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ACCESSION NBR:9206250008 DOC.DATE: 92/06/17 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.
MECREDY,R.C. Rochester Gas & Electric Corp.
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 92-006-00:on 920518,steam generator high level feedwater isolations occurred.Caused by secondary sys pressure transient.After SG levels stabilized,main feedwater regulating valve returned to automatic control.W/920617 ltr.

DISTRIBUTION CODE: IE22T COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 9
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

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	NRR/DET/EMEB 7E	1	1	NRR/DLPQ/LHFB10	1	1
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EXTERNAL:	EG&G BRYCE,J.H	3	3	L ST LOBBY WARD	1	1
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Vice President
Ginna Nuclear Production

TELEPHONE
AREA CODE 716 546-2700

June 17, 1992

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

Subject: LER 92-006, Feedwater Control Perturbations, Due To A Secondary Side Transient, Causes Steam Generator High Level Feedwater Isolations
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

In accordance with 10CFR50.73, 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 Event Report LER 92-006 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

Cert No P148978413
IE22
1/1

NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WTH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

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 8

TITLE (4) Feedwater Control Perturbations, Due To A Secondary Transient, Causes Steam Generator High Level Feedwater Isolations

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)
0 5	1 8	9 2	9 2	0 0 6	0 0	0 6	1 7	9 2		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 9 7

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

SUPPLEMENTAL REPORT EXPECTED (14)

[X] YES *(If yes, complete EXPECTED SUBMISSION DATE)* [ ] NO

EXPECTED SUBMISSION DATE (15): MONTH 0 7 DAY 0 1 YEAR 9 3

ABSTRACT *(Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines)* (16)

On May 18, 1992 at approximately 1339 EDST, with the Reactor at approximately 97% full power, Main Feedwater Isolations occurred on the "A" Steam Generator (S/G). These feedwater isolations were caused by overfeeding the "A" S/G while in manual control, following a secondary system pressure transient.

Immediate operator action was to manually control the Main Feedwater Regulating Valves (FRV) to reduce the steam generator water levels and stabilize the plant.

The immediate cause of the event is due to a secondary system pressure transient. (This event is NUREG-1022 (X) cause code.) The root cause is undetermined at this time.

Corrective action taken or planned are discussed in Section V of the text.

I. PRE-EVENT PLANT CONDITIONS

The plant was at approximately 97% steady state power with no major activities in progress. Instrument and Control (I&C) Department personnel were investigating the Advanced Digital Feedwater Control System (ADFCS) due to some feedwater oscillations that had occurred at approximately 1110 EDST, May 18, 1992. These feedwater oscillations were being caused by the "A" and "B" Steam Generator (S/G) Main Feedwater Regulating Valves (FRVs) cycling (between 43% and 48%), at twice their normal frequency and in opposite phase from each other. This feedwater response (instability while in automatic control) was new and unexpected by the operators.

II. DESCRIPTION OF EVENT

A. DATES AND APPROXIMATE TIMES OF MAJOR OCCURRENCES:

- May 18, 1992, 1337 EDST: A Secondary Side Transient occurs.
- May 18, 1992, 1339 EDST: Event Date and Approximate Time.
- May 18, 1992, 1339 EDST: Discovery Date and Approximate Time.
- May 18, 1992, 1342 EDST: Control Room operators restore S/G levels to normal and place Main FRVs in automatic.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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

B. EVENT:

On May 18, 1992, at approximately 1337 EDST, with the Reactor at approximately 97% full power, a secondary side condensate system decreasing pressure transient occurred. This pressure decrease traveled throughout the entire secondary side and was reflected as low feedwater header pressure and corresponding decrease in differential pressure across both FRVs. The ADFCS responded to the low feedwater header pressure by increasing the open demand signal to the main FRVs in order to maintain the proper feedwater flow. At this same time, the "C" Condensate Pump automatically started on low condensate system header pressure. Combined with a condensate header pressure low alarm, this alerted the Control Room operators to an abnormal condition. The Control Room operators, after observing that the main FRVs open demand signals were higher than normal and increasing, and that feedwater flow to the "B" S/G was higher than appropriate, took manual control of the Main FRVs to prevent further opening of the valves.

The increased feedwater header pressure was caused by the auto start of the "C" Condensate Pump. Coupled with manual operation of the Main FRVs, the result was momentary overfeeding of the "A" S/G after the "B" FRV was throttled. During this time period, the operator manually secured the "C" condensate pump to decrease feedwater header pressure. At 1339 EDST, May 18, 1992, Main Feedwater Isolation on high level (i.e. >/= 67% narrow range level) occurred three (3) times to the "A" S/G over a period of ten (10) seconds.

