CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9609270247 DOC.DATE: 96/09/19 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

VISSING.G.S.

SUBJECT: LER 96-012-00:on 960820,feedwater transient occurred,due to closure of feedwater regulating valve,causing lo lo steam generator level reactor trip. SGs were restored & missing screw in 1/p-476 was replaced. W/960919 ltr.

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

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ROBERT C. MECREDY
Vice President
Nuclear Operations

September 19, 1996

U.S. Nuclear Regulatory Commission
Document Control Desk
Attn: Guy S. Vissing
Project Directorate I-1
Washington, D.C. 20555

Subject: LER 96-012, Feedwater Transient, Due to Closure of Feedwater Regulating Valve, Causes a Lo Lo Steam Generator Level Reactor Trip
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Vissing:

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 96-012 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: Mr. Guy S. Vissing (Mail Stop 14C7)
PWR Project Directorate I-1
Washington, D.C. 20555

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

Ginna Senior Resident Inspector

IE22
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NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

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TITLE (4)
Feedwater Transient, Due to Closure of 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
08	20	96	96 --	012 --	00	09	19	96	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	1
POWER LEVEL (10)	100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)	X	50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
John T. St. Martin - Technical Assistant	(716) 771-3641

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
A	SJ	TD	R369	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 August 20, 1996, at approximately 1442 EDST, with the plant in Mode 1 at approximately 100% steady state reactor power, the "B" main feedwater regulating valve went to the fully closed position. At 1443 EDST, the reactor tripped on Lo Lo level in the "B" Steam Generator. The Control Room operators performed the actions of procedures E-0 and ES-0.1. Following the reactor trip, all systems operated as designed, and the reactor was stabilized in Mode 3.

The underlying cause of the closure of the "B" main feedwater regulating valve was determined to be a loss of electrical continuity, caused by a missing screw in the current-to-pressure transducer for the "B" main feedwater regulating valve.

Corrective action was to replace the missing screw.

This event is NUREG-1022 Cause Code (A).

Corrective action to prevent recurrence is outlined in Section V.B.

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

I. PRE-EVENT PLANT CONDITIONS:

On August 20, 1996, the plant was in Mode 1 at approximately 100% steady state reactor power. At approximately 1442 EDST, the Control Room operators received several Main Control Board Annunciator alarms. These alarms indicated that there was a problem in the Advanced Digital Feedwater Control System (ADFCS), and that a main feedwater regulating valve (MFRV) was now in manual control. The Control Room operators observed that the "B" MFRV had closed and feedwater flow to the "B" SG was not adequate for 100% steady state power operation. The Control Room operators responded to these alarms and attempted to restore adequate flow to the "B" Steam Generator (SG) by opening the MFRV. Attempts were unsuccessful, and water level in the "B" SG was rapidly decreasing due to the loss of feedwater flow to that SG.

II. DESCRIPTION OF EVENT:

A. DATES AND APPROXIMATE TIMES OF MAJOR OCCURRENCES:

- August 20, 1996, 1442 EDST: Valve positioner failure.
- August 20, 1996, 1443 EDST: Event date and time.
- August 20, 1996, 1443 EDST: Discovery date and time.
- August 20, 1996, 1444 EDST: Control Room operators verify both reactor trip breakers open and verify all control and shutdown rods inserted.
- August 20, 1996, 1450 EDST: Control Room operators manually close both main steam isolation valves to limit a reactor coolant system cooldown.
- August 20, 1996, 1453 EDST: Control Room operators manually stop both main feedwater pumps to limit a reactor coolant system cooldown.
- August 20, 1996, 1545 EDST: Plant is stabilized in Mode 3.

B. EVENT:

On August 20, 1996, at approximately 1443 EDST, the plant was in Mode 1 at approximately 100% steady state reactor power. Feedwater flow to the "B" SG was inadequate, and water level in the "B" SG was rapidly decreasing. When the "B" SG level was at 20% (and still decreasing), the Control Room Foreman ordered a manual reactor trip. Before the Control Room operators performed a manual reactor trip, the reactor automatically tripped on Lo Lo level in the "B" SG (< 17%).

The Control Room operators performed the immediate actions of Emergency Operating Procedure E-0, "Reactor Trip or Safety Injection". They 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. (They also verified that the auxiliary feedwater (AFW) pumps had started as designed, on the Lo Lo SG level.)

