CATEGORY 1

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ACCESSION NBR:9612110450 DOC.DATE: 96/12/05 NOTARIZED: NO DOCKET #
FACIL:50-244 Robert Emmet Ginna Nuclear Plant, Unit 1, Rochester G 05000244
AUTH.NAME AUTHOR AFFILIATION
MECREDY,R.C. Rochester Gas & Electric Corp.
RECIP.NAME RECIPIENT AFFILIATION
VISSING,G.S.

SUBJECT: Forwards LER 96-014 re presuure relieving capability.LER withheld.

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ROBERT C. MECREDY
Vice President
Nuclear Operations

December 5, 1996

U.S. Nuclear Regulatory Commission
Document Control Desk
Attn: Guy S. Vissing
Project Directorate I-1
Washington, D.C. 20555

Subject: LER 96-014, Pressure Relieving Capability Could be Degraded Due to Single Failure of DC Power, Which Could Prevent Mitigating the Consequences of an Accident
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Vissing:

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 nuclear power plant being ... In a condition that was outside the design basis of the plant", and 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 96-014 is hereby submitted.

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

Very truly yours,

Joseph A. Widay for
Robert C. Mecredy

xc: Mr. Guy S. Vissing (Mail Stop 14C7)
PWR Project Directorate I-1
Washington, D.C. 20555

U.S. Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406

Ginna Senior Resident Inspector

NRC FORM 366 (4-95)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 04/30/98
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)		ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT

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TITLE (4)

Pressure Relieving Capability Could be Degraded Due to Single Failure of DC Power, Which Could Prevent Mitigating the Consequences of an Accident

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
11	05	96	96 --	014 --	00	12	05	96	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	5
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)	X	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)		50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)	X	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 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 15 single-spaced typewritten lines) (16)

On November 5, 1996, at approximately 1300 EST, the plant was in Mode 5. It was discovered that loss of a single train of DC power could prevent the pressure relieving capability of the pressurizer power operated relief valves. These valves are credited with mitigating the consequences of a steam generator tube rupture event.

The plant was restricted to operation below Mode 3 until this condition was resolved. Corrective action was to modify the control cabling for the affected valves.

The underlying cause of this condition was a change in the design function without an adequate review of the plant-specific features affected by the change in function.

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 November 5, 1996, at approximately 1300 EST, the plant was in Mode 5. Unrelated to plant conditions, personnel from Nuclear Engineering Services (NES) were updating the results of the Ginna Station Probabilistic Safety Assessment (PSA) model. These engineers were quantifying accident sequences for a Steam Generator Tube Rupture (SGTR) event. The results of these sequences were being reviewed by the Nuclear Safety and Licensing (NS&L) group within NES to assure the accuracy of the model.

II. DESCRIPTION OF EVENT:

A. DATES AND APPROXIMATE TIMES OF MAJOR OCCURRENCES:

1. Completion of plant modification in 1978 for LTOP: Original design now includes SOV-8619A and SOV-8619B DC power supply configuration.

2. November, 1987: SGTR Analysis is performed utilizing a new methodology, which credits use of a PORV to depressurize the primary system. Event Date.

3. November 5, 1996, 1300 EST: Discovery date and time.

4. November 5, 1996, 1425 EST: NRC is notified of this condition per 10CFR50.72 (b) (2) (iii) (D).

5. November 9, 1996: Control and power supplies for block valves are modified.

B. EVENT:

On November 5, 1996, at approximately 1300 EST, the plant was in Mode 5. In activities unrelated to plant conditions, NS&L engineers were reviewing the results of a PSA modeling analysis of a SGTR event. It was discovered that the plant was vulnerable to a configuration which could compromise the capability to meet the SGTR accident analysis assumptions. A single direct current (DC) electrical power system failure, combined with an existing closure of a motor-operated block valve (MOV) for a pressurizer (PRZR) power operated relief valve (PORV), could render both PORV flow paths inoperable, potentially degrading the ability to mitigate a SGTR. Actual plant configuration was reviewed to confirm the model results. It was further discovered that the plant had previously operated with the block valves closed such that it was vulnerable to a single DC power system failure, which resulted in operation outside the plant design basis.

