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FACIL:50-244 Robert Emmet Ginna Nuclear Plant, Unit 1, Rochester G 05000244
AUTH.NAME AUTHOR AFFILIATION
ST.MARTIN,J.T. Rochester Gas & Electric Corp.
MECREDY,R.C. Rochester Gas & Electric Corp.
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 93-006-00:on 931110,feedwater transient occured,due to ability controll feedwater regulating valve.Caused by LO LO steam genorator level reactor trip.New screw & nut installed in leakage arm.w/931210 ltr.

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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

December 10, 1993

U.S. Nuclear Regulatory Commission
Attn: Allen R. Johnson
Project Directorate I-3
Document Control Desk
Washington, DC 20555

Subject: LER 93-006, Feedwater Transient, Due to Loss of Ability to Control Feedwater Regulating Valve, Causes a Lo Lo Steam Generator Level Reactor Trip
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 a manual or automatic actuation of any engineered safety feature (ESF), including the reactor protection system (RPS)", the attached Licensee Event Report LER 93-006 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

xc: U.S. Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406

Ginna USNRC Senior Resident Inspector

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NRC FORM 366 (5-92)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95

LICENSEE EVENT REPORT (LER)

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ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4) Feedwater Transient, Due to Loss of Ability to Control Feedwater Regulating Valve, Causes a Lo Lo Steam Generator Level Reactor Trip

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 NAME	DOCKET NUMBER
11	10	93	93	--006--	00	12	10	93	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	N
POWER LEVEL (10)	097

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (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
	20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		(Specify in Abstract below and in Text, NRC Form 366A)
	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 (Include Area Code)
John T. St. Martin - Director, Operating Experience	(315) 524-4446

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	JB	LCV	B042	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 15 single-spaced typewritten lines) (16)

On November 10, 1993, at approximately 0848 EST, with the reactor at approximately 97% reactor power, the ability to control the "A" main feedwater regulating valve was lost. This resulted in steam generator level transients. At 0850 EST, the reactor tripped on Lo Lo level (</= 17%) in the "A" steam generator. The Control Room operators performed the actions of procedures E-0 and ES-0.1.

The underlying cause was determined to be disconnection of the "A" main feedwater regulating valve positioner feedback linkage arm from the valve actuator linkage rod, due to disengagement of the connecting screw and nut. (This event is NUREG-1220 (B) cause code.)

Corrective action was to install a new screw and nut. Corrective action to preclude repetition is outlined in Section V (B).

I. PRE-EVENT PLANT CONDITIONS

The plant was at approximately 97% steady state reactor power. The monthly surveillance test of the "A" auxiliary feedwater (AFW) pump was in progress, using procedure PT-16M-A (Auxiliary Feedwater Pump "A" - Monthly).

II. DESCRIPTION OF EVENT

A. DATES AND APPROXIMATE TIMES OF MAJOR OCCURRENCES:

- November 10, 1993, 0850 EST: Event date and time.
- November 10, 1993, 0850 EST: Discovery date and time.
- November 10, 1993, 0850 EST: Control Room operators verify both reactor trip breakers open, and all control and shutdown rods inserted.
- November 10, 1993, 0851 EST: Control Room operators manually stop both main feedwater pumps to limit a reactor coolant system cooldown.
- November 10, 1993, 0852 EST: Control Room operators manually close both main steam isolation valves to limit a reactor coolant system cooldown.
- November 10, 1993, 1045 EST: Plant stabilized at hot shutdown condition.

B. EVENT:

On November 10, 1993, surveillance test procedure PT-16M-A was initiated, at approximately 0835 EST. As part of this test, the "A" AFW pump was started. Control of the main feedwater regulating valve (MFRV) and bypass feedwater regulating valve for the "A steam generator (S/G) was shifted to the "Manual" mode, and the Control Room operator slightly closed the "A" MFRV. The "A" MFRV initially started closing, and then appeared to drift open, based on indications of increased feedwater flow to the "A" S/G.

Despite the efforts of the Control Room operators to close the "A" MFRV, the "A" MFRV continued to drift open. The Advanced Digital Feedwater Control System (ADFCS) responded as designed, and shifted all feedwater regulating valves (for both S/Gs) to "Manual". The Control Room operators terminated PT-16M-A and turned off the "A" AFW pump. "A" S/G level continued to increase, until it reached the high level override setpoint of 67%. The "A" MFRV closed as designed at 67% level. All feedwater flow was now directed to the "B" S/G. The "B" S/G level also reached the 67% high level override setpoint, and the "B" MFRV closed.

