CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9604150112 DOC.DATE: 96/04/08 NOTARIZED: NO DOCKET #
FACIL:50-244 Robert Emmet Ginna Nuclear Plant, Unit 1, Rochester G 05000244

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SUBJECT: LER 96-002-00:on 960307,secondary transient occurred.Caused by loss of B condenser circulating water pump.C/As: thermography performed.W/960408 ltr.

DISTRIBUTION CODE: IE22T COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 10
TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

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ROBERT C. MECREDY
Vice President
Nuclear Operations

April 8, 1996

U.S. Nuclear Regulatory Commission
Document Control Desk
Attn: Allen R. Johnson
PWR Project Directorate I-1
Washington, D.C. 20555

Subject: LER 96-002, Secondary Transient, Caused by Loss of "B" Condenser Circulating Water Pump, Results in Manual 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 a manual or automatic actuation of any engineered safety feature (ESF), including the reactor protection system (RPS)", the attached Licensee Event Report LER 96-002 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
Mr. Allen R. Johnson (Mail Stop 14B2)
PWR Project Directorate I-1
Washington, D.C. 20555

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NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 04/30/98
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT

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TITLE (4)

Secondary Transient, Caused by Loss of "B" Condenser Circulating Water Pump, Results in Manual 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
03	07	96	96	-- 002 --	00	04	08	96	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	1
POWER LEVEL (10)	97

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)		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

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

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

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

On March 7, 1996, at approximately 1814 EST, with the plant at approximately 97% steady state reactor power (Mode 1), the "B" condenser circulating water pump tripped. This resulted in a reduction of main condenser heat removal capability. Immediate action was to decrease turbine load to less than 50%, as per procedure AP-CW.1. However, due to condenser backpressure increasing above the limit for the satisfactory operating region for the main turbine, at approximately 1822 EST, the Shift Supervisor conservatively ordered a manual reactor trip. The Control Room operators performed the actions of procedures E-0 and ES-0.1. Following the reactor trip, all systems operated as designed, and the reactor was stabilized at hot shutdown conditions (Mode 3).

The underlying cause of the tripping of the "B" condenser circulating water pump was due to actuation of the power factor protection relay, which tripped the circuit breaker for the pump. The cause of the reactor trip was manual operator action.

This event is NUREG-1022 Cause Code (E).

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:

The main circulating water system supplies cooling water to the main condensers to condense steam exhausted from the two low pressure turbines. The system consists of two headers, each of which is supplied by a circulating water (CW) pump. Each header supplies a main condenser. The headers are cross-connected upstream of the main condensers to allow for reduced power operations with a single operating CW pump. After passing through the main condensers, CW is returned to the lake via a common discharge canal.

The plant was at approximately 97% steady state reactor power (Mode 1) with no significant activities in progress. On March 7, 1996, at approximately 1814 EST, the Control Room operators received Main Control Board Annunciator J-16 (Motor Off CW-EH Emerg Oil Seal Oil BU), caused by the trip of the "B" CW pump.

The trip of the "B" CW pump was followed by closure of the discharge valve for the "B" CW pump. This resulted in a decrease in total CW flow and an imbalance in CW flow to the two main condensers. This caused a vacuum imbalance between the main condenser hotwells. (There was a loss of approximately 40% of the main condenser heat removal capability due to the CW pump trip.) Since the condensers are connected, the difference in backpressure resulted in condensate water being "pushed" from the "B" hotwell to the "A" hotwell. Voiding of the "B" hotwell resulted in reduction of Net Positive Suction Head (NPSH) to the condensate pumps, thus also reducing main feedwater (MFW) pump suction pressures.

These abnormal conditions developed because the "B" CW pump had tripped rather than having been secured as part of an orderly transition to single CW pump operation. Although the plant can operate at up to 50% power on a single CW pump, the system must first be reconfigured; specifically, the discharge isolation valve for the pump to be secured must be closed before the CW pump is stopped. In this case, the discharge isolation valve was initially open when the "B" CW pump tripped. As a result, cross-connected flow from the "A" CW pump discharged back to the idle "B" CW pump rather than being forced through the "B main condenser, until the discharge isolation valve for the "B" CW pump closed.

