CATEGORY 1

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ACCESSION NBR:9906040057 DOC.DATE: 99/05/27 NOTARIZED: NO DOCKET #
FACIL:50-244 Robert Emmet Ginna Nuclear Plant, Unit 1, Rochester G 05000244

AUTH.NAME	AUTHOR AFFILIATION
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MECREDY,R.C.	Rochester Gas & Electric Corp.
RECIP.NAME	RECIPIENT AFFILIATION
VISSING,G.S.	

SUBJECT: LER 99-008-00:on 990427,overtemperature delta T reactor trip occurred due to faulted bistable during calibr of redundant channel.Plant was stabilized in mode 3 & faulted bistable was subsequently replaced.With 990527 ltr.

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ROBERT C. MECREDY
Vice President
Nuclear Operations

May 27, 1999

U.S. Nuclear Regulatory Commission
Document Control Desk
Attn: Guy S. Vissing
Project Directorate I-1
Washington, D.C. 20555

Subject: LER 1999-008, Faulted Bistable During Calibration of Redundant Channel Results in Overtemperature Delta T Reactor Trip
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Vissing:

The attached Licensee Event Report LER 1999-008 is submitted 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 a manual or automatic actuation of any engineered safety feature (ESF), including the reactor protection system (RPS)".

Very truly yours,

Robert C. Mecredy

xc: Mr. Guy S. Vissing (Mail Stop 8C2)
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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IE22

U.S. NRC Ginna Senior Resident Inspector

NRC FORM 366 (6 1998) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

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TITLE (4)

Faulted Bistable During Calibration of Redundant Channel Results in Overtemperature Delta T 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 NAME	DOCKET NUMBER
04	27	1999	1999	008	00	05	27	1999		05000
									FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9): 1

POWER LEVEL (10): 090

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)	X	50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
John T. St. Martin - Technical Assistant	(716) 771-3641

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
B	JC	TS	N430	YES					

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

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On April 27, 1999, at approximately 1248 EDST, with the plant in Mode 1 at approximately 90% reactor power and a power escalation in progress after the 1999 refueling outage, Instrument and Control technicians had placed one channel of the reactor protection system in the tripped condition for calibrations. A bistable in a different channel entered the tripped condition, resulting in two channels tripped and causing a reactor trip due to Overtemperature Delta T.

The Control Room operators performed the appropriate actions of procedures E 0 and ES-0.1. Following the reactor trip, all systems operated as designed, and the reactor was stabilized in Mode 3.

The underlying cause of the reactor trip was a faulted bistable.

Immediate corrective was taken to stabilize the plant in Mode 3. The faulted bistable was subsequently replaced.

Corrective action to prevent recurrence is outlined in Section V.B.

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

I. PRE-EVENT PLANT CONDITIONS:

On April 27, 1999, the plant was in Mode 1 at approximately 90% reactor power. A power escalation was in progress as the plant started up from the 1999 refueling outage. Instrument and Control (I&C) technicians were in the process of several calibrations and tests in channel 2 of the reactor protection system (RPS), and had defeated various RPS outputs from channel 2. Among procedures being used were CPI-TRIP-TEST-5.20, CPI-AXIAL-N-42, CPI-PRESS-946, and CPI-PRESS-949. Channel defeat involves placing numerous bistables into the tripped condition in protection racks in the Control Room, and results in a similar number of Main Control Board (MCB) annunciator alarms.

Unrelated to these calibrations, the "B" main feedwater regulating valve (MFWRV) was in "manual" valve position control due to concerns with automatic control of this valve.

II. DESCRIPTION OF EVENT:

A. DATES AND APPROXIMATE TIMES OF MAJOR OCCURRENCES:

- April 27, 1999, 1248 EDST: Event date and time.
- April 27, 1999, 1248 EDST: Discovery date and time.
- April 27, 1999, 1248 EDST: Control Room operators verify both reactor trip breakers open and verify all control and shutdown rods inserted.
- April 27, 1999, 1255 EDST: Control Room operators manually close both main steam isolation valves to limit a reactor coolant system cooldown.
- April 27, 1999, 1258 EDST: Control Room operators ensure both main feedwater pumps are secured to limit a reactor coolant system cooldown.
- April 27, 1999, 1325 EDST: Plant is stabilized in Mode 3.

