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ACCESSION NBR:8807080142 DOC.DATE: 88/07/01 NOTARIZED: NO DOCKET #
FACIL:50-244 Robert Emmet Ginna Nuclear Plant, Unit 1, Rochester G 05000244
AUTH.NAME AUTHOR AFFILIATION
BACKUS,W.H. Rochester Gas & Electric Corp.
SNOW,B.A. Rochester Gas & Electric Corp.
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 88-005-00:on 880601,low steam generator water level due to blown fuse in controlling feedwater flow channel.
W/8 ltr.

DISTRIBUTION CODE: IE22D COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 12
TITLE: 50.73 Licensee Event Report (LER), Incident Rpt, etc.

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

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INTERNAL:	ACRS MICHELSON	1	1	ACRS MOELLER	2	2
	AEOD/DOA	1	1	AEOD/DSP/NAS	1	1
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	ARM/DCTS/DAB	1	1	DEDRO	1	1
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	REG FILE 02	1	1	RES TELFORD,J	1	1
	RES/DE/EIB	1	1	RES/DRPS DEPY	1	1
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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 1 1

TITLE (4) Low Steam Generator Water Level Due To Blown Fuse in Controlling Feedwater Flow Channel Causes Reactor Trip and Subsequent Safety Injection

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
0 6	0 1	8 8	8 8	— 0 0 5	— 0 0	0 7	0 1	8 8		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 9 8

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	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
X	J B	X X F V	B 5 6 9	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 1, 1988 at 1932 EDST with the reactor power at approximately 98%, a reactor trip occurred due to low level in the "B" Steam Generator coincident with steam flow - feedwater flow mismatch followed in approximately 3 minutes by a safety injection initiation from pressurizer low pressure.

The two reactor trip breakers opened as required and all shutdown and control rods inserted as designed. All safeguards equipment operated as designed with the exception of two valves.

The underlying cause of the reactor trip was a transient caused by a blown fuse in the "B" Steam Generator controlling feedwater flow channel power supply. The underlying cause of the safety injection initiation was a reactor coolant system cooldown caused by overfeeding the steam generators.

Immediate corrective action was to stabilize the plant per the Emergency Operating Procedures for Reactor Trip and Safety Injection.

Action taken to prevent recurrence was to investigate and replace all feedwater flow channel power supply fuses and to replace the power supply on the effected channel.

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JE22

I. PRE-EVENT PLANT CONDITIONS

The unit was at approximately 98% full power with no major activities in progress. The "B" Containment Spray Pump was held for seal maintenance.

II. DESCRIPTION OF EVENT

A. EVENT:

On June 1, 1988 at 1932 EDST with the reactor at approximately 98% of full power, a reactor trip occurred due to low level in the "B" Steam Generator (i.e. steam generator level $\leq$ 30%) coincident with steam flow, feedwater flow (SF/FF) mismatch (i.e. steam flow $\geq$ 0.8E6 lbm/hr more than feedwater flow).

The cause of the low level plus SF/FF mismatch reactor trip was due to a fuse opening on the power supply to the "B" Steam Generator flow transmitter FT-476. This flow transmitter was supplying an input signal to the "B" Steam Generator level control system. The fuse opening caused the flow transmitter to fail low. As a result of this the "B" Steam Generator feedwater regulating valve opened more, increasing flow to the "B" Steam Generator and decreasing flow to the "A" Steam Generator. These flow changes plus the failed feedwater flow channel caused steam flow > feed flow alarms on both the "A" and "B" Steam Generators. The Control Room Operators reacted to the alarms on the steam generators by placing both main feedwater regulating valves in manual to maintain steam generator levels per alarm response procedures.

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

The "B" Steam Generator level increased to the 67% high level feedwater isolation setpoint and the "B" Steam Generator feedwater regulating valve closed as designed. After the "B" Steam Generator level decreased below approximately 65%, the "B" Steam Generator feedwater regulating valve reopened to its original position prior to the high level isolation. The combination of decreasing steam generator level and rapidly increasing feedwater flow (i.e. shrink phenomena) caused the "B" Steam Generator level to go below 30%. This level with the "B" Steam Generator SF/FF mismatch signal locked in due to the failed feedwater flow channel, caused a reactor trip. Due to the rapid sequence of events (approximately 80 seconds from the time of the FT-476 failure to the time of the reactor trip), the operator had very little time to diagnose and react to prevent the 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) to stabilize the plant.

