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ROBERT C. MECREDY
Vice President
Nuclear Operations

October 18, 2000

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

Subject: LER 2000-001, Intermediate Range Channel Loss of Control Power, Due to Excessive Signal Noise, Results in Reactor Trip
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Vissing:

The attached Licensee Event Report LER 2000-001, 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

Robert C. Mecredy

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

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
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U.S. NRC Ginna Senior Resident Inspector

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NRC FORM 366 (6-1998)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001

LICENSEE EVENT REPORT (LER)

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

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TITLE (4)

Intermediate Range Channel Loss of Control Power, Due to Excessive Signal Noise, Results in 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
09	18	2000	2000	- 001 -	00	10	18	2000		05000
									FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	3
POWER LEVEL (10)	000

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 EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
B	IG	AMP	W120	Y					

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 September 18, 2000, at approximately 0503 EDST, the plant was in Mode 3 with the reactor coolant system being maintained at a temperature between 540 degrees F and 547 degrees F and a pressurizer pressure of approximately 2235 psig. A planned plant shutdown was in progress, in preparation for beginning the 2000 refueling outage. During this shutdown, a fuse blew in a nuclear instrument system intermediate range circuit, causing a reactor trip.

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.

Immediate corrective action was taken to stabilize the plant in Mode 3.

The cause of the blown fuse was excessive signal noise in the intermediate range circuit.

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 September 18, 2000, a planned plant shutdown was in progress, using normal operating procedure O-2.1, "Normal Shutdown to Hot Shutdown", in preparation for beginning the 2000 refueling outage. The plant was subcritical in Mode 3 and control rods were being inserted into the core to complete the reactor shutdown for the 2000 refueling outage. Rods for Control Banks "D", "C", and "B" had been completely inserted (to zero steps), and Bank "A" rods were being driven in. The reactor coolant system (RCS) was being maintained at a temperature between 540 degrees F and 547 degrees F and a pressurizer pressure of approximately 2235 psig. Although officially in Mode 2 as conservatively documented in the Official Record, the reactor was, in fact, substantially subcritical and in Mode 3. Reactor power was very low in the intermediate range and was continually decreasing due to the large amount of negative reactivity being added by control rod insertion.

At approximately 0502 EDST, Bank "A" rods had been inserted from the full out position to approximately 70 steps. Reactor power, as indicated on Nuclear Instrument System (NIS) intermediate range (IR) channels, was approaching the permissive P-6 reset setpoint of 5E-11 amps. Permissive P-6 is automatically reset from NIS IR channels when 2 of 2 IR channels decrease below 5E-11 amps. Reset of P-6 automatically re-energizes the NIS source range (SR) channels.

II. DESCRIPTION OF EVENT:

A. DATES AND APPROXIMATE TIMES OF MAJOR OCCURRENCES:

- September 18, 2000, 0503 EDST: Event date and time.
- September 18, 2000, 0503 EDST: Discovery date and time.
- September 18, 2000, 0503 EDST: Control Room operators verify both reactor trip breakers open and verify all control and shutdown rods inserted.
- September 18, 2000, 0507 EDST: Control Room operators manually close both main steam isolation valves to limit a reactor coolant system cooldown.
- September 18, 2000, 0531 EDST: Plant is stabilized in Mode 3.

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B. EVENT:

At approximately 0502 EDST, Bank "A" rods had been inserted from the full out position to approximately 70 steps. Reactor power, as indicated on Nuclear Instrument System (NIS) intermediate range (IR) channels, was approaching the permissive P-6 reset setpoint of 5E-11 amps. Permissive P-6 is automatically reset from NIS IR channels when 2 of 2 IR channels decrease below 5E-11 amps. Reset of P-6 automatically re-energizes the NIS source range (SR) channels.

The P-6 reset setpoint was reached at approximately 0503 EDST. At the same time as the change of state of the P-6 bistable for NIS IR channel N-36, a control power fuse blew in channel N-36, resulting in loss of control power to the N-36 channel. This loss of power de-energized the NIS IR high flux trip reactor trip relay for channel N-36 and the reactor tripped on 1 of 2 NIS IR high flux range trip. In addition to numerous Main Control Board (MCB) annunciators already in alarm from the ongoing plant shutdown, the Control Room operators acknowledged MCB annunciator D-18 "Intermediate Range Reactor Trip 1 / 2 25%", indicating a reactor trip from NIS IR channel N-36.

The two reactor trip breakers opened as designed and all shutdown and control rods that were withdrawn inserted as designed.

