ACCELERATED DISTRIBUTION DEMONSTRATION SYSTEM

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9101160181 DOC.DATE: 91/01/11 NOTARIZED: NO DOCKET #
FACIL:50-244 Robert Emmet Ginna Nuclear Plant, Unit 1, Rochester G 05000244
AUTH.NAME AUTHOR AFFILIATION
BACKUS,W.H. Rochester Gas & Electric Corp.
MECREDY,R.C. Rochester Gas & Electric Corp.
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 90-017-00:on 901212,reactor trip relay de-energized & reactor tripped when dc switches in distribution panel opened.Caused by procedural inadequacy.Procedure change process being evaluated.W/910111 ltr.

DISTRIBUTION CODE: IE22T COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 16
TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

NOTES:License Exp date in accordance with 10CFR2,2.109(9/19/72). 05000244

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD1-3 LA	1	1	PD1-3 PD	1	1
	JOHNSON,A	1	1			
INTERNAL:	ACNW	2	2	AEOD/DOA	1	1
	AEOD/DSP/TPAB	1	1	AEOD/ROAB/DSP	2	2
	NRR/DET/ECMB 9H	1	1	NRR/DET/EMEB 7E	1	1
	NRR/DLPQ/LHFB11	1	1	NRR/DLPQ/LPEB10	1	1
	NRR/DOEA/OEAB	1	1	NRR/DREP/PRPB11	2	2
	NRR/DST/SELB 8D	1	1	NRR/DST/SICB 7E	1	1
	NRR/DST/SPLB8D1	1	1	NRR/DST/SRXB 8E	1	1
	REG FILE 02	1	1	RES/DSIR/EIB	1	1
	RGN1 FILE 01	1	1			
EXTERNAL:	EG&G BRYCE,J.H	3	3	L ST LOBBY WARD	1	1
	NRC PDR	1	1	NSIC MAYS,G	1	1
	NSIC MURPHY,G.A	1	1	NUDOCS FULL TXT	1	1

Cont NO
9249194212

NOTE TO ALL "RIDS" RECIPIENTS:

PLEASE HELP US TO REDUCE WASTE! CONTACT THE DOCUMENT CONTROL DESK, ROOM P1-37 (EXT. 20079) TO ELIMINATE YOUR NAME FROM DISTRIBUTION LISTS FOR DOCUMENTS YOU DON'T NEED!

FULL TEXT CONVERSION REQUIRED
TOTAL NUMBER OF COPIES REQUIRED: LTTR 31 ENCL 31

mb / A04

R I D S / A D D S

R I D S / A D D S

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

ROBERT C. [illegible]
Vice President
[illegible]

TELEPHONE
AREA CODE 716 546-2700

January 11, 1991

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

Subject: LER 90-017, Opening of DC Switches (Procedural Inadequacy) Disables Manual and Auto Actuation of Safeguards Sequence Initiation Causing a Condition Outside the Design Basis of the Plant
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

In accordance with 10 CFR 50.73, Licensee Event Report System, item (a)(2)(ii)(B), which requires a report of, "any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded, or resulted in the nuclear power plant being in a condition that was outside the design basis of the plant", the attached Licensee Event Report LER 90-017 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
Region I
475 Allendale Road
King of Prussia, PA 19406

Ginna USNRC Senior Resident Inspector

Cert No P249194212

IE22
1/1

9101160181 910111
PDR ADOCK 05000244
S PDR

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 1 5

TITLE (4) Opening of DC Switches Disables Manual and Auto Actuation of Safeguards Sequence Initiation, Causing a Condition Outside the Design Basis of the Plant

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 2	9 0	9 0	— 0 1 7 —	0 0	0 1	1 9	9 1		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 0 3

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

20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
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)	X 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
Wesley H. Backus Technical Assistant to the Operations 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	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE)   X NO

EXPECTED SUBMISSION DATE (15): MONTH DAY YEAR

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

On December 12, 1990, at 2310 EST, with the reactor at approximately 3% full power, the Control Room Foreman opened two DC switches, as directed by a Maintenance procedure, causing the disabling of manual (pushbutton) and automatic actuation of the safeguards sequence initiation.

The two DC switches were closed, as directed by the Maintenance procedure, approximately twenty (20) minutes later, restoring manual (pushbutton) and automatic actuation initiation.

The underlying cause of the event was procedure inadequacy due to insufficient attention to detail.