The Control Room operators observed that the "B" CW pump had tripped, entered Abnormal Operating Procedure AP-CW.1 (Loss of a CW Pump), and performed the appropriate actions. A turbine load reduction was initiated. Within three minutes turbine load had been reduced to less than 50%. At this point, the load reduction was stopped. Reactor power was at 69%, when the load reduction was stopped, and continued to decrease. Due to reduced CW flow and loss of hotwell level in the "B" hotwell, "B" condenser backpressure increased above the limit for the satisfactory operating region for the main turbine.

II. DESCRIPTION OF EVENT:

A. DATES AND APPROXIMATE TIMES OF MAJOR OCCURRENCES:

- March 7, 1996, 1814 EST: "B" condenser circulating water (CW) pump trips.
- March 7, 1996, 1822 EST: Event date and time.
- March 7, 1996, 1822 EST: Discovery date and time.
- March 7, 1996, 1822 EST: Control Room operators manually trip the reactor, verify both reactor trip breakers open, and verify all control and shutdown rods inserted.
- March 7, 1996, 1834 EST: Control Room operators manually close both main steam isolation valves to limit a reactor coolant system cooldown.
- March 7, 1996, 1848 EST: Plant is stabilized at hot shutdown condition (Mode 3).

B. EVENT:

On March 7, 1996, at approximately 1818 EST, reactor power was at approximately 69% and still decreasing. Due to the loss of the "B" condenser circulating water (CW) pump there was reduced cooling water flow in the "B" main condenser for a period of time. This resulted in an imbalance in condenser cooling in the hotwells as backpressure increased in the "B" condenser, leading to a vacuum imbalance between the "A" and "B" condensers. "B" hotwell level indication showed no level, and "A" hotwell level indication showed high hotwell level. In addition, as per design, condenser steam dump was in operation after the rapid turbine load reduction to decrease reactor coolant system (RCS) temperature to match the turbine load.

The combination of reduced cooling water to the "B" condenser, decreased hotwell level, and steam dump into the hotwell caused the temperature in the "B" hotwell to increase. Part of the condensate pump suction was from the hotter water in the "B" hotwell causing a decrease in condensate pump flow and discharge pressure. Low condensate pressure resulted in an automatic start of the standby condensate pump and automatic opening of the condensate bypass valve.

Main feedwater (MFW) pump net positive suction head (NPSH) and suction pressure decreased due to this transient. The Control Room operators received Main Control Board Annunciators H-1 (Feed Water Pump Lo Suct Press 185 PSI) and H-17 (Feed Pump Net Positive Suction Head) and responded appropriately to these alarms.

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The secondary system transient affected the control signal from the Advanced Digital Feedwater Control System (ADFCS) to the main feedwater regulating valves (MFRV), calling for valve opening due to shrinking steam generator (SG) levels. During the time of the valve open demand, MFW pump suction pressure recovered, resulting in a large increase in feedwater flow. This large flow continued until levels in the "A" and "B" SGs reached the MFW Isolation setpoint of 67%, at which time the "A" and "B" MFRVs closed per design. Beginning at approximately 1819 EST on March 7, MFW Isolation occurred four (4) times for the "A" SG and three (3) times for the "B" SG. Within two minutes, the valve open demand signal from ADFCS moderated, MFW flow stabilized, and SG levels started to stabilize.

Procedure AP-CW.1 provides guidance for operation with elevated condenser backpressure. (The backpressure limit provides protection for the low pressure turbine last stage blading.) When backpressure increased above the limit for the satisfactory operating region and remained there for five (5) minutes, the shift supervisor ordered a manual reactor trip. 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 performance of ES-0.1, the Control Room operators noted that a reactor coolant system (RCS) cooldown was occurring and manually closed both main steam isolation valves (MSIV) at approximately 1834 EST. This action mitigated the RCS cooldown.

