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FACIL:50-244 Robert Emmet Ginna Nuclear Plant, Unit 1, Rochester G 05000244
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SUBJECT: LER 87-007-00:on 871218,discovery of apparent design inadequacy caused potential loss of core cooling.
W/8 ltr.

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TITLE: 50.73 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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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 0 6

TITLE (4) Discovery Of Apparent Design Inadequacy Causes Potential For Loss of Core Cooling During The High Head Recirculation Phase

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 8	8 7	8 7	– 0 0 7 –	0 0	0 1	1 7	8 8		0 5 0 0 0
										0 5 0 0 0

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

OPERATING MODE (9)	N							
POWER LEVEL (10)	1 0 0		20.402(b)		20.405(c)		50.73(a)(2)(iv)	73.71(b)
			20.405(a)(1)(i)		50.36(c)(1)	X	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	
Joseph A. Widay, Technical 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	

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 fifteen single-space typewritten lines) (16)

On December 18, 1987 during the review of a Westinghouse Corporation letter entitled "Operating Plant Feedback - Non-vital Power Supply Used in Valve Interlock Logic," it was discovered that the potential existed for a loss of core cooling during the High Head Recirculation Phase. Even though the review determined no susceptability to the condition as described in the referenced letter, further evaluation revealed the below described deficiency.

The apparent root cause of the event was identified as a design flaw, in that a common power supply was utilized to power a motor operated valve on each train of the High Head Recirculation System. A postulated failure of the electrical power supply prior to opening of the subject valves would result in both flow paths leading to the safety injection and containment spray pumps being blocked, creating potential loss of core cooling. (See Ginna USFAR for configuration description.)

Corrective action to prevent recurrence was to position the affected valve in the one-of-two series arrangement to the open position, thereby eliminating the potential for common mode failure.

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NRC Form 366 (9-83)

I. PRE-EVENT PLANT CONDITIONS

The unit was at 100% Reactor Power at the time of the discovery.

II. DESCRIPTION OF EVENT

A. EVENT:

On December 18, 1987 during the review of a Westinghouse Corporation letter entitled "Operating Plant Feedback - Non-vital Power Supply Used in Valve Interlock Logic," it was discovered that the potential existed for a loss of core cooling during the High Head Recirculation Phase. Even though the review determined no susceptability to the condition as described in the referenced letter, further evaluation revealed the below described deficiency.

The apparent root cause of the event was identified as a design flaw, in that a common power supply was utilized to power a motor operated valve on each train of the High Head Recirculation System. A postulated failure of the electrical power supply prior to opening of the subject valves would result in both flow paths leading to the safety injection and containment spray pumps being blocked, creating potential loss of core cooling. (See Ginna USFAR for configuration description.)

The Residual Heat Removal (RHR) discharge valves MOV 857A, B, C leading to the safety injection and containment spray pump suctions, are normally closed. MOV 857A is in the 6" line from RHR pump "B". MOV 857B and C are series valves in the 6" line from RHR pump "A". MOV 857A is powered from Bus 14 through MCC 1C position 7M. MOV 857B is powered from Bus 16 through MCC 1D position 7M and MOV 857C from Bus 14 through MCC 1C position 15J. RHR pumps "A" and "B" are powered by Bus 14 and 16, respectively.

There is an interlock for MOV 857A, B, C which prevents their opening if MOV 850A or MOV 850B are open and AOV 897 and AOV 898 are both open and MOV 896A and MOV 896B are both open. This prevents discharging water from sump B into the Refueling Water Storage Tank (RWST) during sump recirculation. MOV 857A also has a pressure interlock which prevents its opening when system pressure exceeds 250 psi. This prevents overpressurizing the safety injection suction piping.

During the sump recirculation phase following a LOCA, if high head recirculation is necessary, the 857A, B, C valves must be opened. Present procedure ES-1.3, "Transfer To Cold Leg Recirculation" calls for alignment to the sump and discharging to the reactor through MOVs 852A and 852B. If high head recirculation is required due to reactor coolant system pressure being above the discharge capabilities of the RHR pumps, the 857 valves must be opened. A failure of Diesel Generator (D/G) 1A prior to opening the 857 valves would result in both trains leading to the safety injection/containment spray pumps being blocked and potential loss of core cooling.

Procedure changes have been submitted to require positioning MOV 857C in the open position during normal safeguards system alignment. This action provides assurance of at least one flow path (assuming single failure criteria) during the high-head recirculation phase.

