ACCELERATED DISTRIBUTION DEMONSTRATION SYSTEM

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9406030044 DOC.DATE: 94/05/27 NOTARIZED: NO DOCKET #
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 94-007-00:on 940427,feedwater transient occurred due to loss of ability to control feedwater regulating valve, causing lo lo SG level reactor trip.Caused by improperly secured stroke adjust set screw.W/940527 ltr.

DISTRIBUTION CODE: IE22T COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 10
TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

NOTES:License Exp date in accordance with 10CFR2,2.109(9/19/72). 05000244

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD1-3 PD	1	1	JOHNSON,A	1	1
INTERNAL:	AEOD/DOA	1	1	AEOD/DSP/TPAB	1	1
	AEOD/ROAB/DSP	2	2	NRR/DE/EELB	1	1
	NRR/DE/EMEB	1	1	NRR/DORS/OEAB	1	1
	NRR/DRCH/HHFB	1	1	NRR/DRCH/HICB	1	1
	NRR/DRCH/HOLB	1	1	NRR/DRIL/RPEB	1	1
	NRR/DRSS/PRPB	2	2	NRR/DSSA/SPLB	1	1
	NRR/DSSA/SRXB	1	1	REG FILE 02	1	1
	RES/DSIR/EIB	1	1	RGN1 FILE 01	1	1
EXTERNAL:	EG&G BRYCE,J.H	2	2	L ST LOBBY WARD	1	1
	NRC PDR	1	1	NSIC MURPHY,G.A	1	1
	NSIC POORE,W.	1	1	NUDOCS FULL TXT	1	1

R I D S / A D D S

R I D S / A D D S

NOTE TO ALL "RIDS" RECIPIENTS:

PLEASE HELP US TO REDUCE WASTE! CONTACT THE DOCUMENT CONTROL DESK, ROOM P1-37 (EXT. 20079) TO ELIMINATE YOUR NAME FROM DISTRIBUTION LISTS FOR DOCUMENTS YOU DON'T NEED!

FULL TEXT CONVERSION REQUIRED
TOTAL NUMBER OF COPIES REQUIRED: LTTR 27 ENCL 27

ROCHESTER GAS AND ELECTRIC CORPORATION • 89 EAST AVENUE, ROCHESTER N.Y. 14649-0001

ROBERT C. MECREDY
Vice President
Ginna Nuclear Production

TELEPHONE
AREA CODE 716 546-2700

May 27, 1994

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

Subject: LER 94-007, 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 94-007 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

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

Ginna USNRC Senior Resident Inspector

IE22
1/1

NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

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.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
R.E. Ginna Nuclear Power Plant	05000244	1 OF 9

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
04	27	94	94	--007--	00	05	27	94	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	N
POWER LEVEL (10)	045

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 April 27, 1994, at approximately 1407 EDST, with the reactor at approximately 45% reactor power, the ability to control the "A" main feedwater regulating valve was lost. At 1410 EDST, 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 an improperly secured stroke adjust set screw for the valve positioner signal diaphragm assembly for the "A" main feedwater regulating valve. This event is NUREG-1022 Cause Code (B), "Design, Manufacturing, Construction/Installation."

Immediate corrective action was to install a new valve positioner of a previous design. Corrective action to preclude repetition is outlined in Section V (B).

I. PRE-EVENT PLANT CONDITIONS

The plant achieved full power operation on April 23, 1994, after completion of the 1994 annual refueling and maintenance outage. On April 26, 1994, the plant was manually shut down to repair a small steam leak on an instrument fitting on the high pressure (HP) turbine. Due to stability problems with control of feedwater flow to the "A" Steam Generator (S/G) at steady state conditions, the valve positioner for the "A" main feedwater regulating valve (MFRV) was replaced during this brief shutdown.

On April 27, 1994, a load increase was in progress, controlled by Plant Operating Procedure O-1.2, "Plant Startup from Hot Shutdown to Full Load". The plant was at approximately 45% reactor power. Preparations were being made to start a second main feedwater pump, per System Operating Procedure T-4.F, "Restoring 1B Feedwater Pump to Service After Maintenance or Power Reduction".

II. DESCRIPTION OF EVENT

A. DATES AND APPROXIMATE TIMES OF MAJOR OCCURRENCES:

- o April 27, 1994, 1410 EDST: Event date and time.
- o April 27, 1994, 1410 EDST: Discovery date and time.
- o April 27, 1994, 1410 EDST: Control Room operators verify both reactor trip breakers open, and all control and shutdown rods inserted.
- o April 27, 1994, 1412 EDST: Control Room operators manually stop the operating main feedwater pump to limit a reactor coolant system cooldown.
- o April 27, 1994, 1416 EDST: Control Room operators manually close both main steam isolation valves to limit a reactor coolant system cooldown.
- o April 27, 1994, 1440 EDST: Plant stabilized at hot shutdown condition.

B. EVENT:

On April 27, 1994, a load increase was in progress, following the brief plant shutdown on April 26, 1994. Plant Operating Procedure O-1.2 was being followed to control the load increase. Only one main feedwater pump was in operation, and preparations were being made to start the second main feedwater pump and to continue with the load increase.

