CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9906290303 DOC.DATE: 99/06/21 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 99-001-01:on 990222,deficiencies in NSSS vendor steam-line brake mass & energy release analysis results in plant being outside design bases occurred.Caused by deficiencies in W.Temporary administrative replaced.With 990621 ltr.

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ROBERT C. MECREDY
Vice President
Nuclear Operations

June 21, 1999

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

Subject: LER 1999-001, Revision 1, Deficiencies in NSSS Vendor Steamline Break Mass and Energy Release Analysis Results in Plant Being Outside its Design Basis
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Vissing:

The attached Licensee Event Report LER 1999-001, Revision 1, is submitted 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 the nuclear power plant being ... In a condition that was outside the design basis of the plant", and in accordance with 10 CFR 21, "Reporting of Defects and Noncompliances".

Very truly yours,

Robert C. Mecredy

Robert C. Mecredy

xc: Mr. Guy S. Vissing (Mail Stop 8C2)
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
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Regional Administrator, Region I
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U.S. NRC Ginna Senior Resident Inspector

IE22

NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001

LICENSEE EVENT REPORT (LER)

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

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TITLE (4)

Deficiencies in NSSS Vendor Steamline Break Mass and Energy Release Analysis Results in Plant Being Outside its Design Basis

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
										05000
02	22	1999	1999	– 001 –	01	06	21	1999	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9): 1

POWER LEVEL (10): 75

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)(B)		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)	X	OTHER Part 21
	20.2203(a)(2)(iii)		50.36(c)(1)		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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

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

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On February 22, 1999, Ginna Station was notified by its NSSS vendor (Westinghouse) of two modeling errors in the analysis for a main steamline break inside containment with an assumed single failure of a main feedwater regulating valve. These errors had a nonconservative impact on calculated peak containment pressure.

The first error involved the omission of a previously unaccounted for volume of high temperature feedwater. The second error resulted in an incorrect feedwater isolation time.

The consequences of these two modeling errors previously created the potential for containment to exceed its design basis value of 60 psig for a main steamline break with a concurrent failure of a main feedwater regulating valve for R.E. Ginna Cycles 12 through 25.

The cause of the plant being in a condition that was outside the design basis of the plant from Cycle 12 to Cycle 25 was deficiencies in the Westinghouse main steamline break mass and energy release analysis.

Immediate corrective action for Cycle 27 was to place a temporary administrative restriction of 40 degrees Fahrenheit maximum on screenhouse bay temperature, a measure of Service Water / Containment fan cooler heat removal and pressure reduction capability, for Modes 1 through 4 (when containment is required to be operable) to ensure adequate post-accident environmental conditions margin existed.

Corrective action to justify return to power for Cycle 28 included a cycle-specific analysis showing peak containment pressure less than 60 psig. 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:

In 1992, the NSSS vendor for Ginna Station (Westinghouse) analyzed a full spectrum of cases for the main steam line break (MSLB) and containment response for the R.E. Ginna plant. The set of analyses was performed to support the program to reduce the boric acid storage tank (BAST) boron concentration to 2000 ppm. This analysis entailed over 120 cases, which varied break size and location, initial power level, single failure, offsite power availability, and different reactivity coefficients (rodded versus unrodded). In 1995, Westinghouse re-visited the steamline break mass and energy release event due to replacement steam generators (RSGs) and an increase to 18 month fuel cycles. Many of the previous cases could be eliminated due to:

1. An integral flow restrictor on the RSGs, eliminating any breaks larger than 1.4 square feet (the RSGs have flow restricting outlet nozzles which make any break size larger than the throat area of the restrictor not credible and thus, break flow areas greater than 1.4 square feet no longer need to be considered), and

2. Credit being taken for the main steam non-return check valve, which eliminated several types of breaks and the need to consider a main steam isolation valve (MSIV) single failure for containment overpressurization cases.