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

When the plant was originally constructed in the 1960's, the "A" electrical train of DC power was configured to supply DC control power to MOV-515, the motor-operated block valve for PORV-431C. After the Low Temperature Overpressure Protection (LTOP) system modification in 1978, which installed nitrogen tanks to provide a safety-related pneumatic source for operation of the PORVs, the "A" train also supplied DC power to the safety-related nitrogen admission solenoid valve (SOV-8619A) for the opposite train PORV, PORV-430. Similarly, the "B" train of DC power was configured to supply DC control power to MOV-516 (the block valve for PORV-430) and to supply DC power to SOV-8619B for the opposite train PORV-431C. See the attached sketch of the PORV configuration for clarification.

If block valve MOV-515 were to be closed as allowed by Ginna Station Improved Technical Specifications (ITS) Limiting Condition for Operation (LCO) 3.4.11, and a subsequent postulated failure of the "A" train of DC power were to occur, a condition could exist where there is no available flow path for either PORV. That is, failure of the "A" train of DC power would cause SOV-8619A to fail in the vented position, preventing PORV-430 from opening with nitrogen pressure. With MOV-515 closed, no control power is available to re-open MOV-515, such that use of PORV-431C for pressure relief would be blocked. Similarly, if MOV-516 were initially closed, a postulated failure of the "B" train of DC power would prevent the ability to re-open MOV-516, thus blocking flow through PORV-430, while SOV-8619B would remain vented, preventing PORV-431C from opening with nitrogen pressure.

Thus, a single failure of either DC electrical train could prevent the ability to relieve pressure through both trains of PORVs with one or both PORV block valves initially closed.

The Ginna Station Updated Final Safety Analysis Report (UFSAR) Chapter 15 accident analysis for a SGTR includes an analysis to demonstrate acceptable offsite radiation doses. The SGTR analysis assumes that the most limiting single failure is a failure of the atmospheric relief valve (ARV) on a steam generator. This determination is based on a generic analysis which assumes that there is no single failure which could disable two trains of equipment relied upon to mitigate the consequences of an accident. However, a single failure mechanism which prevents both PORVs from relieving pressure was not analyzed, and may be more limiting since it would prevent using the PORVs for depressurization of the primary system. Since use of the PORVs is an assumed available path for depressurization after a SGTR, this condition is contrary to the SGTR mitigation strategy. Thus, for the specified configuration, the analysis is not bounding, which compromised the capability to meet the single failure criterion for a SGTR.

NS&L engineers notified Operations management of this condition. Operations management directed that the plant be restricted to operation below Mode 3 (the mode of applicability of ITS LCO 3.4.11) pending resolution of this condition. After resolution of this condition, this mode restriction was lifted.

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 condition was self-identified by an NS&L engineer while reviewing PSA modeling results of SGTR accident sequences.

F. OPERATOR ACTION:

NS&L notified the Control Room operators of this condition, and also notified Operations management. Operations management notified higher supervision and the NRC. Pending resolution of this condition, the plant was restricted to operation below Mode 3.

Operations management subsequently notified the NRC per 10 CFR 50.72 (b) (2) (iii) (D), non-emergency four hour notification, at approximately 1425 EST on November 5, 1996.

G. SAFETY SYSTEM RESPONSES:

None

III. CAUSE OF EVENT:

A. IMMEDIATE CAUSE:

The immediate cause of this condition was a potential plant configuration which was not bounded by the assumptions of the SGTR accident analysis.

B. ROOT CAUSE:

In the original plant design, the PRZR PORVs were not specifically credited with any accident mitigation function. Instead, the PORVs were primarily installed to provide the capability for a controlled depressurization of the primary system and to prevent challenges to the PRZR code safety valves. The only accident scenario which required depressurization of the primary system was a SGTR event. However, the original accident analysis never specified which equipment would be used to perform the depressurization. Instead, it was recognized that the PORVs, PRZR spray, and auxiliary PRZR spray could be used for this purpose if required.

