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SUBJECT: LER 90-010-00:on 900609,inadvertent closure of steam generator A causes reactor trip on low steam water level.

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TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

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TELEPHONE
AREA CODE 716 546-2700

July 9, 1990

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

Subject: LER 90-010, Inadvertent Closure Of "A" Steam Generator Main Feedwater Regulating Valve Due To Controller Malfunction Causes A Reactor Trip On Low Steam Generator Water Level
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

In accordance with 10 CFR 50.73, Licensee 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 Licensee Event Report LER 90-010 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

xc: U.S. Nuclear Regulatory Commission
Region I
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Ginna USNRC Senior Resident Inspector

Cert No P340957251

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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)	DOCKET NUMBER (2)	PAGE (3)
R.E. Ginna Nuclear Power Plant	0 5 0 0 0 2 4 4	1 OF 0 6

TITLE (4) Inadvertent Closure Of "A" Steam Generator Main Feedwater Regulating Valve Due To Controller Malfunction Causes A Reactor Trip On Low Steam Generator Water 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 6	0 9	9 0	9 0	— 0 1 0 —	0 0	0 7	0 9	9 0		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 9 7

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
Wesley H. Backus Technical Assistant to the Operations Manager	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	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	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 June 9, 1990, at 0411 EDST with the Reactor at approximately 97% full power, a reactor trip occurred from "A" Steam Generator (S/G) Low Level coincident with "A" S/G Steam Flow/Feed 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 Feedwater Regulating Valve Control System was a faulty Feedwater Flow Controller. The reason for the fault is undetermined at this time.

Immediate Corrective Action was to stabilize the plant in hot shutdown.

Subsequent action was to change out the existing controller with a spare.

I. PRE-EVENT PLANT CONDITIONS

The unit was at approximately 97% steady state full power with no major activities in progress.

II. DESCRIPTION OF EVENT

A. DATES AND APPROXIMATE TIMES OF MAJOR OCCURRENCES:

- June 9, 1990, 0411 EDST: Event date and time.
- June 9, 1990, 0411 EDST: Event discovery date and time.
- June 9, 1990, 0411 EDST: Control Room operators verify both reactor trip breakers open and all control and shutdown rods inserted.
- June 9, 1990, 0421 EDST: Closed both Main Steam Isolation Valves (MSIVs) to terminate plant cooldown.
- June 9, 1990, 0431 EDST: Plant stabilized at hot shutdown.

B. EVENT:

On June 9, 1990, at 0411 EDST, with the reactor at approximately 97% full power, a reactor trip occurred. This trip was due to low level in the "A" Steam Generator (i.e. steam generator level $\leq$ 30%) coincident with steam flow - feed flow 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.

Subsequently, the Main Steam Isolation Valves (MSIVs) were closed to terminate a plant cooldown. It is believed this cooldown was partially due to cooler water being fed to the steam generators and partially due to a delay in closure of the condenser steam dump valves after the trip.

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 Steam Flow - Feedwater Flow (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. The MSIVs were closed subsequent to the trip to terminate a plant cooldown.

G. SAFETY SYSTEM RESPONSES:

None.

NRC Form 366A (9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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

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

III. CAUSE OF EVENT

A. IMMEDIATE CAUSE:

The reactor trip occurred due to "A" S/G Low Level $\leq$ 30%, coincident with "A" S/G SF/FF mismatch $\geq$ 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. Failure of the output current immediately causes the feedwater valve to close. The cause of the controller's current signal output to fail low is undetermined at this time. The I&C Shop could not duplicate or identify any malfunctions that would have caused the controller output current to fail low.

IV. ANALYSIS OF EVENT

This event is reportable in accordance with 10 CFR 50.73, Licensee 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:

- 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.

V. CORRECTIVE ACTION

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

The Instrument and Control (I&C) Shop removed the existing "A" S/G Feedwater Controller, installed the spare feedwater controller in the "A" S/G main feedwater control system, and calibrated and tested it satisfactorily.

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)

B. ACTION TAKEN OR PLANNED TO PREVENT RECURRENCE:

As the root cause of the FC-466A flow controller malfunction is still undetermined, the I&C Shop is continuing to troubleshoot the faulty controller in the I&C Shop.

Any further corrective action will depend upon determinations reached from the above investigation.

VI. ADDITIONAL INFORMATION

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 controller's model number is 62H-4E and Serial Number is 2208968.

B. PREVIOUS LERs ON SIMILAR EVENTS:

A similar LER event historical search was conducted with the following results: The underlying cause of this transient was similar to that for LER 90-007; however, the root cause appears to be different. No other 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:

None.