Extensive corrective actions are being taken to prevent recurrence, including communication of management expectations, HPES evaluations, identifying procedural inadequacies, and a comprehensive upgrade of the procedure change process.

NRC Form 366

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
R.E. Ginna Nuclear Power Plant	0 5 0 0 0 2 4 4	9 0	— 0 1 7	— 0 0	0 2 OF 1 5

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

I. PRE-EVENT PLANT CONDITIONS

The plant was in the process of starting up subsequent to the plant trip of 12/11/90 (discussed in LER 90-013). The reactor was at approximately 3% full power, awaiting clearance that secondary chemistry parameters were within specification.

The two Control Room reactor operators were providing full time attention to maintaining steam generator water level (i.e. using "manual" control of feedwater addition) and controlling reactivity due to a Xenon transient (i.e. with manual control of boron concentration). At approximately 2044 EST, December 12, 1990, the Control Room received Main Control Board Alarm L-14 (Bus 14 Under Voltage Safeguards). Due to this undervoltage failure, the "A" Emergency Diesel Generator automatically started. By design, it did not close into Bus 14 as Bus 14 was still powered by its normal supply. The Control Room operators dispatched auxiliary operators (AO) to check the Bus 14 Undervoltage Monitoring System Cabinets in the Auxiliary Building and Relay Room. One of the AO's reported to the Control Room operators that the Bus 14 Undervoltage Monitoring System Cabinet in the Auxiliary Building had a burnt odor and the relay indicating lights indicated that at least one relay was not operable. This event is discussed in LER 90-015.

The Operations Shift Supervisor (SS) notified station electricians of the above indications. The station electricians then checked the Bus 14 Undervoltage Monitoring System Cabinets and found a faulty solid state switch printed circuit board in the Auxiliary Building Bus 14 Undervoltage Monitoring System Cabinet. These findings were reported to the Control Room.

The Station Electrical Planner then initiated, "Work Request or Trouble Report" Number 9024136, reviewed applicable drawings and prepared a work package (i.e. work order number 9024136) which included PORC approved Maintenance procedure M-48.14 (Isolation of Bus 14 Undervoltage System for Maintenance, Troubleshooting, Rework and Testing). The work package was then reviewed by the Planner Scheduler for compliances with administrative requirements. The Electrical Planner then performed required notifications of the QC Engineer and a Results and Test technician, and briefed them on the contents of the work package. The Electrical Planner took the work package to the Electric Shop and reviewed it with the two electricians who were to perform the work.

The Electrical Planner and two electricians went to the Control Room and reviewed the M-48.14 procedure with the SS. The SS performed a review of the M-48.14 procedure to approve the steps for transferring electrical loads on Bus 14 to the "A" Emergency Diesel Generator. The SS then gave the M-48.14 procedure to the Control Room Foreman (CRF) to actually perform the operational steps. When the CRF began to read step 5.5.1 of M-48.14, he stopped and questioned the Electrical Planner concerning this step. Step 5.5.1 required the opening of two DC switches. The CRF was concerned with the effect of opening these two DC switches, given the current plant conditions. Therefore, the CRF and Electrical Planner reviewed the Initial Conditions of M-48.14, and re-verified that they were adhering to the procedure requirements. The Electrical Planner stated to the CRF that this procedure had been performed before.

II. DESCRIPTION OF EVENT

A. DATES AND APPROXIMATE TIMES OF MAJOR OCCURRENCES:

- o December 12, 1990, 2310 ST: Event Date and Time (i.e. SI DC switches opened)
- o December 12, 1990, 2330 EST: SI DC switches closed
- o December 13, 1990, 0102 EST: Event Discovery Date and Time
- o December 13, 1990, 0150 EST: Nuclear Regulatory Commission (NRC) notified via Emergency Notification System (ENS)

B. EVENT:

On December 12, 1990 at 2310 EST, with the reactor at approximately 3% full power, the CRF opened the two DC switches (required by step 5.5.1 of M-48.14) in the DC distribution panels on the back of the Control Board. The opening of these switches caused Control Board Alarm L-31 (Safeguards DC Failure) to annunciate. The receipt of this alarm was questioned at the time, but the response from the CRF was that the alarm was an expected result of performing step 5.5.1 of M-48.14. Further evaluation of this alarm was deferred to continue with M-48.14.