During the 1970's, the potential for overpressurization events while shutdown were raised. Rochester Gas and Electric (RG&E) responded to this concern by installing the LTOP system in 1978, which utilized the PORVs to provide necessary relief capability. The LTOP system added a safety-related source of motive nitrogen in order to open the PORVs. The control power logic for LTOP was designed to provide redundancy for isolation of the PORVs during operation in Modes 1, 2 and 3. That is, in order to prevent the potential for a loss of coolant accident (LOCA) via a PORV flow path, the PORV nitrogen supply and the associated block valve control power were provided from opposite DC trains. In this manner, no single failure would prevent closure of a PORV flow path, although loss of control power to the nitrogen solenoid valve would cause the PORV to close.

In the 1980's, RG&E joined efforts with other Westinghouse utilities to develop a standardized generic SGTR analysis. The generic methodology which was developed (WCAP-10698) credits the use of a PORV to depressurize the primary system in order to equalize pressure between the primary and secondary systems. Therefore, after the new SGTR Analysis was applied to Ginna Station, the PORVs now had a specific requirement to open for a SGTR event (versus the capability to be closed to isolate a PORV LOCA). The application of this methodology to Ginna Station (WCAP-11668, November, 1987) was approved by the NRC per Improved Technical Specifications Amendment No. 61, in February, 1996.

However, the fact that the PORVs were vulnerable to a single failure was not identified for Ginna Station. Consequently, Ginna Technical Specifications allowed a PORV block valve to be closed indefinitely, creating the potential for this vulnerability. Ginna Station emergency operating procedures (EOPs) were also changed, during the 1980's, to utilize the PORVs during a SGTR as the preferred depressurization path.

Therefore, the underlying cause of this condition was a change in the design function for the PORVs since their original installation, without an adequate review of the plant-specific features affected by the change in function. The Causal Factor that contributed to this condition was Design Configuration and Analysis (unanticipated interaction of system or components).

This condition does not meet the NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants", definition of a "Maintenance Preventable Functional Failure".

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 nuclear power plant being ... In a condition that was outside the design basis of the plant", and 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 potential degradation of pressure relieving capability with a PORV block valve previously closed, due to a single failure, resulted in operation of the plant outside the design basis, since the ability to mitigate the consequences of a SGTR could be degraded.

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

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 this event because:

1. This accident scenario only results from a specific combination of events. The probability of these events occurring simultaneously is very low.

   a. PORV isolated by its block valve

   b. SGTR event occurs

   c. Coincident loss of the DC train that provided control power to the closed PORV block valve

2. The configuration of DC electrical power would only cause operational or safety implications during a SGTR. Since there was no such event while a block valve was closed during previous operating cycles, use of the PORVs to depressurize the primary system was never required.

3. There are two independent methods of depressurizing the PRZR: normal PRZR spray and auxiliary PRZR spray. Although not assumed in the accident analysis, PRZR spray provides an alternative means to depressurize the primary system, as discussed in Chapter 15.6 of the UFSAR. EOPs direct the operator to use PRZR spray in the event that a PORV is unavailable. Therefore, had a SGTR event occurred, operators would have had means to depressurize the primary system.

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:

1. The plant was restricted to operation below Mode 3 until this condition was resolved.

2. The DC electrical power and control cabling for MOV-515 and MOV-516 were modified (swapped). The train "A" DC power is now associated with MOV-516, and train "B" DC power is now associated with MOV-515. This modification ensures that the PORV LTOP nitrogen control trains are powered consistent with their associated PORV and PORV block valve to ensure DC power train redundancy in all required accident scenarios. This modification was completed on November 9, 1996.

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TEXT CONTINUATION

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TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

B. ACTION TAKEN OR PLANNED TO PREVENT RECURRENCE:

1. The industry was notified of this condition via Nuclear NETWORK.

2. Westinghouse was notified of this concern by teleconference with NS&L personnel on November 8, 1996.

3. RG&E has completed an extensive engineering and safety evaluation process upgrade program. The upgraded process has resulted in a more in-depth and inclusive review and approval process for plant changes and modifications. This is intended to ensure that design functions meet design requirements.

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 Nuclear Power Plant could be identified.

C. SPECIAL COMMENTS:

None

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

PORV DC Power Trains

Before and after modifications

A = A Train

B = B Train