NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1) R.E. Ginna Nuclear Power Plant

DOCKET NUMBER (2) 05000244

TITLE (4) Pressurizer Safety Valves Lift Settings Found Above Technical Specifications Tolerance During Post-service Test, Due to Setpoint Shifts, Results in Independent Trains Being Considered Inoperable

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
02	03	95	95	--001--	00	03	06	95	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9): N

POWER LEVEL (10): 098

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (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)	X	50.73(a)(2)(vii)		OTHER
	20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		(Specify in Abstract below and in Text, NRC Form 366A)
	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 John T. St. Martin - Technical Assistant

TELEPHONE NUMBER (Include Area Code) (315) 524-4446

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
B	AB	RV	C170	Y					

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 February 3, 1995, at approximately 1824 EST, with the reactor at approximately 98% steady state power, both pressurizer safety valves, which had been previously installed and then removed for testing, were considered inoperable. Recent test results discovered that the "as-found" set pressure for the lift settings had shifted above the tolerance in the Technical Specifications.

Immediate corrective action was not required, since the valves were not installed.

The underlying cause of the setpoint shift has been attributed to a combination of factors, including long-term operation, removal and shipping to an off-site facility for testing, as well as a restrictive tolerance in the Technical Specifications. This event is NUREG-1022 Cause Code (B).

Corrective action to preclude repetition is outlined in Section V.B.

I. PRE-EVENT PLANT CONDITIONS:

The plant was at approximately 98% steady state reactor power with no major activities in progress. Two new pressurizer (PRZR) code safety valves had been purchased, and were tested in the spring of 1993 at the valve manufacturer's test facilities. The valves were shipped to Rochester Gas and Electric (RG&E) with an "as-left" set pressure of 2485 psig (+/- 1%).

The original safety valves at Ginna Station were removed during the 1993 outage for annual testing, and these two new safety valves were installed. These valves (V-434 and V-435) were then considered operable during the 1993/1994 operating cycle (cycle 23). These two valves were then removed for annual lift testing during the 1994 outage, and the original pair of safety valves (which had been tested in 1994) were installed for the 1994/1995 operating cycle (cycle 24). The removed valves were shipped to a test facility in Huntsville, Alabama, for testing, as per RG&E purchase order NQ-14349-C-JW.

The valves were tested to the requirements of RG&E Test Specification MET-049, "Pressurizer Safety Relief Valve Setpoint Testing", with steam as the test medium, on January 10, 1995 (for V-434) and January 11, 1995 (for V-435). RG&E Quality Assurance (QA) witnessed the tests. The test results showed that the "as-found" setpoints were 2525 psig (for V-434) and 2543 psig (for V-435), which exceeded the 1% lift setting tolerance of Technical Specifications. These results were recognized as nonconforming, and a Nonconformance Report (NCR 95-005) was initiated to document this condition.

On February 3, 1995, during review of NCR 95-005 by System Engineering and Nuclear Engineering Services (NES), it was determined that this represented a potentially reportable condition.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
R.E. Ginna Nuclear Power Plant	05000244	95	-- 001 --	00	3 OF 9

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

II. DESCRIPTION OF EVENT:

A. DATES AND APPROXIMATE TIMES OF MAJOR OCCURRENCES:

1. March, 1993: Newly procured PRZR safety valves are satisfactorily tested at manufacturer's test facility.

2. April, 1993: Newly procured PRZR safety valves are installed for the 1993/1994 operating cycle (cycle 23).

3. January 11, 1995: Testing of safety valves completed at off-site testing facility. Test results show that the lift pressure exceeded the lift setting tolerance. Event date and time.

4. February 3, 1995, 1824 EST: Test results are reviewed with the System Engineer. Discovery date and time.

5. February 3, 1995, 2011 EST: Shift Supervisor notifies NRC per 10 CFR 50.72.

B. EVENT:

On February 3, 1995, at approximately 1824 EST, the reactor was at approximately 98% steady state reactor power, and no major activities were in progress. NES personnel, from Mechanical Engineering and Nuclear Safety and Licensing (NS&L), were reviewing the status of NCR 95-005 with the System Engineer. Review of the NCR suggested an operability question involving these previously installed safety valves. Since both valves were previously installed during cycle 23, it was conservatively assumed that the valves had shifted out of tolerance during cycle 23.

The "as-found" setpoints were 1.6 % (for V-434) and 2.3% (for V-435) above 2485 psig. This is contrary to Ginna Technical Specification 3.1.1.3.c, which states, "Whenever the reactor is at or above an RCS temperature of 350 degrees F, both pressurizer code safety valves shall be operable with a lift setting of 2485 psig +/- 1%." A conservative decision was made to report this event under the criteria of 10 CFR 50.72 (b) (2) (iii) (D), based on input from NS&L that a +/- 1% tolerance for safety valve actuation is an assumption for several design basis events. The NRC was notified at approximately 2011 EST on February 3.

Subsequent evaluations have not been able to conclusively determine the time that the setpoint shift occurred, nor even if the shift occurred during cycle 23. Review of design basis events has confirmed that this condition does not meet the reporting criteria of 10 CFR 50.72.

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

None

D. OTHER SYSTEMS OR SECONDARY FUNCTIONS AFFECTED:

None

E. METHOD OF DISCOVERY:

RG&E QA Surveillance of the test data identified that the test results were unacceptable. This information was forwarded to NES, and NCR 95-005 was initiated. During a review of this NCR between NES and System Engineering, this condition was evaluated as potentially reportable.

