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FACIL:50-244 Robert Emmet Ginna Nuclear Plant, Unit 1, Rochester G 05000244

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RECIP.NAME	RECIPIENT AFFILIATION

SUBJECT: LER 92-002-00:on 920203,reactor trip occurred w/reactor at approx 23% full power just subsequent to turbine trip while at 47% power.Caused by lo lo level in SG A due to design perturbations.New setting calculated.W/920304 ltr.

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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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March 4, 1992

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

Subject: LER 92-002, Feedwater Transient, Due To Loss Of Excitation Induced Turbine/Generator Trip, Causes Lo Lo Steam Generator Level Reactor Trip
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 92-002 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 P748978218

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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): R.E. Ginna Nuclear Power Plant

DOCKET NUMBER (2): 0 5 0 0 0 2 4 4

PAGE (3): 1 OF 1 2

TITLE (4): Feedwater Transient, Due to Loss of Excitation Induced Turbine/ Generator Trip, Causes 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 NAMES	DOCKET NUMBER(S)
0 2	0 3	9 2	9 2	— 0 0 2 —	0 0	0 3	0 4	9 2		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 2 3

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: Wesley H. Backus
Technical Assistant to the Operations Manager

TELEPHONE NUMBER: AREA CODE 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
X	B A	T R B	W 3 1 5	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 February 3, 1992 with the reactor at approximately 23% full power, just subsequent to a turbine trip while at 47% reactor power, a reactor trip occurred due to Lo Lo Level (</=17%) in the "A" Steam Generator (S/G).

The Control Room operators immediately performed the appropriate actions of Emergency Operating Procedures E-0 (Reactor Trip or Safety Injection) and ES-0.1 (Reactor Trip Response). Both Main Steam Isolation Valves (MSIVs) were subsequently closed to limit a Reactor Coolant System (RCS) cooldown and the plant was stabilized at hot shutdown.

The underlying cause of the event was the inability to control the "A" S/G level above the reactor trip setpoint due to design and transient induced perturbations. (This event is NUREG-1022 (x) Cause Code).

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

I. PRE-EVENT PLANT CONDITIONS

The plant was at approximately 47% reactor power due to a load reduction earlier in the day. Part of this load reduction (i.e. to approximately 60% reactor power) was requested by the Rochester Gas and Electric Corporation (RG&E) Power Control Dispatcher to remove a major transmission line from service (circuit 908) for repair. The remainder of this load reduction (from 60% to 47% reactor power) was made to reduce reactor power below permissive P-9 (i.e. Reactor Trip From Turbine Trip Blocked) to perform condenser water box maintenance and turbine on-line trip testing and valve testing.

The main generator voltage control was in the manual mode due to voltage oscillations experienced earlier in the day. The reactor control rods were also in the manual mode to maintain the core axial flux within its operating band. Turbine on-line trip testing and valve testing was in progress with the last test's initial conditions being verified prior to performance of the test.

II. DESCRIPTION OF EVENT

A. DATES AND APPROXIMATE TIMES OF MAJOR OCCURRENCES:

- February 3, 1992, 2220 EST: Main Generator Trip due to loss of excitation.
- February 3, 1992, 2220 EST: Main Turbine trip due to main generator trip.
- February 3, 1992, 2224 EST: Event date and time.
- February 3, 1992, 2224 EST: Discovery date and time.
- February 3, 1992, 2224 EST: Control Room operators verify both reactor trip breakers open, and all control and shutdown rods inserted.

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

- o February 3, 1992, 2229 EST: Control Room operators close both Main Steam Isolation Valves (MSIVs) to limit a Reactor Coolant System (RCS) cooldown.

- o February 3, 1992, 2307 EST: Plant stabilized at hot shutdown condition.

B. EVENT:

On February 3, 1992 at approximately 2220 EST, with the reactor at approximately 47% stable reactor power, the Control Room received turbine trip first out annunciator alarm K-26 (Generator Lockout Relay). As reactor power was less than 50% full power with the main condenser available, (i.e. less than permissive P-9), reactor trip from turbine trip was automatically blocked. The Control Room operators immediately entered Abnormal Procedure AP-TURB.1 (Turbine Trip Without Reactor Trip Required) and performed its applicable actions.

The responses of the Steam Generator (S/G) Feedwater Regulating Valves (FRVs) for different control configurations are noted here for clarity of subsequent events:

- o When all FRVs (Main and Bypass) are in the automatic mode, they will go full open on a turbine trip with RCS average temperature (Tavg) greater than 554°F. When temperature goes below 554°F, these valves will go full closed.

- o All FRVs, both main and bypass, will fully close upon receiving a HI S/G Level (>/= 67%) or safety injection signal regardless of their auto/manual status (feedwater isolation).

- o When the Main FRVs are placed in the manual mode from the above configuration, they will assume the position based on the current controller manual demand signal and stay there until adjusted by the responsible operator. Assuming that the bypass FRVs are left in auto, the bypass FRVs will continue to respond to the automatic feedwater control demand.