After eliminating cases that were no longer credible, cases for the 1995 analyses were selected from the 1992 BAST reduction program. Six of the most limiting cases were selected, including different break sizes, initial power levels, single failures, and offsite power assumptions. Another difference from the previous analyses is explained in the Ginna Station Updated Final Safety Analysis Report (UFSAR) Section 6.2.1.2.3:

> "In the Reference 3 analyses [1992 BAST reduction program], the most limiting feedwater control system failure was assumed to be a failure in the auxiliary feedwater system control. This failure resulted in increased auxiliary feedwater flow rates for the entire duration of the transient. A failure of the isolation function of the main feedwater regulating valves (MFRVs) was not considered because a feedwater isolation signal also results in a main feedwater pump trip and in closure of the pump discharge valve (MFPDV). Thus, no single failure would have resulted in a failure to isolate main feedwater or a significant delay in the isolation of main feedwater to the affected steam generator. It has subsequently been realized that the condensate pumps would continue to pump through the main feedwater pumps at low steam generator pressures and that the pump discharge valve (MFPDV) closure time is approximately 80 seconds. Therefore, a failure of the isolation function of the main feedwater regulating valve (MFRV) could result in main feedwater flow via the condensate pumps for approximately 80 seconds to the affected steam generator. This new failure scenario is a more limiting failure than was assumed in the previous analyses due to the timing and magnitude of the increased flow."

The requirement for the 80 second stroke time requirement of the MFPDV is addressed in the Ginna Station Improved Technical Specifications (ITS) Section 3.7.3.

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II. DESCRIPTION OF EVENT:

A. DATES AND APPROXIMATE TIMES OF MAJOR OCCURRENCES:

- February 22, 1999: Event date.
- February 22, 1999, 1540 EST: Discovery date and time.
- February 22, 1999, 1601 EST: NRC is notified of this condition per 10CFR50.72 (b) (1) (ii) (B).
- March 1, 1999, 2018 EST: Plant is shutdown in Mode 4 for the planned 1999 refueling outage.
- April 16, 1999: NSSS analysis shows MSLB containment pressure is less than 60 psig.

B. EVENT:

On February 22, 1999, in activities unrelated to plant conditions, Ginna Station was notified by its NSSS vendor (Westinghouse) of two modeling errors in the analysis for a MSLB inside containment with an assumed single failure of a MFRV. (Refer to the letter RGE-99-207 from Mr. Steve M. Ira of the Westinghouse Electric Corporation to Mr. Peter Bamford, Rochester Gas and Electric Corporation (RG&E), dated February 22, 1999, Subject: Steamline Break Mass and Energy Release Analysis Nonconformance.) These errors had a nonconservative impact on calculated peak containment pressure.

The first error involves the volume of feedwater that exists between the specified isolation valve and the faulted steam generator. When the MFRV is assumed to fail to perform its isolation function, then the isolation function is performed by the MFPDV. Because of the relative locations of these valves to the steam generator, the volume of water is greater between the MFPDV and the steam generator than between the MFRV and the steam generator. This increased volume of high temperature feedwater was not accounted for in the analysis.

The second error was inadvertently isolating feedwater flow at 15 seconds instead of at 85 seconds (80 seconds valve stroke time and 5 seconds for signal delay) as was intended. This meant that feedwater flow should have continued for 70 seconds longer than was assumed in the analysis.

The consequences of these two modeling errors has the potential for containment to exceed its design basis value of 60 psig for a MSLB with an assumed concurrent failure of a MFRV. This is due to adding higher energy fluid to containment for 70 seconds longer than was assumed in the 1995 MSLB analysis of record. There was also the potential for the containment to exceed its design basis value as documented in the 1992 analyses, due to discovery of a more limiting single failure.

At the time of discovery and until the planned shutdown on March 1, 1999, the screenhouse bay temperature was maintained at approximately 39 degrees Fahrenheit, while lake temperature remained at approximately 35 degrees Fahrenheit. Ginna Station placed a temporary administrative restriction of 40 degrees Fahrenheit maximum on screenhouse bay temperature, a measure of Service Water System temperature, for Modes 1 through 4 (when containment is required to be operable) to ensure adequate margin existed. By maintaining this temperature less than 40 degrees Fahrenheit, increased containment fan cooler heat removal ability was ensured and this, combined with a high cycle-specific excess shutdown margin, was sufficient to ensure containment pressure would remain less than 60 psig for this postulated scenario.

