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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 90-013-00:on 901211,turbine trip w/subsequent reactor trip occurred.Caused by inadvertent ATWS mitigation sys actuation circuitry (AMSAC) actuation.Jumper,omitted in AMSAC circuit design installed.W/910107 ltr.

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

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January 7, 1991

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

Subject: LER 90-013, Turbine Trip Due to an Inadvertent ATWS Mitigation System Actuation Circuitry (AMSAC) Actuation 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 90-013 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

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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 2

TITLE (4) Turbine Trip Due to an Inadvertent ATWS Mitigation System Actuation Circuitry (AMSAC) Actuation 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	1 1	9 0	9 0	– 0 1 3 –	0 0	0 1	0 7	9 1		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)	☒ 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
B	J C	P M C	F 1 8 0	Y					

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
☐ YES (If yes, complete EXPECTED SUBMISSION DATE)	☒ NO				

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

On December 11, 1990 at 1517 EST with the reactor at approximately 97% full power, a turbine trip with subsequent reactor trip occurred due to a inadvertent ATWS Mitigation System Actuation Circuitry (AMSAC) actuation.

The Control Room operators verified the reactor and turbine trips and performed the appropriate actions of E-O (Reactor Trip or Safety Injection) and ES-0.1 (Reactor Trip Response). The plant was subsequently stabilized in the hot shutdown condition.

The intermediate cause of the AMSAC actuated turbine trip was determined to be due to a low voltage potential of a logic output from the AMSAC Logic Circuitry.

The underlying cause was determined to be a vendor circuit design deficiency.

Corrective action taken was to install the jumper omitted in the circuit design.

I. PRE-EVENT PLANT CONDITIONS

The reactor was at approximately 97% steady state full power with the following activities in progress:

- The Instrument and Control (I&C) Department was performing Periodic Test procedure PT-5.40 (Process Instrumentation Reactor Protection Channel Trip Test (Channel 4 Yellow)).

- The I&C Department was also working on Ginna Station work request or trouble report (WR/TR) Number 90-02112 which involved the "B" Steam Generator Main Feedwater Flow Transmitter FT-477.

II. DESCRIPTION OF EVENT

A. DATES AND APPROXIMATE TIMES OF MAJOR OCCURRENCES:

- December 11, 1990, 1517 EST: Event date and time.

- December 11, 1990, 1517 EST: Discovery date and time.

- December 11, 1990, 1517 EST: Control Room operators verify both reactor trip breakers open, all control and shutdown rods inserted and turbine trip.

- December 11, 1990, 1529 EST: Control Room operators close both Main Steam Isolation Valves (MSIVs) to terminate a Reactor Coolant System (RCS) cooldown.

- December 11, 1990, 1540 EST: Plant stabilized at hot shutdown.

B. EVENT:

On December 11, 1990 at 1517 EST, with the reactor at approximately 97% full power, the Control Room received several annunciator alarms. Most notable of these alarms was the red first out annunciator alarm D-24 (Turbine Auto Stop) (indicating a Reactor Trip) and K-3 (AMSAC Actuated) (indicating a Turbine Trip).

The Control Room operators immediately performed the immediate actions of 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.

The Control Room operators, following optional procedure guidances, closed both MSIVs at 1529 EST to terminate an RCS cooldown.

The closing of both MSIVs mitigated the RCS cooldown and the plant was subsequently stabilized in hot shutdown.

Other equipment problems that occurred during the event were:

- The Intermediate Range (IR) Nuclear Instrumentation, Channel N-35, after tracking identically to Channel N-36, down to approximately 10^{-10} amps, and reinstating the source range channels properly, had its indication continue to decrease below 10^{-11} amps (i.e. offscale low). The N-35 channel returned to normal approximately nine hours following the trip. This is an explained phenomenon involving the idling current within the system.

- Approximately 20 minutes following the trip, the "A" Motor Driven Auxiliary Feedwater (MDAFW) pump was observed to be producing no flow and to have steam escaping from both ends of the pump shaft through the packing glands. The pump was removed from service and vented to cool it down. This evolution took approximately 45 minutes. The "A" Motor Driven Auxiliary Feedwater pump was operated again between 1730 and 1930 EST on December 11, 1990 with no observed problems.

