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, Unit No. 1	0 5 0 0 0 2 4 4	1 OF 0 3

TITLE (4)
Potential Loss of Residual Heat Removal (RHR) Capability

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)
0 3	0 7	8 4	8 4	- 0 0 3	- 0 0	0 4	0 6	8 4		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
POWER LEVEL (10)	0 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
G. F. Larizza, Operation Manager	AREA CODE 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	
A	B P	- I S V	D 0 2 5	N							

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-spaced typewritten lines) (16)

On March 7, 1984, while the Reactor was in the cold shutdown condition, the draindown of the Reactor Coolant System (RCS) was in progress in preparation for the Steam Generators (S/G) Annual Inspection.

In the process of draining the RCS to the CVCS Holdup Tanks, while preparing to shift from draining via the Reactor Coolant Drain Tank (RCDT) pump to the Low Pressure Purification Pump, valves MOV-851A and B (Containment Sump B Suction to RHR) were mistakenly opened prior to shutting valve MOV-850A (downstream of MOV-851A and upstream of RCDT pump suction). This resulted in water being drained from the RCS Loop to the Sump B, with potential loss of RHR capability.

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

On March 7, 1984, while the Reactor was in the cold shutdown condition, the draindown of the Reactor Coolant System (RCS) was in progress in preparation for Steam Generator (S/G) Annual Inspection.

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The Operator's first action was to turn the switch for MOV-850A to the "Closed" position. Approximately 30 seconds later the Sump B level alarm was actuated and the 8" level indicator lights were actuated. The operator then noticed that the RCS loop level monitor indicated zero. The running RHR pump was then stopped. To this point the RHR flow had remained at a steady 900 GPM. After MOV-850A was shut, the RHR pump was restarted and 900 GPM flow indicated. Approximately one minute later MOV-856 (Refueling Water Storage Tank to RHR Pump Suction and to RCS Hotleg Loop A) was opened to restore RCS loop level, then closed. The final pressurizer level indication was 150". Delaying the opening of MOV-856 until after the RHR pump was restarted and indicating steady flow, showed that at no time was RHR inoperable.

The RHR temperature prior to the event was approximately 100°F. When the pump was stopped the RHR temperature indicated approximately 115°F. When MOV-856 was opened the temperature decreased to approximately 60°F. When RHR flow was re-established the temperature returned to 100°F.

Both CAM's (continuous air monitors) in the Containment Vessel Basement were alarming on High Gaseous Activity. R-12 (Containment Vent Gaseous Monitor) indication in the Control Room did not increase significantly above the pre-event reading. The containment evacuation alarm was sounded and personnel inside containment evacuated.

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Subsequent investigation revealed that a total of 31 inches of water was drained into the Sump B (Sump B discrete light indications are at 8 inches and 78 inches.) This was verified by actual measurement. This volume of approximately 2000 gallons is consistent with the calculation of RCS loop level change from 20 inches to zero inches. The evaluation further showed that after opening valves MOV-851A and B the RHR pumps were still operable and operating with the 31 inches of water in the sump providing minimum NPSH requirement.

The event that could have caused loss of RHR capability was the closing of MOV-850A with essentially zero inches loop level indication. Had the RHR pumps operated in this mode for greater than or equal to one minute without MOV-856 open they could have become air bound and hence inoperable. It should be noted that the travel time for 851A, 851B, 850A, and 856 is approximately two minutes and calculations show that it would take less than 30 seconds to drain the volume of water from RCS (20" loop level to zero) to Sump B.

A review was made of Operating Procedure O-2.3.1 "Draining The Reactor Coolant System" and of O-2.2 "Plant Shutdown From Hot Shutdown to Cold Shutdown Condition" to see if clarifications were required. A minor change was made to procedure O-2.2 to clarify one step associated with MOV-851A and B.

Operations personnel have been cautioned on strict adherence to operating procedures.

Further discussions of this event will be included in future Operations Personnel and Shift Supervisor's meetings.

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

ROGER W. KOBER
VICE PRESIDENT
ELECTRIC & STEAM PRODUCTION

TELEPHONE
AREA CODE 716 546-2700

April 6, 1984

Dr. Thomas E. Murley, Regional Administrator
U.S. Nuclear Regulatory Commission
Region I
631 Park Avenue
King of Prussia, Pennsylvania 19406

Subject: LER 84-003, Potential Loss of Residual Heat Removal (RHR) Capability

R. E. Ginna Nuclear Power Plant, Unit No. 1
Docket No. 50-244

Dear Dr. Murley:

In accordance with 10 CFR 50.73, Licensee Event Report System, item (a)(2)(v), "any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to:

(a) Shutdown the Reactor and maintain it in a safe shutdown condition;

(b) Remove Residual Heat;

(c) Control the Release of Radioactive Material; or

(d) Mitigate the consequences of an accident."

The attached Licensee Event Report LER 84-003 is hereby submitted.

Very Truly Yours,

Roger W. Kober

Roger W. Kober

GFL/eeg

xc: Document Control Desk (1)

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