Prior to the turbine trip all FRVs were in the automatic control mode with the bypass FRVs full open and the main FRVs controlling S/G level in some midposition.

Subsequent to the turbine trip, the main FRVs went full open per design (the bypass FRVs were already full open) and the Control Room operator transferred the main FRVs to manual control and adjusted them to control S/G levels. The bypass FRVs were left in the automatic control mode.

The narrow range S/G levels were at approximately 53% and rapidly increasing when the main FRVs were shifted to the manual mode. The Control Room operator closed down the main FRVs to approximately 10-13% open to control S/G levels. During this time the "A" S/G FRV isolated on HI Level (i.e. >/= 67% narrow range level). During approximately this same time the full open bypass FRVs rapidly closed because their automatic controlling setpoint was now 39% S/G level. The Control Room operator continued to make adjustments to the main FRVs to compensate for the transient perturbations, but was unsuccessful. At 2224 EST, February 3, 1992, with the reactor at approximately 23% full power a reactor trip occurred due to Lo Lo Level (</= 17%) in the "A" S/G.

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.

Both MSIVs were subsequently closed at 2229 EST to limit the RCS cooldown. The closing of the MSIVs subsequently mitigated the RCS cooldown and the plant was stabilized in hot shutdown at 2307 EST.

Equipment problems that occurred during the event were as follows:

- o The "A" Steam Generator MSIV Main Control Board indication showed the valve to be not fully closed. An auxiliary operator was immediately sent out to check and reported the valve closed based on viewing the valve external indicator. Subsequently, the Main Control Board indicated the valve fully closed approximately 23 minutes after signal receipt.

- o Following the start of the Turbine Driven Auxiliary Feedwater (TDAFW) pump on Lo Lo S/G Levels, it exhibited some oscillations in flow, however total flow remained above the required 400 gallons per minute (GPM) as recorded on the Plant Process Computer System (PPCS).

- o The Intermediate Range Nuclear Instrumentation, Channel N-35, after tracking consistant with Channel N-36 down to approximately 10 E-10 amps, had its indication continue to drop below 10 E-11 amps. The N-35 channel returned to normal (10 E-11 amps) approximately ten hours following the trip.

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:**

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 MSIVs were manually actuated closed approximately four (4) minutes after the trip to prevent further plant cooldown. The plant was subsequently stabilized at hot shutdown. 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 FRVs closed automatically from a feedwater isolation signal.

III. **CAUSE OF EVENT**

A. **IMMEDIATE CAUSE:**

The reactor trip was due to "A" S/G Lo Lo Level (</=17%).

B. **ROOT CAUSE:**

The underlying cause of the "A" S/G Lo Lo Level (</=17%) was determined to be the Control Room operator's inability to control the "A" S/G level above its reactor trip setpoint due to the following contributing factors:

- o The transient perturbations that were occurring due to design (i.e. the design of the FRVs to go full open, when in automatic mode, following a turbine trip, and the design of having the bypass FRVs open at the higher power levels).

- o The shrink and swell phenomenon of the S/G water levels due to the above design induced perturbations and the transient induced perturbations.

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 "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:

There were no safety consequences or implications attributed to the reactor trip because:

- o The two reactor trip breakers opened as required.
- o All control and shutdown rods inserted as designed.
- o The plant was stabilized at hot shutdown.

The Ginna Updated Final Safety Analysis Report (UFSAR) Chapter 15.2.2, "Loss of External Electrical Load", was reviewed and compared to the plant response for this event. The plant is designed to accept a 50% loss of electrical load while operating at full power or a complete loss of electrical load while operating below 50%, without initiating a reactor trip.

Although plant design can accept a 50% loss of load, past experience has shown a vulnerability to S/G shrink and swell. Since the S/G main FRVs are placed in manual, actual plant response will be dependent upon operator response to indicated S/G level.

Ginna UFSAR Chapter 15.2.2 evaluates the plant behavior for a complete loss of load from full power without a direct reactor trip, primarily to show the adequacy of the pressure-relieving devices and also to show that no core damage occurs. Complete loss of load analysis shows that DNBR does not drop below 2.0 and pressurizer pressure does not exceed 2500 psia. Only a small pressure spike (approximately 30 psig) and a small temperature spike (2.5^{o}F) was encountered during this transient.

Following the reactor trip, pressurizer level decreased to approximately 5% level as a result of the cooldown. This is an expected observed transient. The S/G levels both decreased to the narrow range taps. This is an expected transient based upon the encountered shrink in the S/Gs.

A slow cooldown resulted during the post trip recovery period. 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 and a review of post trip data and past plant transients, it can be concluded that the plant has operated as designed and that there was no unreviewed safety questions and 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 The S/G levels were returned to their normal operating levels by addition of Auxiliary Feedwater, subsequent to the reactor trip.