To support operation in Cycle 28 and beyond, the MSLB analysis (with the assumed MFRV failure) was reanalyzed. Significant assumption changes to support operation were as follows:

- o Assumed shutdown margin of 2.40% versus the base case reference cycle value of 1.80%. The Ginna Station Core Operating Limits Report (COLR) was subsequently revised to incorporate this value.
- o Assumed operation at the normal Ginna Station average operating temperature (Tavg) of 561 degrees F versus the upper end of the analyzed Tavg window of 573.5 degrees F. This operating restriction has been placed in Ginna Station operating procedures and corresponds to the full power rod control program setpoint.

These changes, when combined with several minor modeling refinements, led to peak containment pressure under this scenario to be less than 60 psig. Accordingly, operations for all of Cycle 28 was justified. This includes the removal of the lake temperature restriction previously imposed.

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 event was disclosed to RG&E by the notification from its NSSS vendor (Westinghouse) of two modeling errors in the analysis for a MSLB inside containment with an assumed single failure of a MFRV. (Refer to letter RGE-99-207 from Mr. Steve M. Ira of the Westinghouse Electric Corporation to Mr. Peter Bamford, Rochester Gas and Electric Corporation (RG&E), dated February 22, 1999, Subject: Steamline Break Mass and Energy Release Analysis Nonconformance.)

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F. OPERATOR ACTION:

After review of this Westinghouse correspondence, Reactor Engineering and Analysis personnel notified Operations supervision, who notified the Control Room operators. The NRC Resident was notified at this time. At approximately 1540 EST on February 22, 1999, plant staff determined that a non-emergency one hour notification, per 10CFR50.72 (b) (1) (ii) (B), should be made to the NRC Operations Center. The Shift Supervisor made this notification at approximately 1601 EST on February 22, 1999.

Operators maintained a restriction of 40 degrees Fahrenheit maximum on screenhouse bay temperature from the time of discovery until plant shutdown on March 1 to ensure adequate margin existed.

Analyses for Cycle 28 have been completed. These analyses demonstrate that there is adequate containment (CNMT) pressure margin for a MSLB for Cycle 28. The restriction previously imposed on lake temperature was lifted.

G. SAFETY SYSTEM RESPONSES:

None

III. CAUSE OF EVENT:

A. IMMEDIATE CAUSE:

The immediate cause of the plant being in a condition that was outside the design basis of the plant was deficiencies in the Westinghouse MSLB mass and energy release analysis.

B. INTERMEDIATE CAUSE:

The intermediate cause of the deficiencies in the Westinghouse analysis was the analysis did not account for a greater volume of high temperature feedwater and used an incorrect feedwater isolation time. The additional mass and energy release from the MSLB are penalties in the containment pressure calculation.

C. ROOT CAUSE:

The underlying cause of the error in the volume of feedwater assumption was insufficient knowledge by Westinghouse personnel of the specific configuration of the Ginna Station feedwater piping.

The underlying cause of the error in the assumed feedwater isolation time was human error in the use of the LOFTRAN code by Westinghouse personnel.

IV. ANALYSIS OF EVENT:

This event is reportable in accordance with 10 CFR 21 and 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". The modeling errors in the Westinghouse MSLB mass and energy release analysis had the potential to pressurize the Ginna Station containment beyond its design pressure due to this postulated post-accident scenario.

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 the deficiencies in the Westinghouse MSLB mass and energy release analysis because:

- The MSLB containment analysis is presented in Ginna Station UFSAR Section 6.2.1.2.3. A revised analysis was performed (in March/April 1999) by offsetting the adverse effects of the additional mass and energy release through refinements in the existing analysis assumptions. These improvements do not alter the total quantity of steam released to containment, but provide benefits in lowering the rate of the release and increasing the rate of containment heat removal. Applying these analysis improvements to the MFRV failure cases yields peak containment pressures that are less than the containment design pressure.

