CATEGORY 1

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ACCESSION NBR:9701300185 DOC.DATE: 97/01/22 NOTARIZED: NO DOCKET #
FACIL:50-244 Robert Emmet Ginna Nuclear Plant, Unit 1, Rochester G 05000244
AUTH.NAME AUTHOR AFFILIATION
ST MARTIN,J.T. Rochester Gas & Electric Corp.
MECREDY,R.C. Rochester Gas & Electric Corp.
RECIP.NAME RECIPIENT AFFILIATION

VISSING,G.S.

SUBJECT: LER 96-015-00:on 961223,discovered thermally induced overpressure transient could occur.Caused by thermal expansion of fluid during design basis accident condition. Installed relief valve on affected line.W/970122 ltr.

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ROBERT C. MECREDY
Vice President
Nuclear Operations

January 22, 1997

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

Subject: LER 96-015, Based on Review of NRC Generic Letter 96-06, a Thermally Induced Overpressure Transient Could Occur
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) (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 96-015 is hereby submitted.

Very truly yours,

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

JE22
1/1

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

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

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

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TITLE (4)

Based on Review of NRC Generic Letter 96-06, a Thermally Induced Overpressure Transient Could Occur

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
12	23	96	96	-- 015 --	00	01	22	97	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	1	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)
POWER LEVEL (10)	100		20.2203(a)(1)		20.2203(a)(3)(i)		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 December 23, 1996, at approximately 1032 EST, the plant was in Mode 1 at approximately 100% steady state reactor power. During a review of piping systems, as requested by NRC Generic Letter 96-06, it was discovered that a piping system was susceptible to overpressurization due to thermal expansion of fluid during a design basis accident condition. This piping system could potentially rupture as a result of a thermally induced overpressure transient, which could divert the delivery of some of the Containment Spray flow.

Immediate corrective action was to isolate and vent the affected line to assure there could be no pressure buildup due to thermal expansion of fluid. Long term corrective action was to install a relief valve on the affected piping. The line was subsequently returned to its normal alignment.

The underlying cause of this condition was the effects of heatup during design basis accident conditions were not considered in the original design specifications of the system.

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:

Personnel from Nuclear Engineering Services (NES) had been reviewing piping system configurations, in preparation for responding to NRC Generic Letter (GL) 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions". NES personnel had already made preliminary lists of some piping systems that required further evaluation to determine if they were susceptible to thermal expansion of fluid during design basis accident conditions. When affected systems were identified, deterministic evaluations were performed to assess the safety impact and to identify suitable corrective actions or compensatory measures.

On December 20, 1996, at approximately 0730 EST, the plant was in Mode 1 at approximately 100% steady state reactor power. In activities unrelated to plant conditions, NES personnel identified that the Containment Spray (CS) Charcoal Filter Dousing line was susceptible to overpressurization during the injection phase of a design basis accident condition. A rupture of this line could divert some of the CS flow, reducing the amount of CS flow delivered to the CS ring headers. The affected area is on a two inch diameter line between two check valves and normally closed motor-operated valves. This piping is connected to the rest of the CS system and is inside the containment (CNMT). At the time of identification, preliminary deterministic evaluations revealed that overpressurization would not result in rupture of the affected piping, due to the presence of entrapped air in the line.

As a conservative measure, NES personnel recommended temporary corrective action to isolate the affected piping from the rest of the CS system. The dousing function is not credited in any accident analysis in the Ginna Station Updated Final Safety Analysis Report (UFSAR). The affected line was isolated and vented to assure there could be no pressure buildup due to thermal expansion of fluid. This action ensured that a potential rupture could not occur. At approximately 0017 EST on December 21, 1996, this piping was isolated and vented.

NES personnel later formally calculated the stresses on the CS Charcoal Filter Dousing line, and concluded that if a thermally induced overpressure transient were to occur, there would be the potential for a rupture of this line.

II. DESCRIPTION OF EVENT:

A. DATES AND APPROXIMATE TIMES OF MAJOR OCCURRENCES:

- December 20, 1996, 0730 EST: Preliminary evaluations reveal that the CS Charcoal Filter Dousing line may be susceptible to a thermally induced overpressure transient. The results at this time are that the line would not rupture.
- December 21, 1996, 0017 EST: CS Charcoal Filter Dousing line is isolated and vented.