During the performance of ES-0.1, the Control Room operators noted a continuing reactor coolant system (RCS) cooldown and increasing level in the "A" SG. They referred to Functional Restoration Procedures FR-H.3, "Response to Steam Generator High Level" and FR-H.5, "Response to Steam Generator Low Level", due to the high SG level in the "A" SG and the low SG level in the "B" SG. Using the guidance of procedure FR-H.3, they initiated actions to secure all feedwater flow to the "A" SG.

Level in the "A" SG continued to increase due to the "B" MFRV valve positioner failure and increased above the setpoint for main feedwater isolation (>85%). Main feedwater isolation of the "A" SG occurred at that time. Actions were taken per plant procedures to prevent SG overfill and mitigate the increase in "A" SG level. The feedwater isolation signal cleared within five (5) minutes.

Due to the RCS cooldown that was occurring, the Control Room operators manually closed both main steam isolation valves (MSIVs) at approximately 1450 EDST and manually stopped both main feedwater (MFW) pumps at approximately 1453 EDST. These actions mitigated the RCS cooldown.

The Control Room operators received Main Control Board Annuciator K-3 (AMSAC Actuation) at approximately 1446 EDST (due to 3/4 FF channels <25%) and verified that the turbine-driven AFW pump had started due to a signal from the ATWS Mitigation System Actuation Circuitry (AMSAC). The plant was subsequently stabilized in Mode 3 at approximately 1545 EDST.

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 Main Control Board indication of inadequate feedwater flow to the "B" SG. The reactor trip was immediately apparent due to plant response and alarms and indications in the Control Room.

F. OPERATOR ACTION:

After the reactor trip, the Control Room operators performed the appropriate actions of Emergency Operating Procedures E-0 and ES-0.1. Feedwater flow to the "A" SG was stopped to mitigate the increase in "A" SG level. The MSIVs were manually closed and both MFW pumps stopped to limit further RCS cooldown. Appropriate actions were taken to restore level in the "B" SG and to minimize level increase in the "A" SG.

The setting for lifting of the SG atmospheric relief valves (ARV) was lowered from 1050 PSIG to minimize a subsequent RCS heatup (and prevent PRZR overpressure). The plant was stabilized in Mode 3.

Subsequently, the Control Room operators notified higher supervision and the NRC per 10 CFR 50.72 (b) (2) (ii), non-emergency four hour notification, at approximately 1755 EDST on August 20, 1996.

G. SAFETY SYSTEM RESPONSES:

All safeguards equipment functioned properly. Both motor-driven AFW pumps started when "B" SG level decreased below 17% after the reactor trip. The turbine-driven AFW pump started as per design, due to a starting signal from AMSAC. Main feedwater isolation occurred on high level in the "A" SG (i.e., >85% narrow range level).

III. CAUSE OF EVENT:

A. IMMEDIATE CAUSE:

The immediate cause of the reactor trip was due to "B" SG Lo Lo level (< 17%), caused by inadequate feedwater flow to the "B" SG.

B. INTERMEDIATE CAUSE:

The intermediate cause of the inadequate feedwater flow to the "B" SG was the closure of the "B" MFRV, caused by the current-to-pressure (I/P) transducer not responding to the input demand signal. This resulted in loss of input demand signal to the "B" MFRV valve positioner.

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C. ROOT CAUSE:

The underlying cause of the loss of input demand signal to the "B" MFRV valve positioner was a loss of electrical continuity from the terminal block to the circuit board on the terminal block inside the current-to-pressure transducer (I/P-476) that supplies air pressure to the "B" MFRV.

This loss of continuity was the result of a missing screw which caused an unreliable input signal connection, resulting in loss of the signal to the transducer, and caused the output air signal to decrease to minimum. On minimum air pressure, the MFRV goes fully closed.

The basic design of the Rosemount Model 3311 I/P transducer (I/P-476) is significantly different when compared to other instrumentation. The mounting of the circuit board to the terminal block is unique, and special instructions or guidance were absent in the manufacturer's technical manual. Four screws are installed in the terminal block in these Rosemount transducers. Two are used for field wire connections, and two are used to hold down the terminal block connection board.

This event is NUREG-1022 Cause Code (A), "Personnel Error". A Human Performance Enhancement System (HPES) evaluation was initiated for this event. The HPES evaluation concluded that, in the event a screw was discovered missing on the terminal block for these transducers, it had been a previously accepted practice for Instrument and Control (I&C) technicians not to replace the screw, and to reconnect any wiring onto a different screw, as long as it was the same electrical point, same terminal block, and same terminal number. This practice does not affect electrical continuity for transducers of a different design, since no screws on the terminal block hold down the terminal block connection board. However, on Rosemount transducers, all four screws are required for their specific function.