Following the reactor trip, both steam generator water levels decreased below 17% causing the two motor driven and the steam driven auxiliary feedwater pumps to start and deliver full flow of approximately 800 gpm to the steam generators (approximately 400 gpm per steam generator). In addition to the above auxiliary feedwater flow, the main feedwater system was still operating providing approximately full main feedwater flow to the steam generators. This was due to the main feedwater regulating valves being in manual versus auto control, (i.e., the main feedwater regulating valves would go closed when the reactor coolant system average temperature decreased below 554^{o}F if they were in auto, but since both regulating valves were in manual, they remained open). Because of the large volume of cool feedwater addition to the steam generators, the reactor coolant system was rapidly cooled and pressurizer pressure decreased to the safety injection initiation value of $\leq$ 1750 psig and pressurizer level decreased to 0% indicated. The safety injection initiation occurred approximately two minutes following the reactor trip.

Due to the safety injection initiation, main feedwater addition to the steam generators was isolated and the reactor coolant system cooldown and depressurization rate was reduced significantly. As Emergency Operating Procedures actions continued, it was discovered that the main steam control valve (AOV-3425) to the 1A reheater had failed to close on turbine trip and was still supplying main steam to the 1A reheater. The main steam isolation valves were closed to isolate the above steam flow to the 1A reheater. This action terminated the cooldown and allowed recovery of reactor coolant system temperature, pressure and pressurizer level. Reactor coolant system pressure reached a low of approximately 1510 psig with no injection of water into the reactor coolant system.

Following the safety injection initiation, the Control Room Operators performed the actions of Emergency Operating Procedures E-0 (Reactor Trip or Safety Injection), and ES-1.1 (SI Termination) followed by Normal Operating Procedure O-2.1 (Normal Shutdown to Hot Shutdown) and stabilized the plant in hot shutdown.

One procedural problem was identified regarding ES-1.1 (SI Termination). This problem was the need to change the method of flushing the high head safety injection pump lines following safety injection termination. With the present method, the boric acid storage tanks become diluted during the flush. This problem has already been addressed in draft Emergency Operating Procedure Upgrade rewrites and any further flushes will be accomplished using this new guidance.

Other equipment problems encountered during the reactor trip or safety injection were as follows:

- o Intermediate Range Nuclear Instrument Channel N-36 remained high following the reactor trip and source range nuclear instrumentation was re-instated manually.

- o 1A Boric Acid Storage Tank Level Indication LI-172 went high during the event.

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

None.

C. DATES AND APPROXIMATE TIMES FOR MAJOR OCCURRENCES:

- o June 1, 1988, 1932 EDST: Event date and time

- o June 1, 1988, 1932 EDST: Discovery date and time

- o June 1, 1988, 1934 EDST: Safety Injection Initiation on low pressurizer pressure $\leq$ 1750 psig

- o June 1, 1988, 1959 EDST: Closed both main steamline isolation valves

- o June 1, 1988, 2013: Unit stabilized in hot shutdown

D. OTHER SYSTEMS OR SECONDARY FUNCTIONS AFFECTED:

None.

E. METHOD OF DISCOVERY:

The event was immediately apparent due to alarms and indication in the Control Room.

F. OPERATOR ACTION:

Following 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) to stabilize the plant.

Following the safety injection initiation the Control Room Operators performed the actions of Emergency Operating Procedures E-0 (Reactor Trip or Safety Injection), and ES-1.1 (SI Termination), followed by Normal Operating Procedure O-2.1 (Normal Shutdown to Hot Shutdown) and stabilized the plant in hot shutdown.

Subsequent to the safety injection initiation the Control Room Operators closed the main steam line isolation valves to terminate a primary system cooldown, due to the 1A reheater main steam control valve failing to close.

III. CAUSE OF EVENT

A. IMMEDIATE CAUSE:

The reactor trip occurred due to low level in the "B" Steam Generator coincident with the "B" Steam Generator SF/FF mismatch (i.e. steam generator level $\leq$ 30% and SF $\geq$ 0.8E6 lbm/hr more than FF) because of control system transient caused by the failure low of the controlling "B" Steam Generator feedwater flow channel.

B. ROOT CAUSE:

The failure low of the controlling "B" Steam Generator feedwater flow channel FT-476 was due to a random failure of a fuse supplying the FT-476 flow transmitter power supply.

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)". The reactor trip resulting from the "B" Steam Generator low level coincident with the "B" Steam Generator SF/FF mismatch was an automatic actuation of the RPS and the subsequent safety injection initiation on low pressurizer pressure of $\leq$ 1750 psig was an automatic actuation of the ESF.