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

- December 23, 1996, 1032 EST: Event date and time and Discovery date and time. Completed calculations indicate that the line may have ruptured had an overpressurization occurred.

- December 23, 1996, 1228 EST: The NRC Operations Center is notified of this condition, as per 10CFR50.72 (b) (2) (iii) (D).

- January 3, 1997: CS Charcoal Filter Dousing line is modified by installation of a relief valve and line is returned to its normal alignment.

B. EVENT:

On December 23, 1996, at approximately 1032 EST, the plant was in Mode 1 at approximately 100% steady state reactor power. In activities unrelated to plant conditions, NES personnel reported to the Shift Supervisor that formal calculations showed, if the CS Charcoal Filter Dousing line had been in its normal alignment, a thermally induced overpressure transient could have occurred during a design basis accident condition. The postulated overpressurization could have resulted in stresses that could potentially rupture the line. Such a rupture could have diverted some of the CS flow, reducing the delivery of CS flow to the CS ring headers below what is assumed in the accident analysis.

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 NES engineer while reviewing piping systems for susceptibility to a thermally induced overpressure transient, as requested by NRC GL 96-06.

F. OPERATOR ACTION:

When initially notified that this piping may be susceptible to overpressurization, Control Room operators isolated and vented the affected piping.

When notified of the results of the formal calculations, the Shift Supervisor notified the Control Room operators of this condition. The Shift Supervisor also notified higher supervision and the NRC.

The Shift Supervisor subsequently notified the NRC Operations Center per 10 CFR 50.72 (b) (2) (iii) (D), non-emergency four hour notification, at approximately 1228 EST on December 23, 1996.

Following installation of the relief valve, the Control Room operators returned the affected line to its normal alignment.

G. SAFETY SYSTEM RESPONSES:

None

III. CAUSE OF EVENT:

A. IMMEDIATE CAUSE:

The immediate cause of the condition was a potential for a thermally induced overpressure transient, as confirmed by a formal calculation that showed that stresses resulting from overpressurization were large enough to potentially rupture the CS Charcoal Filter Dousing line. The postulated rupture could have diverted the delivery of some of the CS flow.

B. ROOT CAUSE:

The underlying cause of the potential overpressure condition was that the effects of heatup on isolated piping systems during design basis accident conditions was not considered in the original design specifications of the system. This condition is now recognized as an industry-wide generic concern, and not solely as a plant-specific concern. The Causal Factor that contributed to this condition was Design Configuration and Analysis:

- inadequate failure modes and effects evaluation
- unanticipated interaction of systems 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".

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IV. ANALYSIS OF EVENT:

This event is reportable 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 potential rupture of the CS Charcoal Filter Dousing line could degrade the ability to mitigate the consequences of an accident.

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

- Failure of the CS Charcoal Filter Dousing line would divert the delivery of some of the CS flow to the CS headers. The reduced spray flow rate would have reduced the amount of fission products that would be removed from the CNMT atmosphere as a result of a loss of coolant accident (LOCA) event. Thus, the radiological consequences would have increased. However, current LOCA analysis off-site dose calculations indicate greater than a factor of ten margin of safety for this scenario.

- In the unlikely event that a design basis accident condition had occurred, the following factors would have mitigated the severity of the consequences:

  a. the presence of entrapped air in the CS Charcoal Filter Dousing line (which would prevent the line from being completely water solid and provide a dampening effect on the pressure buildup during heatup of the fluid)

  b. the tendency of a system, when its design pressure is exceeded, is to leak at weak points (such as MOV packing glands and flange gaskets, valve seats, bonnets) thereby relieving excess pressure prior to rupture of the line

  c. typical pipe fracture mechanisms (resulting from overpressurization) manifest themselves at cracks prior to rupture. Leakage at cracks or packing will be much less than from a full guillotine rupture of the pipe, thereby minimizing the impact on diversion of flow. Thus, the pressure buildup would be gradually relieved, reducing the likelihood of pipe rupture.