Pressurizer (PRZR) level decreased to a low level of approximately 8% during the RCS cooldown, automatically closing the letdown isolation valves when level decreased below 13%. PRZR level was increased above 13% within five (5) minutes, and letdown was subsequently reinstated by the Control Room operators.

The plant was stabilized in Mode 3 (hot shutdown) (at approximately 1848 EST) using Plant Operating Procedures O-2.1 (Normal Shutdown to Hot Shutdown) and O-3 (Hot Shutdown with Xenon Present). The "B" CW pump was restored to service at approximately 2038 EST on March 8, 1996, after completion of an inspection of the associated protective relays, circuit breaker, and electrical circuitry from the circuit breaker to the motor.

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

None

D. OTHER SYSTEMS OR SECONDARY FUNCTIONS AFFECTED:

None

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

E. METHOD OF DISCOVERY:

The trip of the "B" CW pump was immediately apparent to the Control Room operators due to Main Control Board (MCB) Annunciator J-16 and MCB indicating lights for the "B" CW pump. The reactor trip was manually initiated and was confirmed by plant response, alarms, and indications in the Control Room.

F. OPERATOR ACTION:

The Control Room operators promptly identified the loss of the "B" CW pump and performed the appropriate actions of AP-CW.1. The shift supervisor conservatively ordered a reactor trip when condenser backpressure increased above the limit for the satisfactory operating region.

After the reactor trip, the Control Room operators performed the appropriate actions of procedures E-0 and ES-0.1. The MSIVs were manually closed approximately twelve (12) minutes after the trip to limit further RCS cooldown. Appropriate actions were taken to restore levels in the "A" and "B" SGs and to increase PRZR level. When PRZR level was increased, letdown was manually restored to service. The plant was stabilized in Mode 3 (hot shutdown).

Subsequently, the Control Room operators notified higher supervision and the NRC per 10CFR50.72 (b) (2) (ii), non-emergency four hour notification, at approximately 2150 EST on March 7, 1996.

G. SAFETY SYSTEM RESPONSES:

All safeguards equipment functioned properly. All three auxiliary feedwater (AFW) pumps started when SG levels decreased below 17% after the reactor trip.

III. CAUSE OF EVENT:

A. IMMEDIATE CAUSE:

The immediate cause of the reactor trip was manual trip initiation, ordered by the Shift Supervisor due to elevated condenser backpressure.

B. INTERMEDIATE CAUSE:

The intermediate cause of the elevated condenser backpressure was a trip of the "B" CW pump.

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

C. ROOT CAUSE:

The underlying cause of the trip of the "B" CW pump was the tripping of the power factor protective relay for the "B" CW pump motor. This relay was activated due to a decrease in the power factor for this motor.

- The "B" CW pump motor was initially operating at a reduced lagging power factor.

- Long term oxidation of the variable auto transformer contact surfaces was noted, which reduced DC current to the motor field, thus reducing the power factor.

The combination of these factors caused the power factor protective relay to activate at its design setpoint.

This event is NUREG-1022 Cause Code (E), "Management / Quality Assurance Deficiency".

The tripping of the "B" CW pump meets the NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants", definition of a "Maintenance Preventable Functional Failure".

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 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 manual reactor trip is an actuation of the RPS, and MFW Isolation and AFW pump starts are actuations of an ESF.

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 trip of the "B" CW pump and subsequent manual reactor trip because:

- The two reactor trip breakers opened as required.

- All control and shutdown rods inserted as designed.

- The plant was stabilized at Mode 3 (hot shutdown).