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

There were no safety consequences or implications attributed to the "B" Steam Generator low level coincident with the "B" Steam Generator SF/FF mismatch reactor trip because;

o The two reactor trip breakers opened as required.

o All control and shutdown rods inserted as designed.

o The reactor was rendered subcritical with proper shutdown margin as designed.

o Both steam generator water levels were maintained on scale in the narrow range instrumentation thus assuring an adequate heat sink for decay heat removal.

There were no safety consequences or implications attributed to the safety injection actuation because:

o All equipment and systems required for the ESF function operated as designed except as noted below.

  a. MOV-814 (Component Cooling Water Return From Reactor Support Cooling) did not close as designed on the Containment Isolation Signal. Upon verification of failure to close, operator action was taken immediately to electrically close MOV-814 from the Main Control Board. The safety significance of this failure is minimized due to the fact that this is a closed system.

  b. MOV-4616 (Auxiliary Building Service Water Isolation Valve 1A) closed with the safety injection initiation. This closure was not as designed. This valve is designed to close with safety injection initiation coincident with undervoltage on Bus 14. MOV-4616 was reopened electrically from the Main Control Board following safety injection reset. The safety significance of this failure mode is negligible since the valve travelled to its safeguard position for isolation.

  c. LI-172 (1A Boric Acid Storage Tank Level Indication) went high during the event. This failure had no adverse effect on the event as the Boric Acid Storage Tank swapover to the Refueling Water Storage Tank would still have taken place from the redundant level channel.

o There was no injection of safety injection water into the reactor coolant system as the minimum reactor coolant system pressure attained during the event was above the shutoff head of the safety injection pumps.

o The Emergency Operating Procedures proved to be an effective tool in dealing with the event.

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)

Chapter 15 of the Ginna Station UFSAR was reviewed to determine if any accident analysis assumptions, were violated. The directly applicable accident to this event is the "increase in heat removal by the secondary system". This review concluded that all assumptions of the safety analysis were met.

Based on the above, it can be concluded that all systems which were required, performed as designed or the deviations were acceptable, thus assuring the public's health and safety.

V. <u>CORRECTIVE ACTION</u>

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

- The "B" Steam Generator water level was returned to its normal operating band by feedwater addition.

- The high head safety injection lines were sampled and verified to contain low concentrations of boric acid.

- MOV-814 - contacts 5 and 6 were found to be in the wrong state on containment isolation relay R1-5-Y. Maintenance Work Request 88-3928 was written to return the above contacts to the proper state. This relay was post-maintenance tested satisfactorily. For more details see NRC Inspection Report 50-244/88-11 (Item 7).

- MOV-4616 - contacts on 18 Bx relay were found normally open instead of normally closed. NCR G 88-287 and Maintenance Work Request 88-3972 were issued to correct this problem. For more details see NRC Inspection Report 50-244/88-11 (Item 8).

- FT-476 - A blown fuse was found in the flow transmitter power supply. The fuses and power supply were replaced and calibrated and tested satisfactorily.

- LI-172 - the sensing line was found partially plugged. The sensing line was reamed out per procedure PT-21 (Cleaning Boric Acid Tank Sensing Lines) and LI-172 was returned to service.

- AOV-3425 - The positioner arm was found loose and also a faulty valve stem was found. Valve was overhauled and positioner arm tightened. The valve was calibrated and stroked satisfactorily.

B. ACTION TAKEN OR PLANNED TO PREVENT RECURRENCE:

- All feedwater flow channel power supply fuses were replaced.

- FT-476 power supply was replaced.

- All other corresponding reheater steam supply valves were inspected to insure proper operation.

VI. ADDITIONAL INFORMATION

A. FAILED COMPONENTS:

The fuse that opened on flow transmitter FT-476 power supply was a "Bussman" style MDL 1/10 Amp, 125 volt fuse.

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

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 Station could be identified.

C. SPECIAL COMMENTS:

None.

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

TELEPHONE
AREA CODE 716 546-2700

July 1, 1988

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

Subject: LER 88-005, Low Steam Generator Water Level Due To Blown Fuse in Controlling Feedwater Flow Channel Causes Reactor Trip and Subsequent Safety Injection
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 88-005 is hereby submitted.

This event has in no way affected the public's health and safety.

Very truly yours,

Bruce A. Snow

Bruce A. Snow
Superintendent of
Nuclear Production

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

Ginna USNRC Senior Resident Inspector

IE22
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