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FACIL:50-244 Robert Emmet Ginna Nuclear Plant, Unit 1, Rochester G 05000244
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BACKUS,W.H. Rochester Gas & Electric Corp.
MECREDY,R.C. Rochester Gas & Electric Corp.
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 90-019-00:on 901221,reactor tripped due to lo lo level in steam generator A.Caused by main feedwater pump A tripping due to feed pump seal water low differential pressure.Procedures will be revised.W/910121 ltr.

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

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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

January 21, 1991

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

Subject: LER 90-019, Lo Lo Level in "A" Steam Generator, During Plant Startup, Due to Main Feedwater Pump Trip, Causes a 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 manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)", the attached Licensee Event Report LER 90-019 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

Cert No P249194216

IE22

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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) Lo Lo Level in "A" Steam Generator, During Plant Startup, Due to Main Feedwater Pump Trip, Causes a 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 NAMES	DOCKET NUMBER(S)
1 2	2 1	9 0	9 0	— 0 1 9 —	0 0	0 1	2 1	9 1		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 1 6

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 (AREA CODE)
Wesley H. Backus Technical Assistant to the Operations Manager	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

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 December 21, 1990 at 1237 EST, with the reactor at approximately 16% full power, a reactor trip occurred due to Lo Lo level (≤ 17%) in the "A" Steam Generator.

The Control Room operators immediately performed the appropriate actions of E-0 (Reactor Trip or Safety Injection) and ES-0.1 (Reactor Trip Response). Both main steam isolation valves were subsequently closed to limit an RCS cooldown and the plant was stabilized in hot shutdown.

The intermediate cause of the event was the "A" main feedwater pump tripping due to feed pump seal water low differential pressure caused by a condensate low header pressure transient.

The underlying cause of the event was a deficiency in the operating philosophy for the proper number of condensate pumps running during low power conditions.

Corrective action will be to change the appropriate procedures involved, to be consistent with the new operating philosophy.

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

I. PRE-EVENT PLANT CONDITIONS

The plant was in the process of starting up subsequent to the Dropped Rod Event of 12/20/90 (discussed in LER 90-018). The reactor was at approximately 16% full power with the turbine latched and rolling to 600 RPM.

II. DESCRIPTION OF EVENT

A. DATES AND APPROXIMATE TIMES OF MAJOR OCCURRENCES:

- o December 21, 1990, 1237 EST: Event Date and Time
- o December 21, 1990, 1237 EST: Discovery Date and Time
- o December 21, 1990, 1237 EST: Control Room operators verify both reactor trip breakers open, and all control and shutdown rods inserted.
- o December 21, 1990, 1243 EST: Control Room operators close both main steam isolation valves (MSIVs) to limit a reactor coolant system (RCS) cooldown.
- o December 21, 1990, 1257 EST: Plant stabilized at hot shutdown.

B. EVENT:

On December 21, 1990 at 1237 EST, with the reactor at approximately 16% full power, the reactor tripped due to lo lo level ($\leq$ 17%) in the "A" Steam Generator (S/G).

The Control Room operators immediately performed the immediate actions of Emergency Operating Procedure E-O (Reactor Trip or Safety Injection) and transitioned to ES-0.1 (Reactor Trip Response) when it was verified that safety injection was not actuated or required.

Both MSIVs were subsequently closed at 1243 EST to limit an RCS cooldown. The closing of both MSIVs mitigated the RCS cooldown and the plant was subsequently stabilized in hot shutdown.

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

The Control Room operators notified higher supervision and the Nuclear Regulatory Commission (NRC) of the event.

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:

Subsequent to the reactor trip, the Control Room operators performed the appropriate actions of Emergency Operating Procedures E-0 (Reactor Trip or Safety Injection) and ES-0.1 (Reactor Trip Response) to stabilize the plant. The MSIVs were closed approximately six (6) minutes after the trip to prevent further plant cooldown.

G. SAFETY SYSTEM RESPONSES:

None.

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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 lo lo level (≤ 17%) in the "A" S/G.

B. INTERMEDIATE CAUSE:

The lo lo level (≤ 17%) in the "A" S/G was due to the tripping of the "A" main feedwater pump.

The "A" main feedwater pump tripped due to feed pump seal water low differential pressure.

The feed pump seal water low differential pressure was due to a secondary side condensate header low pressure transient caused by the condensate header rejecting high hotwell level to the storage tanks with only one condensate pump running.

C. ROOT CAUSE:

The underlying cause of the event was a deficiency in the operating philosophy, in that procedural guidance did not require two (2) condensate pumps in operation during the initial increase in power and turbine roll.

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 reactor trip from lo lo level in the "A" S/G was an automatic actuation of the RPS.

An assessment was performed considering both the safety consequences and implication 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 to shut the reactor down as designed.

- o The plant was quickly stabilized in hot shutdown.

The transient was compared to the assumptions of the accidents evaluated in Section 15 of the Ginna Updated Final Safety Analysis (UFSAR). No assumptions specified in Chapter 15 of the UFSAR were violated during this event.

A slow cooldown resulted during the post trip recovery period. Tavg decreased to approximately 537°F and the MSIVs were closed. This cooldown is bounded by the plant accident analysis and does not exceed the technical specification limit of 100°F per hour. Additional protection was provided by closure of the MSIVs.

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

V. <u>CORRECTIVE ACTION</u>

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

- o During a subsequent startup, interim guidance was provided, to place the second condensate pump in service prior to starting a main feedwater pump. This guidance produced appropriate results.

B. ACTION TAKEN OR PLANNED TO PREVENT RECURRENCE:

- o The new operating philosophy endorses the interim guidance discussed above. Appropriate startup procedures will be changed to direct the placing in service of the second condensate pump prior to starting a main feedwater pump.

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

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 1 9 —	0 0	0 6 OF 0 6

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

VI. ADDITIONAL INFORMATION

A. FAILED COMPONENTS:

None.

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 could be identified. However, LER 85-009 was a similar event with a different root cause.

C. SPECIAL COMMENTS:

None.