The CRF, continuing with M-48.14, opened the Bus 14 Normal Feed Breaker to allow the "A" Emergency Diesel Generator to tie into Bus 14. This action resulted in momentarily de-energizing the 1B Instrument Bus. The reactor trip relay from Nuclear Instrumentation System Intermediate Range Channel N-36 (powered from this Bus) de-energized and a reactor trip occurred. The reactor trip is discussed in LER 90-016.

The Control Room operators immediately performed the applicable actions of E-0 (Reactor Trip or Safety Injection) and ES-0.1 (Reactor Trip Response) and stabilized the plant in hot shutdown.

After completing the applicable steps of E-0 and ES-0.1, the Control Room operators completed their part of M-48.14, by closing the two DC switches that had been opened in step 5.5.1 of M-48.14. This was accomplished at approximately 2330 EST, December 12, 1990.

The oncoming SS, who had been in the Control Room during this event, resumed the evaluation of the consequences of alarm L-31 after plant conditions had stabilized. (The cause of the alarm had already been determined.) He performed another review of M-48.14 and called other knowledgeable members of the plant staff at their homes (at approximately 0100 EST, December 13, 1990) to discuss his concerns about the effect of opening these two DC switches. After receiving confirmation that his concerns were legitimate, he made the proper notifications to higher supervision and the Nuclear Regulatory Commission (NRC).

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

None.

D. **OTHER SYSTEMS OR SECONDARY FUNCTIONS AFFECTED:**

None.

E. **METHOD OF DISCOVERY:**

The event was made apparent during the oncoming SS review of the consequences of Control Board Alarm L-31 (Safeguards DC Failure) and subsequent discussions with knowledgeable plant staff.

F. OPERATOR ACTION:

Factors that influenced operator actions during the event were as follows:

- o The Control Room operators questioned step 5.5.1 in procedure M-48.14, but information in M-48.14, the DC switch labels, and Alarm Response procedure AR-L-31 did not provide sufficient operational information to determine the consequences of opening these two switches.

- o The Control Room operators had confidence in a Plant Operating Review Committee (PORC) approved procedure that had also been reviewed by the Electrical Planner.

As the event was over prior to discovery, no operator actions other than normal were performed.

G. SAFETY SYSTEM RESPONSES:

None.

III. CAUSE OF EVENT

A. IMMEDIATE CAUSE:

A condition outside the design basis of the plant was caused by the disabling of manual (pushbutton) and automatic actuation of the safeguards sequence initiation (i.e. auto and manual SI).

B. INTERMEDIATE CAUSE:

The disabling of manual (pushbutton) and automatic actuation of the safeguards sequence initiation was caused by switch #12 in the 1A DC Distribution Panel and switch #9 in the 1B DC Distribution Panel being open at the same time. Both of these panels are on the back of the Main Control Board.

Procedure M-48.14 had been initially written in 1983 for use at hot or cold shutdown conditions. In 1985, the procedure was modified to allow use only at cold shutdown conditions. The initial procedure and the revisions were technically correct and received a multi-disciplined review and approval. During 1988, M-48.14 received a major rewrite and Revision 6 became effective March 23, 1989. The purpose of the rewrite was to change the method of feeding Bus 14, and as part of this process the plant mode was changed to "Any Mode of operation." This change received a thorough review prior to becoming effective. One day later, it was erroneously concluded that certain steps, which had previously been in the procedure for use at cold shutdown, were inadvertently omitted from the current procedure. Thus, M-48.14 was changed to insert the current step 5.1.1. This change is inappropriate in modes other than cold shutdown, but this was not recognized during the review and approval process.

C. ROOT CAUSE:

The root cause was determined to be failure of the organization to attribute sufficient attention to detail in the procedure change process.

o A Maintenance procedure which was previously correct, was changed to require inappropriate actions, in that the procedure directed the opening of switches in the DC power supply to the SI sequences during all modes of operation.

o The procedure was reviewed by the Plant Operating Review Committee and approved for use by plant management, for all modes of plant operation, although the opening of the two DC switches was clearly intended for use at cold shutdown conditions to prevent spurious SI actuation during Bus transfer to the Emergency Diesel Generators.

- o A contributing factor is the need for a questioning attitude. The CRF, in questioning the performance of step 5.1.1 of M-48.14, did not go far enough with his questioning attitude.

IV. ANALYSIS OF EVENT

This event is reportable in accordance with 10 CFR 50.73, Licensee Event Report System, item (a)(2)(ii)(B), which requires a report of "any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded, or resulted in the nuclear power plant being in a condition that was outside the design basis of the plant", in that the disabling of manual (pushbutton) and automatic actuation of the safeguards sequence initiation placed the plant in a condition outside its design basis.