- Condenser Steam Dump valve AOV-3355 indicated partially open when it should have been closed. Local investigation verified that AOV-3355 was closed.

- Reheater Steam Supply valves AOV-3426 and AOV-3428 did not indicate fully closed when they should have. Local investigation verified that the valves were closed.

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) and stabilized the plant. The MSIVs were closed approximately twelve (12) minutes after the trip to prevent further plant cooldown.

G. SAFETY SYSTEM RESPONSE:

None

III. CAUSE OF EVENT

A. IMMEDIATE CAUSE:

The reactor trip occurred due to a turbine trip.

B. INTERMEDIATE CAUSE:

The turbine trip was due to an inadvertent AMSAC actuation.

The inadvertent AMSAC actuation was determined to be due to a low voltage potential of one of the logic outputs from a Foxboro N-2CCA-DF Control Module in the AMSAC Logic Circuitry.

C. ROOT CAUSE:

The underlying cause of the event was determined to be a vendor circuit design deficiency.

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 the AMSAC-actuated turbine 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:

- 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 535°F and the MSIVs were closed. The closing of the MSIVs terminated the cooldown. 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.

Technical Specifications were reviewed in respect to the post trip review data. The following is the results of that review:

- o Pressurizer level decreased to approximately 10% following the reactor trip and subsequent cooldown. Technical specification 3.1.1.5 states in part, that for RCS temperatures at or above 350°F, the pressurizer level will be maintained between 12% and 87% of level span to be considered operable. Technical Specification 3.1.1.5 also states in part, that if the pressurizer is inoperable due to level, restore the pressurizer to operable status within six (6) hours or have the reactor below a RCS temperature of 350°F and the RHR system in operation within an additional six (6) hours. Pressurizer water level recovered to greater than 12% level well before the six (6) hour action statement.

- o Both Steam Generators (S/G) levels decreased to less than narrow range indication following the reactor trip, thus rendering them technically inoperable (i.e. due to low level) even though both loops were still in operation and performing their intended function of decay heat removal. Technical Specification 3.1.1.1(c) states in part that except for special tests, when the RCS temperature is at or above 350°F with the reactor power less than or equal to 130 MWT (8.5%), at least one reactor coolant loop and its associated steam generator and reactor coolant pump shall be in operation. Both reactor coolant loops were in operation, but the S/Gs were technically inoperable due to low narrow range level indication. Taking the most conservative Technical Specification action statement approach for the above interpretation, would lead to Technical Specification 3.1.1.1(d)(ii) which states: "If neither loop is in operation, suspend all operations involving a reduction in boron concentration in the reactor coolant system and immediately initiate corrective action to return a coolant loop to operation.

The above actions were taken and both loops were returned to technically operable status within approximately four (4) minutes following the reactor trip.

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

V. CORRECTIVE ACTION

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

o Troubleshooting of the AMSAC Logic Circuitry was performed by the I&C Department. Results of this troubleshooting identified that one of the logic outputs from a Foxboro N-2CCA-DF control module, (3 of 4 feedwater flow logic) was approximately seven (7) volts. The output voltage should have been either zero (0) volts for logic high or fifteen (15) volts for logic low. Discussions were held with the AMSAC vendor who attributed the low voltage potential to the omission of a jumper connecting the module negative output logic terminal to the module signal common. This jumper was not specified in the circuit design, nor shown on the system drawings.

  A jumper was installed as directed by the vendor. Testing of the logic circuitry verified that the logic output voltage was greater than the 10 volt minimum required by the N-2CCA-DF control module.

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 3 —	0 0	0 9 OF 1 2

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

During troubleshooting, a second unrelated problem was also identified which required the replacement of the AMSAC timer signal processing module. The defective module was replaced and successfully tested with no further problems detected.

o Following the reactor trip on 09/26/90 (discussed in LER 90-012) and this event, IR Channel N-35 indication went off scale low. Both IR channels decreased together and the source ranges properly reinstated. However, at less than 9 x 10^{-11} amps, N-35 continued to decrease until it went off scale low. N-36 remained greater than 10^{-11} amps. Approximately nine hours after the trips, N-35 came back on scale and responded the same as N-36 thereafter.