- Prior to the interim corrective action taken on December 21, 1996, the potential existed for a reduction in the capability of the CS system to deliver the required flow to the CS ring headers in certain design basis accident conditions. This reduction in capability is the direct result of diverting a portion of the spray flow delivered to the CS ring headers, caused by a postulated leak or break in the CS Charcoal Filter Dousing line. The amount of flow diverted could vary significantly based upon several factors.

- CS flow is a contributor in the mitigation of the consequences of certain design basis accident conditions. Analyses typically assume the worst case single failure, resulting in the limiting results:

  a. A Large Break LOCA is discussed in UFSAR Section 15.6.4.2 and Table 15.6-18. The limiting analysis results from the assumption that both trains of CS are delivering a maximum flow of 1800 GPM each, and that all four CNMT recirculation fan coolers are operating. A reduction of CS flow would improve the results of the LOCA analysis with respect to the peak clad temperature and fuel integrity. Therefore, diverting a portion of the flow caused by a break in the Dousing line has no effect on this analysis.

  b. The Main Steam Line Break (MSLB), CNMT Integrity Analysis is discussed in UFSAR Section 6.2.1.2.3. This analysis considered a spectrum of secondary side break sizes and operating conditions. The results of the analyses are itemized in UFSAR Table 6.2-12. The minimum value of 1300 GPM was assumed for CS flow. While a reduction in CS flow would have had the potential to increase the peak CNMT pressure and temperature, there are less limiting radiological consequences that result from the steam line break events. On that basis, there would not have been an adverse effect on the health and safety of the public in satisfying the criteria of 10 CFR 100.

  c. The Large Break LOCA, CNMT Integrity Analysis is discussed in UFSAR Section 6.2.1.2.2. The results are less limiting than from the MSLB analysis discussed in paragraph 4.b. above with respect to peak CNMT pressure and temperature.

- Since a reduction in delivered CS flow could cause an increase in peak CNMT pressure, the potential reduction in CS flow was examined using the hydraulic flow model of the Ginna Emergency Core Cooling System (ECCS) using the computer program "KYPIPE". The hydraulic model showed that a reduction in delivered spray flow would occur. The effect of the reduction in flow would be more limiting if a single failure is assumed for the "B" train. A computer code run that was performed by Rochester Gas and Electric Corporation, utilizing this limiting reduced flow for the "A" train, showed no impact on peak CNMT pressure for a LOCA. (Even when a postulated complete severance is assumed at a piping location which would produce the highest break flow, the CS flow for the "B" train nearly meets the assumed flow used in the accident analysis.)

- UFSAR Section 15.6.4.3 discusses the environmental consequences of the LOCA and the results are summarized in UFSAR Table 15.6-23. The value of 27.3 REM Thyroid and 1.75 REM Whole Body doses are well within the 10 CFR 100 limits of 300 REM and 25 REM, respectively.

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- As discussed above, the predicted CS flow rate is shown to vary significantly depending on the break location and the particular ECCS train that is assumed to fail. Should both trains of ECCS function as designed, even with the complete severance of the 2 inch CS Charcoal Filter Dousing line, the CS flow would exceed the accident analysis value of 1300 GPM. Therefore, there would be no effect on the accident analysis.

V. CORRECTIVE ACTION:

A. ACTION TAKEN TO RETURN AFFECTED SYSTEMS TO PRE-EVENT NORMAL STATUS:

- The affected line was isolated and vented at approximately 0017 EST on December 21, 1996.
- A relief valve was installed on the affected line on January 3, 1997.
- The affected line was returned to its normal alignment after completion of the modification.

B. ACTION TAKEN OR PLANNED TO PREVENT RECURRENCE:

- Action will be taken in accordance with the requested actions of NRC GL 96-06 to identify and correct affected piping systems. The RG&E reply to GL 96-06 will provide this information.
- UFSAR Section 6.5.1 will be updated to reflect the changes made to the configuration of the CS Charcoal Filter Dousing line, and include pertinent information relative to GL 96-06 in the next scheduled UFSAR update.

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