The MFRVs reopened as designed when S/G levels decreased to less than 67%. Due to the positioner feedback linkage failure, the Control Room operators had lost the ability to control "A" S/G level. The "A" S/G level decreased to < 17%, resulting in a reactor trip on S/G Lo Lo level, at 0850 EST.

The Control Room operators performed the immediate actions of Emergency Operating Procedure E-0 (Reactor Trip or Safety Injection), and transitioned to Emergency Operating Procedure ES-0.1 (Reactor Trip Response) when it was verified that both reactor trip breakers were open, all control and shutdown rods were inserted, and safety injection was not actuated or required. During performance of E-0, the Control Room operators noted the continuing RCS cooldown and increasing S/G levels, and referred to Functional Restoration procedure FR-H.3 (Response to Steam Generator High Level). The operators verified that the AFW pumps had started, as designed, on the Lo Lo S/G level. Using the guidance of FR-H.3, they manually stopped both main feedwater pumps. In addition, both main steam isolation valves (MSIVs) were manually closed by the Control Room operators. These actions mitigated the RCS cooldown.

The plant was subsequently stabilized in hot shutdown, using procedure O-2.2 (Plant Shutdown From Hot Shutdown to Cold Shutdown) at approximately 1045 EST.

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

None

D. OTHER SYSTEMS OR SECONDARY FUNCTIONS AFFECTED:

None

E. METHOD OF DISCOVERY:

This event was immediately apparent due to the loss of ability to control feedwater flow to the "A" S/G. The reactor trip 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). The main feedwater pumps were manually stopped and the MSIVs were manually closed to limit further RCS cooldown. The plant was stabilized at hot shutdown. Subsequently, the Control Room operators notified higher supervision and the Nuclear Regulatory Commission per 10CFR50.72, Non-Emergency, 4 Hour Notification at approximately 1030 EST.

G. SAFETY SYSTEM RESPONSES:

None

III. CAUSE OF EVENT

A. IMMEDIATE CAUSE:

The reactor trip was due to "A" S/G Lo Lo level (</= 17%).

B. INTERMEDIATE CAUSE:

The "A" S/G Lo Lo level (</= 17%) was due to decreased feedwater flow to the "A" S/G, caused by loss of ability to control the "A" MFRV.

C. ROOT CAUSE:

The underlying cause of the loss of ability to control the "A" MFRV was the disconnection of the positioner feedback linkage arm from the valve actuator linkage rod on the "A" MFRV, due to disengagement of the connecting screw and nut. (This event is NUREG-1220 (B) cause code, Design, Manufacturing, Construction/ Installation).

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 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 "A" S/G Lo Lo level reactor trip was an automatic actuation of the RPS. The closures of the MFRVs at 67% S/G levels were also automatic actuations of an ESF component.

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

There were no safety consequences or implications attributed to the reactor trip because:

- The two reactor trip breakers opened as required.
- All control and shutdown rods inserted as designed.
- The plant was stabilized at hot shutdown.

The Ginna Updated Final Safety Analysis Report (UFSAR) transient, as described in Chapter 15.2.6, "Loss of Normal Feedwater", describes a condition where the reactor trips on Lo Lo S/G level. This UFSAR transient was reviewed and compared to the plant response for this event. The UFSAR transient is a complete loss of Main Feedwater (MFW) at full power, with only one AFW pump available one (1) minute after the loss of MFW, and secondary steam relief (i.e. decay heat removal) through the safety valves only. The protection against a loss of MFW includes the reactor trip on Lo Lo S/G level and the start of the AFW pumps. These protection features operated as designed.

Based on the above evaluation, the plant transient of November 10, 1993 is bounded by the UFSAR Safety Analysis assumptions.

There were no operational or safety consequences or implications attributed to the closure of the MFRVs at 67% S/G level because:

- o The valve closure signals occurred at the required S/G level.
- o The plant was quickly stabilized to mitigate any consequences of the event.
- o As the valves closed as designed, the assumptions of the UFSAR for steam line break were met.

Technical Specifications (TS) were reviewed in respect to the post trip review data. The following are the results of that review:

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

- o Following the reactor trip, PRZR water level decreased to below 0%, due to a moderate RCS cooldown. This cooldown occurred during the post trip recovery period. This cooldown was bounded by the plant accident analysis, and did not exceed the TS limit of 100 degrees F per hour. Additional mitigation was provided by closing the MSIVs and stopping the main feedwater pumps. TS 3.1.1.5 states, in part, that when the RCS temperature is at or above 350 degrees F, the pressurizer water level will be maintained between 12% and 87% of level span to be considered operable. TS 3.1.1.5 also states, in part, that if the pressurizer is inoperable due to water level, restore the pressurizer to operable status within six (6) hours or have the reactor below an RCS temperature of 350 degrees F and the RHR system in operation within an additional six (6) hours. Pressurizer water level recovered to greater than 12% level within ten (10) minutes, well before the six (6) hour action statement.

