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 2

TITLE (4)
Automatic Actuation of Reactor Protection System

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 4	0 6	8 5	8 5	— 0 0 6	—	0 0	0 5	0 6	8 5		0 5 0 0 0
											0 5 0 0 0

OPERATING MODE (9)		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
	N	20.402(b)	20.405(c)	X 50.73(a)(2)(iv)	73.71(b)
POWER LEVEL (10)	0 1 5	20.405(a)(1)(i)	50.36(c)(1)	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
G.F. Larizza, Operations 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	
A	S J	- I - F I	B 0 8 0	N							

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

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

On April 6, 1985, while performing a plant startup from a refueling outage with reactor power at approximately 5%, a reactor trip occurred with a subsequent turbine trip. The reactor trip was the result of the "B" steam generator level, which was being manually controlled, reducing below 30% narrow range level while a feedwater flow less than steam flow bistable was in the tripped position, per the applicable steps of a calibration procedure. The steam generator level was being manually controlled between 30% and 39% while the calibration was being completed. All systems operated as designed and the reactor was stabilized at hot shutdown conditions.

IE22
1/1

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
R.E. Ginna Nuclear Power Plant	0 5 0 0 0 2 4 4	8 5	— 0 0 6	— 0 0	0 2 OF 0 2

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

At 1902 hours on April 6, 1985, during a plant startup from a refueling outage with reactor power at approximately 5%, a reactor trip occurred with a subsequent turbine trip. The reactor trip was due to low steam generator level with a feedwater flow less than steam flow bistable tripped. The Control Room Operators manually isolated the Main Steam Line Isolation Valves (MSIV) to limit the reactor coolant system cooldown. All systems operated as designed and the reactor was stabilized at hot shutdown conditions.

The reactor trip was the result of the "B" Steam Generator (SG) level, which was being manually controlled, reducing below 30% narrow range level while a feedwater flow (FF) less than steam flow (SF) bistable was in the tripped condition. One out of two low SG level channels plus one out of two FF less than SF channels in a given loop result in a reactor trip. The FF less than SF bistable (FC-477B) was placed in the tripped position in accordance with the applicable steps of procedure CP-477 (Calibration and/or Maintenance of Feedwater Flow Channel 477), which was being performed to calibrate a feedwater flow indicator (FI-477) used for the unit power calorimetric.

The steam generator water levels were being controlled between 30% and 39% at the time of the event, the 39% was due to a caution in the operating procedure O-1.2, which warns the Operator to avoid steam generator levels above 39% at low power due to the analysis for a steam line rupture. This narrow a band of level did not allow the Operator much room for control, as he was using the main feedwater regulating bypass valves and a main feedwater pump for level control. The turbine was being slowly brought up to synchronous speed which causes steam generator level fluctuations. Due to the large number of activities being performed by the Operators, they were not fully cognizant of the trip logic and level control demands, that would be imposed on the steam generator during the performance of the calibration procedure while the turbine was being warmed up. To prevent reoccurrence of this, the involved Operators have been reminded to give permission for calibrations and tests only after they are fully aware of the implications of the procedure.

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

May 6, 1985

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

Subject: LER 85-006, Automatic Actuation of the Reactor Protection System (RPS)
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 requests a report of, "any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)," the attached Licensee Event Report LER 85-006 is hereby submitted.

Very truly yours,

Roger W. Kober

Roger W. Kober

RWK/eeg

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

IE22
1/1