Based on conversations with Westinghouse (the supplier of the IR detectors), the age of the detector can affect the low level instrument response. N-35 was replaced during the 1990 annual outage. N-36 has been installed for many years. Because N-36 has been irradiated for significantly longer, its components will be more highly activated. The gamma emitted by the activated components is added to the idling current to maintain on-scale indication. Both IR channels decreased together through the range where compensation is important (i.e. 10^{-7} amps to 10^{-10} amps); therefore, the compensating voltage is properly set. Westinghouse experts have stated that as long as both IR channels track together while decreasing and increasing power, the channels can be considered fully operable. In addition, the protective functions of both IR channels were tested satisfactorily using Periodic Test procedure PT-6.2 (N.I.S. Intermediate Range Channels).

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 3 —	0 0	1 0 OF 1 2

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

- It was determined that the "A" MDAFW pump had become steam bound due to running dead-headed (i.e. zero flow conditions).

  The "A" MDAFW pump running dead-headed was due to the following circumstances:

  - The discharge cross-tie valves were open between the two MDAFW pumps.

  - The "B" MDAFW pump, based on testing results, has approximately a 15 psig higher discharge pressure than the "A" MDAFW pump.

  Since there is very little piping run between the MDAFW pumps while the cross-ties are open, it is postulated that the higher discharge pressure of the "B" MDAFW pump forced the "A" MDAFW pump discharge check valve closed. This would create the zero flow conditions for the "A" MDAFW Pump and result in heat being transferred to the condensate, resulting in steaming through both pump shaft packing glands.

  - The condenser steam dump valve AOV-3355 position indication problem was due to a bent position switch actuating arm. The arm was replaced and the limit switch adjusted for proper position indication.

  - The reheater steam supply valves, AOV-3426 and AOV-3428 position indication problem was due to their position indication mechanisms being out of adjustment. The affected valves position indication mechanisms were adjusted for proper indication.

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 3 —	0 0	1 1 OF 1 2

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

B. ACTION TAKEN OR PLANNED TO PREVENT RECURRENCE:

- The jumper, omitted in the AMSAC circuit design, was installed to obtain the 10 volt minimum voltage.

- The AMSAC design was reviewed to identify any other logic-to-logic connections. No similar connections were found.

- Detailed system checkout was performed on the AMSAC system, with plant conditions simulated to be similar to those during the initial AMSAC actuation. This checkout, performed for more than eight (8) hours, did not show any observed problems with the low voltage potential, and no other problems were observed with the performance of the AMSAC system.

- The industry was notified, via Nuclear NETWORK, of the design deficiency in the control module.

- Calibration procedures will be revised to include an annual check of the voltage measurement of the logic outputs.

- Parallel operation of the MDAFW pumps will not be allowed in Emergency Operating Procedure (EOP) ES-0.1. This procedure has been changed to require that one running MDAFW pump be secured should the discharge cross-tie valves be open.

VI. <u>ADDITIONAL INFORMATION</u>

A. FAILED COMPONENTS:

- The AMSAC logic output module which lacked the jumper was a Foxboro N-2CCA-DF control module manufactured by the Foxboro Company.

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 3	— 0 0	1 2 OF 1 2

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

- The faulty AMSAC timer signal processing module was also a Foxboro N-2CCA-DF control module manufactured by the Foxboro Company.

B. **PREVIOUS LERs ON SIMILAR EVENTS:**

A similar LER event historical search was conducted with the following results: LER 89-004 (Turbine Trip During Manual Unblock of ATWS Mitigation System Actuation Circuitry (AMSAC), Due to Modification Program Inadequacy) was a similar event with a different root cause. The root cause was different and the corrective action for LER 89-004 was not applicable to preventing this event.

C. **SPECIAL COMMENTS:**

None