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

None

C. DATES AND APPROXIMATE TIMES FOR MAJOR OCCURRENCES:

o December 18, 1987, 1000 EST: discovery date and time

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

D. OTHER SYSTEMS OR SECONDARY FUNCTIONS AFFECTED:

As noted, only the potential loss of core cooling due to the failure of the 857 valves was identified. All other valves within these systems were provided with proper train separation.

E. METHOD OF DISCOVERY:

Self-identified through an engineering assessment of a Westinghouse letter regarding the non-vital power supply used in valve interlock logic. Even though there was no susceptability to the condition as described in the letter, further evaluation revealed the above described design inadequacy.

F. OPERATOR ACTION:

None.

G. SAFETY SYSTEM RESPONSES:

None.

III. <u>APPARENT CAUSE OF EVENT</u>

A. ROOT CAUSE:

The root cause of the event was determined to be an apparent design inadequacy in that a common mode failure renders both trains of core cooling for the condition of high head recirculation inoperable.

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

IV. ANALYSIS OF EVENT

This event is reportable in accordance with 10 CFR 50.73, Licensee Event Report System, item (a)(2)(v)(D) "any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to mitigate the consequences of an accident."

An assessment was performed considering both the safety consequence and implications of this event with the following results and conclusions:

- The RHR pump and 857 valves for each train are not powered by the same source. Pump "A" (Bus 14) is in the line with MOVs 857B (MCC 1D) and MOV 857C (MCC 1C). Pump "B" (Bus 16) is in the line with MOV 857A (MCC 1C). With this configuration a failure of D/G 1A just prior to opening of 857 valves will cause RHR pump "A" to stop while 857A (Train "B") and 857C (Train "A") could not be opened from the control room. Although pump "B" would remain operable in this case, Train "B" would be unavailable since MOV 857A could not be opened. Manual operator action would be required to mitigate the consequences due to the above described failure.

- Failure of MOVs 857A and 857C to stroke to the open position due to the common power supply failure could be alleviated through operator action by manually opening the affected valves. This action, though possible, is not in the interest of ALARA and was not provided in existing procedures.

- An alternate line has been provided so that high head recirculation can continue in the event that the recirculation flowpath downstream of the 857 valves is unavailable. This design can also be utilized to provide a flow path to the 1C safety injection pump piping in the event of failure to open the 857 valves. Two normally closed manual valves in series,

1816A and B, in a separate 4" line upstream of MOV 857B and C can provide core cooling from the "A" train of the RHR system to the 1C safety injection pump suction. This action is recognized to mitigate valve opening failure only, and not entire train failure.

V. CORRECTIVE ACTION

A. ACTION TAKEN OR PLANNED TO PREVENT RECURRENCE:

- MOV 857C has been placed in the open position during normal safeguards alignment to eliminate the potential for common mode failures.
- The Emergency Operating Procedures have been changed, validated, and verified to reflect the new normal safeguards system alignment.
- An Engineering Work Request has been generated to evaluate the current configuration design.

VI. ADDITIONAL INFORMATION

A. FAILED COMPONENTS:

None.

B. PREVIOUS LERs ON SIMILAR EVENTS:

A similar LER event historical search was conducted with the following results: No documentation of similar LER events could be identified.

C. SPECIAL COMMENTS:

- The corrective action planned will be tracked by Corrective Action Report (CAR) # 1831.
- Westinghouse Electric Corporation has been notified of the above apparent design deficiency.

ROCHESTER GAS AND ELECTRIC CORPORATION • 89 EAST AVENUE, ROCHESTER, N.Y. 14649-0001

TELEPHONE
AREA CODE 716 546-2700

January 18, 1988

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

Subject: LER 87-007 Discovery Of Apparent Design Inadequacy Due To Susceptibility To A Common Mode Power Supply Failure Causes Potential For Loss of Core Cooling During The High Head Recirculation Phase
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

In accordance with 10 CFR 50.73, Licensee Event Report System, item (a)(2)(v)(D) which requires a report of, "any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to mitigate the consequences of an accident", the attached Licensee Event Report LER 87-007 is hereby submitted.

This event has in no way affected the public's health and safety.

Very truly yours,

Bruce A. Snow

Bruce A. Snow

xc: U.S. Nuclear Regulatory Commission
Region I
631 Park Avenue
King of Prussia, PA 19406

Ginna USNRC Resident Inspector

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