B. EVENT:

On April 27, 1999, the plant was in Mode 1 at approximately 90% reactor power. A power escalation was in progress as the plant started up from the 1999 refueling outage. Instrument and Control (I&C) technicians were in the process of several calibrations and tests in channel 2 of the reactor protection system (RPS), and had defeated various RPS outputs from channel 2. Among procedures being used were CPI-TRIP-TEST-5.20, CPI-AXIAL-N-42, CPI-PRESS-946, and CPI-PRESS-949. Channel defeat involves placing numerous bistables into the tripped condition in protection racks in the Control Room, and results in a similar number of Main Control Board (MCB) annunciator alarms.

Unrelated to these calibrations, the "B" main feedwater regulating valve (MFWRV) was in "manual" valve position control due to concerns with automatic control of this valve.

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At approximately 1248 EDST, a reactor trip occurred due to 2/4 Overtemperature Delta T trip. In addition to the MCB annunciators already in alarm from the channel 2 defeats, the Control Room operators acknowledged MCB annunciator D-3 (O T DT Trip), indicating a reactor trip from Overtemperature Delta T.

The Control Room operators performed the immediate actions of Emergency Operating Procedure E-0, "Reactor Trip or Safety Injection". They 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.

During the transient, level in the "A" steam generator (SG) decreased below 17%, which resulted in automatic start of both motor-driven auxiliary feedwater (AFW) pumps. The Control Room operators verified that the AFW pumps had started as designed on Lo Lo SG level. With the "B" MFWRV still in "manual", feedwater was not automatically limited to the "B" SG, and the Control Room operators manually closed the "B" MFWRV. Level in the "B" SG did not decrease below 17%.

The Control Room operators received Main Control Board Annunciator K 3 (AMSAC Actuation) at approximately 1249 EDST (due to 3/4 FF channels <25%) and verified that the turbine-driven AFW pump had started due to a signal from the ATWS Mitigation System Actuation Circuitry (AMSAC).

During the transient, low seal water differential pressure occurred with the "A main feedwater (MFW) pump, and the "A" MFW automatically tripped at approximately 1249 EDST as per design. During the performance of ES-0.1, the Control Room operators noted that a reactor coolant system (RCS) cooldown was occurring, due to addition of feedwater and a low decay heat level after the recently completed refueling outage. Due to this RCS cooldown, the Control Room operators closed both main steam isolation valves (MSIVs) and secured the turbine-driven AFW pump at approximately 1255 EDST and manually stopped the "B" MFW pump at approximately 1258 EDST. These actions mitigated the RCS cooldown.

The plant was stabilized in Mode 3 at approximately 1325 EDST and the Control Room operators transitioned to normal plant operating procedures.

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

None

D. OTHER SYSTEMS OR SECONDARY FUNCTIONS AFFECTED:

None

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E. METHOD OF DISCOVERY:

This event was immediately apparent due to Main Control Board indication of the reactor trip, due to plant response and alarms and indications in the Control Room.

F. OPERATOR ACTION:

After the reactor trip, the Control Room operators performed the appropriate actions of Emergency Operating Procedures E-0 and ES-0.1. The MSIVs were manually closed and feedwater pumps were stopped to limit further RCS cooldown. Appropriate actions were taken to restore level in the "A" SG. The plant was stabilized in Mode 3.

Subsequently, the Control Room operators notified higher supervision and the NRC per 10 CFR 50.72 (b) (2) (ii), non-emergency four hour notification, at approximately 1604 EDST on April 27, 1999.