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

During the above event, manual (pushbutton) and automatic actuation of the safeguards sequence initiation was disabled, however, the various pumps and valves were operable and could be operated by the Control Board switches. The Control Room operators perform immediate actions upon reactor trip per procedure E-0. Through these procedural immediate actions the operators evaluate whether a condition requiring safety injection exists, and if required, verify operation of safeguards equipment or manually start and align that equipment. Their evaluation would be based upon the appropriate annunciator alarms (all of which were unaffected by the DC switch positions), or a review of control board parameter indications (i.e. RCS pressure, Steam Generator pressure, etc.).

The effect of the potential delay in actuating safeguards equipment upon those events analyzed in the UFSAR was evaluated. The accidents effected by this action are those accidents which result in depressurization of the primary system causing SI. These are primarily the following:

- o Feed Line Break (FLB)
- o Steam Generator Tube Rupture (SGTR)
- o Small Break Loss of Coolant Accident (SBLOCA)
- o Large Break Loss of Coolant Accident (LOCA)
- o Small Steam Line Break (Small SLB)
- o Large Steam Line Break (SLB)

An analysis of these accidents was performed to determine the effect of the disabling of manual (pushbutton) and automatic actuation of the safeguards sequence initiation with the following results:

<u>Feed Line Break</u>

This accident was analyzed by the Ginna Updated Final Safety Analysis Report (UFSAR) as a heat up event with auxiliary feedwater available in ten (10) minutes. As a heatup event, RCS pressure never decreased below the SI setpoint, but rapidly increased above the SI pump shutoff head. Therefore, SI was not necessary and auxiliary feedwater, when available within ten (10) minutes, is sufficient to mitigate the event. Operator actions to start auxiliary feedwater within ten (10) minutes is consistent with the Ginna licensing basis. If the FLB was re-evaluated as a cooldown event from 3% power the results would be bounded by a SLB.

Steam Generator Tube Rupture

SGTR is bounded by SBLOCA from the RCS depressurization standpoint. The leak rate from a SGTR is small compared to break flow for a SBLOCA. There is no significant effect due to lack of manual (pushbutton) or automatic SI since the main steps in the procedure deal with isolation of the ruptured SG, depressurization of the RCS, and termination of SI.

Small Break Loss of Coolant Accident

When manual (pushbutton) and automatic SI was de-activated, the reactor was operating at 3% power. The reactor had been at 3% power for approximately ten (10) hours. Prior to that, the reactor had been subcritical for twenty-two (22) hours following a trip.

Westinghouse Owner's Group letter WOG 90-113, dated July 2, 1990, "Shutdown LOCA Program - Draft Report", evaluated a mode 4 LOCA using a generic two (2) loop plant with a six (6) inch break assumed to occur two and a half (2.5) hours after shutdown. Acceptable results were obtained provided SI was started ten (10) minutes after the break. Assumptions of the mode 4 LOCA analysis are compared with the Ginna Event conditions below:

WOG MODE 4	GINNA EVENT
Decay Heat ~1.3%	Decay Heat ~0.86%
No accumulators available	Accumulators available
RCS pressure 1000 psig	RCS pressure 2235 psig
RCS temperature 425°F	RCS temperature 547°F

The availability of accumulators and the lower decay heat offset the higher RCS temperature and pressure. Sufficient time is available to manually start the SI and RHR pumps and open appropriate valves from the Control Room, and to recover from the SBLOCA. In any case, SBLOCA is bounded by LOCA because less time is available for operator action during a Large Break LOCA.

Large Break Loss of Coolant Accident

An assessment of disabling manual (pushbutton) and automatic SI at 3% power was performed by Westinghouse with respect to the LOCA analysis. The assessment assumed the RCS was at 547°F, 2235 psig. The fuel rods were assumed to be at 600°F which would be the approximate pellet and clad temperature at the end-of-blowdown phase. The vessel lower plenum and the lower portion of the core would be covered with accumulator water. Further, it was assumed that SI must be initiated when the fuel rods are at 1800°F to turn around the cladding temperature before it reaches 2200°F. Decay heat is based on an approximation of power history prior to the event, using the 1971 ANS Model. An adiabatic heatup calculation was performed using properties for a 14 x 14 array Optimum Fuel Assembly (OFA). The calculation indicated SI was necessary in 5.5 to 6 minutes. Simulations on the Ginna specific simulator indicate a 5 to 6 minute operator response during a LOCA is achievable.