The Control Room operators manually restored S/G levels to normal, and at approximately 1342 EDST restored the "A" and "B" Main FRVs to automatic control.

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:

The Control Room operators took immediate manual actions to control S/G levels and stabilize the plant. Subsequently, the Control Room operators notified higher supervision and the Nuclear Regulatory Commission per 10 CFR 50.72, Non-Emergency, 4 hour Notification.

G. SAFETY SYSTEM RESPONSES:

The "A" S/G Main FRV began to close automatically as a result of the feedwater isolation signal. Due to the short duration that the signal was present, the valve never fully closed.

III. CAUSE OF EVENT

A. IMMEDIATE CAUSE:

The feedwater isolation signal to the "A" S/G Main FRV was due to the "A" S/G narrow range level being >/= 67%.

B. INTERMEDIATE CAUSES:

The "A" S/G narrow range level was >/= 67% due to increased flow to the "A" S/G, caused by the perturbations in main feedwater header pressure during manual operation. This situation resulted in inadvertent overfeeding of the "A" S/G. It should also be noted here that just prior to this event, feedwater automatic control had been experiencing instability with flow oscillations, and the operators were sensitive to abnormal indications with ADFCS in automatic control of the FRVs.

C. ROOT CAUSE:

The root cause is believed to be the control instability problems being experienced with the FRVs. These instabilities caused operators to place ADFCS in manual following observation of FRV response. After extensive troubleshooting, the specific root cause has not been determined. Various contributing causes have been identified, and intermediate and long-term corrective actions (see Section V.B) address the need to determine the root cause.

IV. ANALYSIS OF EVENT

This event is reportable in accordance with 10CFR50.73, 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 feedwater isolation of the "A" S/G was an automatic actuation of an ESF system.

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 feedwater isolations because:

- o The feedwater isolation signals occurred at the required S/G level.
- o The plant was quickly stabilized and manual control of FRVs was accomplished to mitigate any consequences of the event.
- o As the feedwater isolation signals occurred as designed, the assumptions of the FSAR for steam line break were met.

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 After S/G levels were stabilized, the Main FRVs were returned to automatic control.

B. ACTION TAKEN OR PLANNED TO PREVENT RECURRENCE:

After reviewing the results of troubleshooting and discussions with Westinghouse, the following is an outline of the corrective actions being taken or planned in response to the feedwater control instability problems (and ADFCS transient response for secondary system transients):

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

- o Short Term Response:

  a) I&C technicians loosened and lubricated the actuator to stem connection block for the "B" FRV.

  b) Operations was directed to place the Main FRV bypass valves in full open position and in automatic control, to reduce the feedwater control instability problem.

  c) The packing on both FRVs was relaxed to minimum torque, to reduce any potential friction of the valve stem.

- o Intermediate Term Response:

  a) Westinghouse has evaluated ADFCS "tuning constants" and has supplied new ADFCS gain values, in an attempt to reduce the ADFCS impact on feedwater control instability problems.

  b) New ADFCS gain values have been installed in the ADFCS software. Instability problems have not been eliminated as a result of these new gain values.

  c) Diagnostic testing of FRV performance was completed. Testing focused on valve response to changes in current-to-pneumatic (I/P) transducers, valve positioners, and solenoids and associated instrument air tubing.

o Long Term Response:

a) Valve characteristic data (with and without valve shaft packing in place) will be reviewed.

b) The vendors of the Main FRV actuator and positioner will be consulted requesting their recommendations for valve packing and positioner operation, based on a more detailed review of the diagnostic testing.

c) The calibration procedures for the Main FRVs will be revised to include valve response characteristics during scheduled calibrations.

d) EWR 4773 will apply the information obtained from other actions, and is intended to enhance feedwater control and response to plant transients. It is envisioned that improvements will be obtained in both electronic and pneumatic performance, as a result of these enhancements.

e) After additional actions are completed during the 1993 outage, and after a root cause determination is completed, a supplemental report to this LER will be submitted.

VI. <u>ADDITIONAL INFORMATION</u>

A. FAILED COMPONENTS:

None.

B. PREVIOUS LERs ON SIMILAR EVENTS:

A similar LER event historical search was conducted with the following results: LER 91-009 was a similar event with a different root cause.

C. SPECIAL COMMENTS:

None.