On April 27, 1994, at approximately 1407 EDST, the Control Room operators noticed a slight decrease in feedwater flow to the "A" S/G. They attempted to manually increase feedwater flow, but the "A" MFRV did not respond to the demand signal to open the valve from the Main Control Board. At approximately 1408 EDST, Main Control Board annunciator G-22, "ADFCS System Trouble" alarmed, due to a deviation between the demand signal and actual position of the "A" MFRV. Level continued to decrease in the "A" S/G. The Shift Supervisor ordered a rapid load reduction, in an attempt to decrease the need for feedwater flow. Within two minutes, power had been decreased by approximately 15%, and actual feedwater flow being delivered to the "A" S/G exceeded steam flow. However, level in the "A" S/G decreased to < 17%, resulting in a reactor trip on S/G Lo Lo level, at 1410 EDST.

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 ES-0.1, the Control Room operators noted that an anticipated reactor coolant system (RCS) cooldown was occurring, and manually stopped the operating main feedwater pump. In addition, both main steam isolation valves (MSIVs) were manually closed by the Control Room operators. These actions mitigated the RCS cooldown.

During this event, pressurizer (PRZR) level decreased below the setpoint for letdown isolation, closing the letdown isolation valves and deenergizing the PRZR heaters. After PRZR level was restored above the setpoint, the Control Room Foreman directed that letdown and PRZR heaters be restored to service. The plant was subsequently stabilized in hot shutdown (at approximately 1440 EDST) using Plant Operating Procedures O-3, "Hot Shutdown with Xenon Present", and O-2, "Plant Shutdown".

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 apparent due to Main Control Board indications of 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 operating main feedwater pump was 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.

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%), caused by decreased feedwater flow to the "A" S/G.

B. INTERMEDIATE CAUSE:

The decreased feedwater flow to the "A" S/G was due to loss of ability to control the "A" MFRV, caused by the valve positioner for the "A" MFRV not responding to the demand open signal.

C. ROOT CAUSE:

The underlying cause of the valve positioner for the "A" MFRV not properly responding to a change in input demand signal from the controller on the Main Control Board was an improperly secured stroke adjust set screw. The set screw was found in a backed out condition in the valve position signal diaphragm assembly for the "A" MFRV. It is postulated that this stroke adjust allen head set screw did not have adequate thread sealant to prevent it from backing out of the signal diaphragm assembly cover when subjected to vibration. With the set screw backed out, the signal diaphragm was restricted from responding to an input demand signal to open the MFRV. Note that the set screw is a factory-set adjustment. This particular set screw backed out when subjected to less than twelve hours of operation, and this failure mode has not occurred in other valve positioners that have accumulated thousands of hours of operation. This event is NUREG-1022 Cause Code (B), "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.

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

o 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.

o 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 auxiliary feedwater (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 April 27, 1994, is bounded by the UFSAR Safety Analysis assumptions.

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

* Following the reactor trip, PRZR water level decreased to approximately 12.5 % 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 pump. TS 3.1.1.5 states, in part, that when the RCS temperature is at or above 350 degrees F, at least 100 KW of PRZR heaters will be operable. TS 3.1.1.5 also states, in part, that if the PRZR is inoperable due to heaters, restore the PRZR to operable status within six (6) hours. PRZR water level was restored above the setpoint for letdown isolation within two (2) minutes, restoring the PRZR heaters to operable status, well before the six (6) hour action statement.

* Both S/G levels decreased following the reactor trip. "A" S/G level decreased to < 0%, and "B" S/G level decreased to < 14%. This is an expected 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 restored to operable status when S/G levels were restored to >/= 16% ("B" S/G level in less than four (4) minutes, and "A" S/G level in approximately ten (10) minutes).

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, that there were no unreviewed safety questions, and that the public's health and safety was assured at all times.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
R.E. Ginna Nuclear Power Plant	05000244	94	-- 007 --	00	8 OF 9

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

V. CORRECTIVE ACTION

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

o The valve positioner for the "A" MFRV (Bailey Model AV112100, or "Type AV1") had been replaced on April 26, 1994, with another Type AV1 positioner. This specific positioner failed in less than twelve hours of use and was replaced after this event (on April 28, 1994) with a Bailey Model 5321030A10 (or "5321030") valve positioner.

The Bailey Model 5321030 is the original model of valve positioner for the MFRV application and had operated successfully since plant startup in 1969. This positioner model was changed out in 1991 as part of EWR 4773, which installed the Advanced Digital Feedwater Control System (ADFCS). The Type AV1 positioners have had a history of reliability and stability concerns in this MFRV application.

o The valve positioner for the "B" MFRV was also replaced on April 28, 1994, for the reasons discussed above.

o Valve ramp and step change diagnostic testing was performed for both MFRVs to verify proper valve positioning and response.

B. ACTION TAKEN OR PLANNED TO PREVENT RECURRENCE:

o The positioners for both MFRVs were replaced with Model 5321030 positioners as discussed above.

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

VI. ADDITIONAL INFORMATION

A. FAILED COMPONENTS:

The failed component was the stroke adjust allen head set screw for the valve positioner signal diaphragm assembly for the "A" MFRV. The positioner is a Model AV112100 positioner, 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, LER 93-006 was a similar event, in that there was loss of ability to control the "A" MFRV; and LERs 88-005, 90-007, and 90-010 were similar events with different root causes.

C. SPECIAL COMMENTS:

None