Foxboro model 62H series controllers will be identified and will be inspected prior to the end of the 1991 Outage, to ensure there is proper separation of these transistors.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
R.E. Ginna Nuclear Power Plant	0 5 0 0 0 2 4 4	9 0	— 0 0 7 —	0 0	0 7 OF 0 7

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

VI. <u>ADDITIONAL INFORMATION</u>

A. FAILED COMPONENTS:

The failed component was the "A" S/G main feedwater regulating valve controller FC-466A. This controller was manufactured by Foxboro Company. The controllers Model Number is 62H-4E and Serial Number is 2208968.

B. PREVIOUS LER'S ON SIMILAR EVENTS:

A similar LER event historical search was conducted with the following results: no documentation of similar LER events with the same root cause at Ginna Station could be identified. However, LERs 85-006, 85-019, 88-003, and 88-005 were similar events with different root causes.

C. SPECIAL COMMENTS:

Subsequent to this failure, FC-466A exhibited a similar failure mode on June 9, 1990. This subsequent failure also caused closure of the "A" S/G main feedwater regulating valve, and resulted in a reactor trip. This event will be reported as LER 90-010.