- The Westinghouse analysis, as of March 24, 1999, showed that the peak containment pressure design basis value of 60 psig would not be exceeded provided that service water (SW) temperature (SW supplies the containment recirculation fans) is limited to less than 45 degrees Fahrenheit. Prior to entering Mode 4 for the planned 1999 refueling outage, the screenhouse bay temperature was being maintained at approximately 39 degrees Fahrenheit, while lake temperature was approximately 35 degrees Fahrenheit. Thus, plant operation remained within its design basis since the time of discovery.

- The current Westinghouse analysis, completed during the 1999 refueling outage, shows that the peak containment pressure design basis value of 60 psig was not exceeded for Cycle 27, nor will it be exceeded for Cycle 28 (the current operating cycle). Operation during future cycles are also justified, with restrictions similar to those imposed for Cycle 28.

- Any small increase in containment pressure beyond the 60 psig design basis value would not be expected to fail the containment. In 1996, the containment was successfully tested to 72 psig during the Structural Integrity Test for the steam generator replacement project.

- A steam line break does not result in severe radiological consequences at Ginna Station; thus full containment integrity is not considered required for this event (even though containment isolation is assumed to occur).

- o For Ginna Station Cycles 1 (the original cycle in 1970) through 11, the MSLB containment pressure was not part of the Ginna Station licensing basis. For Cycles 12 through 25, containment pressure on a MSLB was part of the Ginna Station licensing basis and the potential existed to exceed 60 psig under the postulated accident scenario described above. Cycles 26 and 27 operated at the current Tavg of 561 degrees F, and would have had a peak containment pressure less than 60 psig under the postulated conditions. Cycle 28 has also been shown be acceptable.

- o At no time in the operation of Ginna Station has a main steam line break occurred.

Based on the above, it can be concluded that there were no unreviewed safety questions, and 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:

- o For the operation of Ginna Station between the time of discovery and plant shutdown on March 1, 1999, the screenhouse bay temperature was maintained at approximately 39 degrees Fahrenheit, while lake temperature remained at approximately 35 degrees Fahrenheit. Thus, plant operation remained within its design basis from the discovery date until the 1999 refueling outage.

- o Ginna Station placed a temporary administrative restriction of 40 degrees Fahrenheit maximum on screenhouse bay temperature for Modes 1 through 4 (when containment is required to be operable) to ensure adequate margin existed.

- o Completion of analyses demonstrated that operation for Cycle 28 would be within design basis.

- o The restriction previously imposed on lake temperature was lifted.

B. ACTION TAKEN OR PLANNED TO PREVENT RECURRENCE:

- o Additional analyses were performed to support current and future cycle operation. The MSLB analysis (with the assumed MFRV failure) was reanalyzed. Significant assumption changes to support operation were as follows:

  a. Assumed shutdown margin of 2.40% versus the base case reference cycle value of 1.80%. The Ginna Station Core Operating Limits Report (COLR) was subsequently revised to incorporate this value.

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TEXT CONTINUATION

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b. Assumed operation at the normal Ginna Station average operating temperature (Tavg) of 561 degrees F versus the upper end of the analyzed Tavg window of 573.5 degrees F. This operating restriction has been placed in Ginna Station operating procedures and corresponds to the full power rod control program setpoint.

- Plant modifications to recover the full range of the previously analyzed Tavg window and to restore the previous shutdown margin limits will be considered in the future.
- Nuclear Engineering Services (NES) has implemented a process for independent review of vendor inputs supplied by RG&E and is in the process of performing an independent review of all accident analysis assumptions.
- Westinghouse is evaluating the need for internal corrective actions as part of their review of this condition.

VI. ADDITIONAL INFORMATION:

A. FAILED COMPONENTS:

None

B. PREVIOUS LERs ON SIMILAR EVENTS:

A similar LER 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