This error was a cognitive error, in that the I&C technicians did not understand the detailed function of each screw, and did not recognize that their practice could cause unreliable connections in the transducer. This error was not contrary to any approved procedures and is not covered in detail in any procedure. There are no unusual characteristics of the locations for any of these transducers.

The failure of the "B" MFRV I/P transducer meets the NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants", definition of a "Maintenance Preventable Functional Failure".

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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 a manual or automatic actuation of any engineered safety feature (ESF), including the reactor protection system (RPS)". The "B" SG Lo Lo level reactor trip was an automatic actuation of the RPS, and MFW isolation and AFW pump starts are 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 operational or 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 in Mode 3.

The Ginna Station Improved Technical Specifications (ITS) Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) were reviewed with respect to the post trip review data. The following are the results of that review:

- PRZR pressure decreased below 2205 PSIG during the transient after the reactor trip. During this time a thermal power step >10% occurred due to the reactor trip, which is within the limits of ITS LCO 3.4.1. Therefore, compliance with ITS was maintained. The RCS temperature DNB limit (577.5 degrees F) was not approached. Additional mitigation was provided by closing the MSIVs and stopping the MFW pumps. Minimum PRZR pressure was approximately 2092 PSIG.

- After the reactor trip, the RCS cooled down to approximately 539 degrees F and was subsequently stabilized at 547 degrees F. The cooldown was within the limits of ITS LCO 3.4.3. In addition, the required shutdown margin was maintained at all times during the RCS cooldown.

- Both SG levels decreased following the reactor trip. "B" SG level decreased below 16% indicated narrow range level. SR 3.4.5.2 states that in order to demonstrate that a reactor coolant loop is operable, the SG water level shall be >/= 16%. Thus, the "B" coolant loop was inoperable, even though it was still in operation and performing its intended function of decay heat removal.

  Both SGs were available as a heat sink, and sufficient AFW flow was maintained for adequate steam release from both SGs. The "B" coolant loop was restored to operable status when SG level was restored to >/= 16%, in approximately thirty-five (35) minutes. This is within the limits of ITS LCO 3.4.5 ACTION A.

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- The Ginna Station 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 SG 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 AFW pumps 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 SG level and the start of AFW pumps. These protection features operated as designed. Based on the above evaluation, the plant transient of August 20, 1996, is bounded by the UFSAR Safety Analysis assumptions.

- The UFSAR transient, as described in Chapter 15.1.2, "Increase in Feedwater Flow at Full Power", describes a condition where the automatic operation of the main feedwater isolation provided protection from potential SG overfill and damage to the turbine and steam piping due to water carryover. Prudent operator action provided the necessary action to reduce SG level. The high level in the "A" SG that resulted during the transient is bounded by the UFSAR Safety Analysis assumptions.

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.

V. CORRECTIVE ACTION:

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

- The SGs were restored to operable status when SG level in the "B" SG increased above 16% level, by addition of AFW. Subsequently, levels were restored to their normal operating levels.

- The missing screw in I/P-476 was replaced.

- Both MFRVs were operated fully open and fully closed from the Main Control Board hand controller to verify proper valve positioning and response.

B. ACTION TAKEN OR PLANNED TO PREVENT RECURRENCE:

- There are six (6) Rosemount Model 3311 I/P transducers in use at Ginna Station. All 6 were inspected. In addition to the missing screw for I/P-476, a broken field wire connection screw was found in I/P-466 (for the "A" MFRV), and the field wire was landed on one of the terminal board screws. A terminal board screw was missing in the transducer for the "B" SG atmospheric relief valve, and was later found in a nearby conduit. The configurations of all Rosemount transducers were restored to approved configurations. I&C technicians have been made aware of the unusual arrangement of the terminal block screws in Rosemount transducers.

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- Calibration procedures for all 6 Rosemount transducers have been changed to ensure that all four screws are in place and wires are landed on the correct terminal points.

- Nuclear Training Work Requests (NTWR) have been written to incorporate the lessons learned into the I&C training program.

VI. ADDITIONAL INFORMATION:

A. FAILED COMPONENTS:

The failed component (I/P-476) was a Rosemount Model 3311 I/P transducer.

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 93-006 (due to connecting screw for linkage feedback arm) and 94-007 (due to set screw backing out of valve position signal diaphragm assembly) were similar events, in that there was a loss of ability to control a MFRV which resulted in a reactor trip. LERs 85-006, 88-003, 88-005, 90-007, 90-010, 92-002, and 92-003 were similar events (reactor trip from Lo SG level) with different root causes.

C. SPECIAL COMMENTS:

None