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

- The Ginna Updated Final Safety Analysis Report (UFSAR) transient, "Loss of Condenser Vacuum", can occur from failure of the circulating water system, as stated in Section 15.2.4 of the UFSAR. In the event of loss of condenser vacuum, the turbine will be tripped, and, therefore, the event is bounded by the turbine trip event (UFSAR Section 15.2.2). This UFSAR transient was examined and compared to the plant response for the actual event. The plant behavior was found to be consistent with the assumptions detailed in the accident analysis. As described in Section 15.2.2.4, a loss of load with or without a direct or immediate reactor trip presents no hazard to the integrity of the RCS or the main steam system. The integrity of the core is maintained by operation of the reactor protection system prior to exceeding any thermal design limits.

- The total time of operation of the turbine with elevated condenser backpressure did not exceed the recommendations of the turbine manufacturer.

The Ginna 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:

- PRZR pressure decreased below 2205 PSIG during the transient prior to the reactor trip. During this time a thermal power ramp load reduction was in progress within the limits of LCO 3.4.1. Therefore, compliance with ITS was maintained. The RCS temperature DNB limit (577.5 degrees F) was not approached.

- After the reactor trip, the RCS cooled down to approximately 535 degrees F and was subsequently stabilized at 547 degrees F. The cooldown was within the limits of LCO 3.4.3. In addition, the required shutdown margin was maintained at all times during the RCS cooldown.

- Both SG levels decreased following the reactor trip to 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, both coolant loops were inoperable, even though both loops were still in operation and performing their 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. Both loops were restored to operable status when SG levels were restored to >/= 16% ("A" SG level in less than five (5) minutes, and "B" SG level in less than four (4) minutes). As required by LCO 3.4.5 Required Actions C.1, C.2, and C.3, the reactor trip breakers were open, the CRDMs were deenergized, no operation involving a reduction in RCS boron concentration occurred, and actions to restore both loops to operable status were immediately taken during the time SG levels were <16%. Therefore, compliance with LCO 3.4.5 was maintained.

Based on the above and the review of post trip data and past plant transients, it can be concluded 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 SGs were restored to operable status when SG level increased above 16% levels, by addition of auxiliary feedwater. Subsequently, levels were returned to their normal operating levels.
- PRZR level was increased to its normal operating level, and letdown flow was restored.
- The power factor protective relay for the "B" CW pump motor was inspected and tested satisfactorily.
- Thermography was performed on the "B" CW pump motor circuit breaker and internal components. This thermography indicated there was high resistance on the variable auto transformer wiper contact. The transformer was cleaned, and resistance was restored to acceptable values.
- CW motor test data was reviewed and found satisfactory.

B. ACTION TAKEN OR PLANNED TO PREVENT RECURRENCE:

- Procedure AP-CW.1 has been quarantined due to content that does not adequately control plant response. In the interim, Operations management has directed that upon loss of a CW pump, the reactor will be tripped, and further actions will be as per Procedure E-0.
- The loss of CW pump transient will be evaluated, and changes will be made to procedure AP-CW.1 prior to removing AP-CW.1 from quarantine.
- A maintenance procedure has been revised to have the electrical parameters of the motors for both CW pumps monitored by electricians, and adjustments to power factor will be made under rules in the Work Control System.
- A preventive maintenance repetitive task (REPTASK) has been initiated to ensure periodic cleaning of the variable auto transformer wiper contacts (during CW pump breaker maintenance).

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VI. ADDITIONAL INFORMATION:

A. FAILED COMPONENTS:

The "B" CW pump motor is a Westinghouse "Life Line" series motor, Model # 110P662H01, frame size HR-111-SPL, rated for 4000 volts and 1750 horsepower.

B. PREVIOUS LERs ON SIMILAR EVENTS:

A similar LER event historical search was conducted with the following results:

LER 85-019 was a similar event (loss of CW pump caused a plant transient, resulting in an automatic reactor trip) with a different root cause for the loss of the CW pump. The corrective action to prevent recurrence would not have prevented LER 96-002.

LER 95-008 was a similar event (loss of CW pump caused a plant transient, resulting in a manual reactor trip) with a different root cause for the loss of the CW pump. The corrective action to prevent recurrence would not have prevented LER 96-002.

C. SPECIAL COMMENTS:

None