- o Both S/G levels decreased to less than 0% following the reactor trip. This is an expected observed transient. TS 4.3.5.5 states that in order to demonstrate that a reactor coolant loop is operable, the S/G water level shall be >/= 16%. Thus, both coolant loops were inoperable, even though both loops were still in operation and performing their intended function of decay heat removal. Both S/Gs were available as a heat sink, and sufficient AFW flow was maintained for adequate steam release from both S/Gs. TS 3.1.1.1(c) states, in part, that except for special tests, when the RCS temperature is at or above 350 degrees F with the reactor power less than or equal to 130 MWT (8.5%), at least one reactor coolant loop and its associated S/G and reactor coolant pump shall be in operation. Both reactor coolant loops were in operation, but the S/Gs were inoperable due to level indication. Both loops were returned to operable status. "A" S/G level was restored to > 16% within one (1) minute, and "B" S/G level was restored to > 16% in approximately ten (10) seconds.

- o Condensate Storage Tank (CST) level decreased to less than 22,500 gallons of water, due to a malfunction of the condensate makeup and reject valves. The malfunction caused the CSTs to fill the main condenser hotwell. TS 3.4.3 states, in part, that with the RCS temperature at or above 350 degrees F, one or more CSTs with a minimum of 22,500 gallons of water, shall be operable as a source of auxiliary feedwater. With the CSTs inoperable, within four (4) hours either restore the CSTs to operable status, or be in at least hot shutdown within the following six (6) hours and at an RCS temperature less than 350 degrees F within the following six (6) hours. The reactor was already at hot shutdown, and the CSTs were restored to operable status within approximately fifty (50) minutes, well before the twelve (12) hour action statement.

Based on the above and a review of post trip data and past plant transients, it can be concluded that the plant operated as designed and that there were no unreviewed safety questions and 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:

- o The "A" MFRV Bailey positioner was replaced. (Refer to LER 92-006, Rev. 1, Docket No. 50-244.) The newly installed positioner was reattached to the valve actuator linkage rod using a vendor-recommended screw with an elastic stop nut.

**LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION**

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

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

- The "B" MFRV positioner was inspected, and rework of the positioner was performed. This included installation of a vendor-recommended screw with an elastic stop nut.

- The "A" and "B" MFRVs were repacked. (Refer to LER 92-006, Rev. 1, Docket No. 50-244.)

- Linkage connections for other accessible valves with Bailey positioners were inspected to ensure satisfactory integrity of the connections.

- The malfunction of the condensate makeup and reject valves was caused by a fitting leak which was associated with the common demand signal shared by both valves. The fitting was tightened, and both valves were verified to respond correctly to control main condenser hotwell level.

B. ACTION TAKEN OR PLANNED TO PREVENT RECURRENCE:

- Applicable procedures will be upgraded to ensure Bailey vendor manual information concerning feedback linkage arm connections is addressed.

- Training will be conducted to enhance the knowledge of appropriate personnel on the general topic of fasteners, and stressing linkage arm connections, specifically.

- Rework of positioner feedback linkage arm connections, for all other valves with Bailey positioners, will be accomplished. This will include installation of vendor-recommended screws and elastic stop nuts.

VI. ADDITIONAL INFORMATION

A. FAILED COMPONENTS:

The failed component was the "A" MFRV positioner feedback linkage arm connection. This assembly is a Bailey pneumatic positioner, model AV 112100, manufactured by Bailey Controls Inc.

B. PREVIOUS LERs 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 Nuclear Power Plant could be identified. However, LERs 88-005, 90-007, and 90-010 were similar events with different root causes.

C. SPECIAL COMMENTS:

LER 92-006, Rev. 1, identified three corrective actions related to MFRVs: replace the "A" MFRV actuator with a rebuilt actuator, replace the "A" MFRV positioner, and repack both MFRVs. Two of these corrective actions were accomplished in response to this event (LER 93-006). Post-maintenance testing of MFRVs, and subsequent satisfactory operation of both MFRVs, have demonstrated that replacement of the "A" MFRV actuator is no longer required.