Small Steam Line Break

This accident is bounded by the Large SLB because longer times are available for operator response.

Large Steam Line Break

Westinghouse assessed the effect of no manual (pushbutton) or automatic SI on the Steam Line Break analysis. Based on their experiences with Steam Line Break analysis as well as a review of the available margin to the acceptance criteria, it was judged that if the accident were re-analyzed at 3% power with no manual (pushbutton) or automatic SI, acceptable results would be obtained.

Rochester Gas and Electric Corporation (RG&E) performed a computer analysis of the SLB using the Westinghouse LOFTRAN Code. A base case was compared to a case where SI was delayed for ten (10) minutes. The comparison indicated negligible change in minimum DNBR. There was an insignificant change in mass released to containment because mass release is dominated by initial steam generator level and auxiliary feedwater flow, neither of which are affected by delayed SI. Comparing energy out the break for both cases, showed negligible differences. Therefore, delaying SI has negligible effect on minimum DNBR and mass/energy out the break.

In conclusion, delay of manual (pushbutton) and automatic SI with the reactor at 3% power would not cause Non-LOCA events to exceed the acceptance criteria. A delay of 5.5 to 6 minutes in the LOCA can be tolerated without unacceptable results. Based on operator training, this is sufficient time for operator response.

Based on the above, it can be concluded 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 affected system was restored to normal when the two (2) DC switches were closed twenty (20) minutes after they were opened.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
R.E. Ginna Nuclear Power Plant	0 5 0 0 0 2 4 4	9 0	— 0 1 7	— 0 0	1 3 OF 1 5

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

B. ACTION TAKEN OR PLANNED TO PREVENT RECURRENCE:

1. Short Term Actions

- o Senior Management met with key plant personnel to communicate management expectations for a questioning attitude and attention to detail. These groups included all the operating shifts, the Maintenance planning staff, and PORC members.

- o Policies were issued addressing the Operations Shift Supervisor review of procedures prior to giving authorization to proceed, and the Maintenance planners' responsibilities for Work Package review.

- o All plant procedures were screened for possible operational impact inadequacies, and potentially deficient procedures were made unavailable for use. Procedures that were made unavailable for use, but were immediately required for safe plant operation, were carefully reviewed prior to being made available for use.

- o A Human Performance Enhancement System (HPES) evaluation was performed on Control Room activities associated with this event, to identify the need for any additional short term corrective actions. Additional actions were identified. Actions identified by the HPES process were implemented where appropriate, including additional upgrades of switch labels, a more detailed review of Alarm Response procedures, improved wording of procedural steps for Human Factors concerns, and additional information to be made available to the operators.

- o For all involved operations personnel, their operating experience and training histories were reviewed for adequacy.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
R.E. Ginna Nuclear Power Plant	0 5 0 0 0 2 4 4	9 0	— 0 1 7 —	0 0	1 4 OF 1 5

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

- o An independent assessment of the status of short term actions, and of the procedure screening, was conducted before management authorized restart of the reactor.

2. Long Term Actions

- o An HPES evaluation of the procedure change process is being performed. From this evaluation, recommendations for long term improvements will be implemented. Among these improvements are the increased involvement of Operations in the review of proposed changes, requirements for more accurate descriptions of proposed changes, and more rigor in the review of changes at PORC meetings.

- o To ensure the effectiveness of the short term corrective actions, follow-up evaluations will be conducted. Based on these evaluations, management will determine the need for additional reinforcement of these actions.

- o The training programs for Operations personnel, PORC members, and Maintenance planners will be re-evaluated. We expect to make long term improvements in these programs, and in the content of these programs, and also in the training programs for other plant personnel.

- o Procedures that are currently unavailable for use will be reviewed in detail prior to release for use. We are expediting the review of those procedures, placing highest priority on those procedures which are expected to be needed during the course of routine operations, to ensure their availability in the near future.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
R.E. Ginna Nuclear Power Plant	0 5 0 0 0 2 4 4	9 0	— 0 1 7	— 0 0	1 5 OF 1 5

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

- o Policies that were initiated as a result of this event will be reviewed for consistency with pre-existing policies and procedures. Where appropriate, existing policies will be altered, superseding the new policies.

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 with the same root cause at Ginna Station could be identified.

C. SPECIAL COMMENTS:

None.