The Control Room operators performed the appropriate 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, steam generator blowdown flow was causing a reactor coolant system (RCS) cooldown. Both main steam isolation valves (MSIVs) were manually closed to limit the RCS cooldown.

The plant was stabilized in Mode 3 at approximately 0531 EST and the Control Room operators transitioned back to normal plant operating procedure O-2.1.

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The reactor trip (referred to as "scram" in NRC Performance Indicators) occurred when the reactor was subcritical. Thus, this scram did not meet the definition for the NRC Performance Indicator (PI) "Unplanned Scrams Per 7,000 Critical Hours". The scram also did not meet the definition for the NRC PI "Scrams With a Loss of Normal Heat Removal" since the reactor was subcritical prior to the scram and the normal heat removal paths (as listed in NEI 99-02,Revision 2) were removed due to intentional operator actions.

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

None

D. OTHER SYSTEMS OR SECONDARY FUNCTIONS AFFECTED:

None

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 closed to limit a RCS cooldown, and 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 0647 EDST on September 18, 2000.

G. SAFETY SYSTEM RESPONSES:

None. For Maintenance Rule purposes, this event is classified as a Functional Failure. However, it does not meet the definition for the NRC PI "Safety System Functional Failure": "any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems . . . ". The NIS IR high flux trip is a backup to the NIS power range low range flux trip and is not credited in any safety analysis. In addition, the loss of control power resulted in the reactor protection system going to its fail-safe (de-energized) condition and tripping the reactor.

III. CAUSE OF EVENT:

A. IMMEDIATE CAUSE:

The immediate cause of the reactor trip was achieving the 1 of 2 reactor protection system (RPS) trip logic for NIS IR high flux trip on channel N-36.

B. INTERMEDIATE CAUSE:

The intermediate cause of achieving 1 of 2 RPS trip logic was de-energizing the reactor trip relay for NIS IR channel N-36 high flux, due to a blown control power fuse.

C. ROOT CAUSE:

The underlying cause of blowing of the control power fuse for channel N-36 was excessive signal noise in the N-36 detector circuit. This noise was attributed to high AC ripple on the output of the logarithmic current amplifier (log current amplifier) in the N-36 drawer. This AC ripple caused the current signal (nuclear flux signal) output from the faulted log current amplifier to fluctuate. When the permissive P-6 setpoint was approached, this cycled the permissive P-6 bistable excessively. This changed the state of the P-6 bistable, rapidly and repeatedly energizing and de-energizing the P-6 relay, which resulted in blowing the control power fuse.

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

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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 reactor trip was an 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 operational or safety consequences or implications attributed to the blown fuse resulting in reactor trip because:

- The two reactor trip breakers opened as required.
- All control and shutdown rods that were withdrawn inserted as designed.
- The plant was stabilized in Mode 3.
- The plant was already shutdown with the reactor subcritical, so there were no power, temperature or pressure transients related to the reactor trip.

Based on the above and a 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 Control Room operators performed the appropriate actions of Emergency Operating Procedures E-0 and ES-0.1 and the plant was stabilized in Mode 3.
- The blown fuse in NIS IR channel N-36 was replaced.

B. ACTION TAKEN OR PLANNED TO PREVENT RECURRENCE:

NOTE: There are no NRC regulatory commitments in this Licensee Event Report.

- The faulted log current amplifier in the NIS SR N-36 drawer was replaced.
- Calibration procedures for NIS IR channels N-35 and N-36 will be revised to require a measurement of the AC ripple on the output of the log current amplifier.
- The log current amplifier for NIS IR channel N-35 was checked for AC ripple. Noise levels were acceptable.

VI. ADDITIONAL INFORMATION:

A. FAILED COMPONENTS:

The log current amplifier is part number 2372A27G01, manufactured by Westinghouse Electric Corporation.

B. PREVIOUS LERs ON SIMILAR EVENTS:

A similar LER event historical search was conducted with the following results: LER 90-003 and LER 1999-008 were similar events with a similar root cause.

C. SPECIAL COMMENTS:

None

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D. IDENTIFICATION OF COMPONENTS REFERRED TO IN THIS LER:

COMPONENT	IEEE 803 FUNCTION	IEEE 805 SYSTEM IDENTIFICATION
log current amplifier	AMP	IG
fuse	FU	IG
control rod	ROD	AA
nuclear instrument system	JIC	IG
main steam isolation valve	ISV	SB
reactor trip breaker	52	JC