- o The "A" MSIV, manufactured by Atwood and Morrill, is a 30 inch air operated swing check valve, installed in the reverse direction to use S/G steam flow to ensure proper closure. As with any swing check valve, the closing moment must be large enough to overcome the friction on the valve shaft due to the valve packing. Complete closure is accomplished by the force of the fluid flow on the valve disc. The "A" MSIV was subsequently stroked several times successfully to ensure operability and adequate closure capability. Results of these tests support the conclusion that failure of the "A" MSIV to fully seat during the reactor trip was not due to internal valve distortion and bending, but was the result of a lack of flow across the valve disc. Failure to close is attributed to the closure operation occurring in a quiescent environment.

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

Valve closure is dependent upon two factors: The moment, developed by the weight of the valve disc and the spring provided to assist in valve closure, plus sufficient steam flow across the valve disc, without which the valve was not capable of completing its closing operation. When the demand signal for MSIVs to close was generated d/p across the A MSIV was lower than the d/p across the B MSIV. The d/p across the B MSIV was enough to fully seat the valve while the d/p across the A MSIV did not provide enough force to overcome shaft packing frictional forces. Approximately 23 minutes later, the d/p across the A MSIV increased approximately 2 psid which resulted in complete closure of the valve.

For all design basis accidents, where MSIV closure is required, the accident transients would develop a large enough differential pressure to obtain complete valve closure. RG&E is continuing to evaluate various packing materials which have a low friction coefficient and can perform the required sealing function.

- The TDAFW pump was subsequently tested to determine the cause of the flow oscillations, but the test had to be aborted due to a steam leak on the governor valve. The governor valve was disassembled, inspected and placed back in service. The steam leak was the result of a gasket leak (probably caused due to excessive travel of the governor valve). The gasket was replaced with a qualified spare. The cause of the flow oscillations was due to the "hunting" of the governor valve. The hunting of the governor valve was caused by a feedback nut being out of proper position. The position of the feedback nut was corrected and the TDAFW pump was subsequently tested successfully and returned to service.

[illegible]

- As the Intermediate Range NIS Channel N-35 tracked NIS Channel N-36 for its normal operating range and returned to normal approximately ten (10) hours after the trip, no immediate action was deemed necessary. This abnormality has been observed and researched extensively in the past in cooperation with the NSSS vendor, Westinghouse. No technical basis has been identified as to why the 10 E-11 idle current does not maintain indication at 10 E-11 amps. RG&E and Westinghouse concurred that the channel was operable and capable of performing all intended functions. Further evaluations of the response characteristics of NIS channel N-35 will be performed during the 1992 Annual Refueling and Maintenance Outage.

- This event was initiated by a main generator trip due to loss of excitation. Extensive examination, evaluation and testing was performed on the main generator voltage control system, with the following results and conclusions:

  The automatic and manual voltage control units were extensively tested and found to operate satisfactorily. The Minimum Excitation Limiter (MEL) data taken during the testing indicated that the setting was too close to normal operating points of the generator. A new MEL setting was calculated, reviewed by Westinghouse (the vendor) and implemented. The revised setting will allow operation to approximately 190 MVAR underexcited at 500 MW.

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

The unit was synchronized to the system with the voltage regulator in manual and no unusual events were noted. When the voltage regulator was placed in the automatic mode, the loop within the voltage regulator system was unstable and the voltage regulator was returned to the manual mode. The operation of the voltage regulator damping module was verified and the gain was reduced from maximum setting to the mid point on the potentiometer. The voltage regulator was returned to the automatic mode and operated satisfactorily.

The plant has now operated at approximately full power and has gone through several normal voltage adjustments such as lowering the voltage in the evening and raising the voltage in the morning as required by system load. No abnormalities have been encountered.

B. **ACTION TAKEN OR PLANNED TO PREVENT RECURRENCE:**

As the underlying cause of the event was determined to be the inability of the Control Room operator to control the "A" S/G level above its reactor trip setpoint due to design and transient induced level perturbation, the following actions have been taken or are being planned:

o Applicable operating procedures have been changed to require that the bypass FRVs be placed in manual closed when increasing above approximately 30% reactor power.

o Applicable operating procedures have been changed to require returning the bypass FRVs to automatic control when decreasing below 30% reactor power.

- o A modification is planned for the 1993 Annual Refueling and Maintenance Outage that would modify the existing feedwater isolation logic for fail-open or fail-closed of the main and bypass FRVs upon turbine trip with main FRVs in automatic control mode. The planned modification will delete the existing fail-open logic and replace the fail-closed logic with actuation upon reactor trip as opposed to turbine trip.

As the event was initiated by the Main Generator trip due to loss of excitation, the following actions are planned to prevent recurrence:

- o RG&E is planning to purchase and install a replacement voltage regulator unit.

- o Routine testing and maintenance will be performed on the existing voltage regulator unit during the 1992 Annual Refueling and Maintenance Outage to attain a high degree of confidence that the unit will operate without incident for the entire fuel cycle.

VI. ADDITIONAL INFORMATION

A. FAILED COMPONENTS:

The TDAFW pump turbine is a 465 horsepower noncondensing steam turbine, serial number 26635, manufactured by the Worthington Corporation.

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 underlying cause at Ginna Station could be identified.

C. SPECIAL COMMENTS:

None.