G. SAFETY SYSTEM RESPONSES:

All safeguards equipment functioned properly. Both motor driven AFW pumps started when "A" SG level decreased below 17% after the reactor trip. The turbine driven AFW pump started as per design, due to a starting signal from AMSAC.

III. CAUSE OF EVENT:

A. IMMEDIATE CAUSE:

The immediate cause of the reactor trip was achieving the 2/4 reactor protection system (RPS) trip logic for Overtemperature Delta T trip.

B. INTERMEDIATE CAUSE:

The intermediate cause of achieving 2/4 RPS trip logic was the bistable for channel 3 Overtemperature Delta T (TC-407C/D) entering the tripped condition, with the channel 2 Overtemperature Delta T bistable already tripped for channel calibrations.

C. ROOT CAUSE:

The underlying cause of bistable TC-407C/D entering the tripped condition was an instantaneous spurious fault within the bistable. The faulted bistable was subsequently bench tested by I&C. Erratic behavior was observed during the bench testing.

This event is NUREG-1022 Cause Code (B), "Design, Manufacturing, Construction / Installation".

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IV. ANALYSIS OF EVENT:

Th event is reportable 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 a manual or automatic actuation of any engineered safety feature (ESF), including the reactor protection system (RPS)". The reactor trip was an automatic actuation of the RPS, and AFW pump starts are actuations of an ESF component.

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 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 in Mode 3.

o The Ginna Station Improved Technical Specifications (ITS) Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) were reviewed with respect to the post trip review data. The following are the results of that review:

a. Pressurizer (PRZR) pressure decreased below 2205 PSIG during the transient after the reactor trip. During this time a thermal power step > 10% occurred due to the reactor trip, which is within the limits of ITS LCO 3.4.1. Therefore, compliance with ITS was maintained. The RCS temperature DNB limit (577.5 degrees F) was not approached. Additional mitigation was provided by closing the MSIVs and stopping the MFW pumps. Minimum PRZR pressure was approximately 2097 PSIG, and PRZR pressure was restored > 2205 PSIG within 10 minutes.

b. After the reactor trip, the RCS cooled down to approximately 531 degrees F and was subsequently stabilized at 547 degrees F. The cooldown was within the limits of ITS LCO 3.4.3. In addition, the required shutdown margin was maintained at all times during the RCS cooldown.

c. "A" SG level decreased following the reactor trip "A" SG level decreased below 16% indicated narrow range level. This is an expected transient. SR 3.4.5.2 states that in order to demonstrate that a reactor coolant loop is operable, the SG water level shall be >/= 16%. Thus, the "A" coolant loop was inoperable, even though it was still in operation and performing its intended function of decay heat removal. Both SGs were available as a heat sink, and sufficient AFW flow was maintained for adequate steam release from both SGs. The "A" coolant loop was restored to operable status when SG level was restored to >/= 16%, in approximately four (4) minutes. This is within the limits of ITS LCO 3.4.5 ACTION A.

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Based on the above and a review of post trip data and past plant transients, it can be co luded that the plant operated as designed, that there were no unreviewed safety questions, and that the public's health and safety was assured at all times.

V. CORRECTIVE ACTION:

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

The "A" SG was restored to operable status when SG level in the "A" SG increased above 16% level, by addition of AFW. Subsequently, levels were restored to their normal operating levels.

B. ACTION TAKEN OR PLANNED TO PREVENT RECURRENCE:

During subsequent bench testing of the faulted bistable, erratic behavior was observed. The bistable will be sent to NUS Scientech Inc. for further analysis.

VI. ADDITIONAL INFORMATION:

A. FAILED COMPONENTS:

The faulted component was a bistable module, manufactured by NUS Scientech Inc., Model DAM 503-01 differential bistable. This bistable was installed as TC-407C/D in the Overtemperature Delta T function in December 1997, and had performed acceptably in service until this event.

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 Nuclear Power Plant could be identified. Recent reactor trips with different root causes include LERs 96-002, 96-012, and 1999-007.

C. SPECIAL COMMENTS:

None