F. OPERATOR ACTION:

The System Engineer notified the Shift Supervisor of the test results that affected both PRZR safety valves that were previously installed and considered operable during cycle 23, and that these results did not affect currently installed equipment. A decision was made to notify the NRC per 10 CFR 50.72 (b) (2) (iii) (D). This notification was made at approximately 2011 EST on February 3, 1995.

Since this event did not affect installed plant equipment, no other operator actions were necessary.

G. SAFETY SYSTEM RESPONSES:

None

III. CAUSE OF EVENT:

A. IMMEDIATE CAUSE:

The immediate cause for both PRZR safety valves being considered inoperable was that the "as-found" lift settings for these valves were above the setpoint tolerance of Technical Specification 3.1.1.3.c.

B. INTERMEDIATE CAUSE:

The intermediate cause for the "as-found" lift settings above the setpoint tolerance of Technical Specifications was a shift in the setpoints from the "as-left" conditions of March, 1993, to the "as-found" conditions of January, 1995.

C. ROOT CAUSE:

The underlying cause of the shift in the setpoints is attributed to a combination of factors, including attendant variables affecting the long-term operation of the valves, and the subsequent removal, decontamination, handling, and shipping of the valves to an off-site facility for testing.

The Ginna Technical Specification requirements of +/- 1% may not be appropriate with respect to the allowances for normal setpoint shifts during operation, removal, and shipping.

This event is NUREG-1022 Cause Code (B), "Design Manufacturing, Construction / Installation."

IV. ANALYSIS OF EVENT:

This event is reportable in accordance with 10 CFR 50.73, Licensee Event Report System, item (a) (2) (vii) (D), which requires a report of, "any event where a single cause or condition caused at least one independent train or channel to become inoperable in multiple systems or two independent trains or channels to become inoperable in a single system designed to ... mitigate the consequences of an accident." Both independent trains of pressure relief for the PRZR were considered inoperable due to the "as-found" lift settings above the tolerance of the Technical Specifications.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
R.E. Ginna Nuclear Power Plant	05000244	95	-- 001 --	00	7 OF 9

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

The two limiting design basis events that challenge reactor coolant system (RCS) integrity and rely on the PRZR safety valves to mitigate their consequences are the Locked Rotor transient and the Loss of Load transient. These transients were reanalyzed by Westinghouse on behalf of RG&E.

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

- The setpoints for the PRZR safety valves shifted at some unknown time between March, 1993, and January, 1995. This condition did not create a significant safety hazard for the following reasons:

  1. While the valves would have been declared inoperable (had the condition been known) based on the Technical Specification tolerance, the reanalyses of the two limiting transients shows that if the valves had lifted at the "as-found" pressure during a design basis event, they would have performed their design function with acceptable results. Thus, the acceptance criteria of all UFSAR Chapter 15 design basis events would still be satisfied.

  2. The design basis conditions bound the actual conditions that existed during cycle 23. Factors that would have made the limiting events less severe are:

     (a) Steady-state reactor power was approximately 98% during cycle 23, versus the design condition of 100%.

     (b) The "as-found" setpoints were bounded by the Westinghouse reanalysis assumptions for the setpoint tolerance.

- No event that would have required PRZR safety valve actuation occurred during cycle 23.

There were no operational or safety consequences or implications attributed to the shift in lift setpoints, because all the required RCS pressure limitations were met, using the "as-found" setpoints.

Based on the above, it can be concluded that the public's health and safety was assured at all times.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
R.E. Ginna Nuclear Power Plant	05000244	95	-- 001 --	00	8 OF 9

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

V. CORRECTIVE ACTION:

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

- The valve seats were lapped and the valves adjusted as necessary to bring the lift settings into conformance with the Technical Specifications tolerance.

- The accident analyses that are affected by a PRZR safety valve with a larger tolerance than required by Technical Specification 3.1.1.3.c have been reanalyzed by Westinghouse. The results are that the functions of the PRZR safety valves were never unacceptable with the "as-found" lift settings.

B. ACTION TAKEN OR PLANNED TO PREVENT RECURRENCE:

- Based on the results of the accident reanalysis, a revision to the Technical Specifications to provide more realistic and achievable lift setting tolerances and acceptance criteria for operability will be pursued on a priority basis, as part of the RG&E/NRC effort to implement the Improved Technical Specifications (ITS).

- To minimize the chance of a shift in the setpoint due to activities associated with on-site removal, handling, decontamination, packaging, shipping, storing, and reinstallation, the administrative controls for removal, shipping, testing, and reinstallation of these valves will be evaluated and enhanced, as appropriate, to ensure that proper controls are in place for key activities that could inadvertently affect the lift settings.

- To increase the repeatability of test results, NES will evaluate the adequacy of the test requirements of MET-049, and revise the test specification, as appropriate.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
R.E. Ginna Nuclear Power Plant	05000244	95	-- 001 --	00	9 OF 9

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

VI. ADDITIONAL INFORMATION:

A. FAILED COMPONENTS:

The failed components are Crosby Valve and Gage Co. safety valves, Model HB-BP-86E, serial numbers N69877-00-0006 and N69877-00-0007.

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