**Eric Larson**
Plant General Manager

CENG
a joint venture of
Constellation Energy
eDF

R.E. Ginna Nuclear Power Plant, LLC
1503 Lake Road
Ontario, New York 14519-9364

585.771.5205
585.771.3900 Fax

Eric.Larson@cengllc.com

December 3, 2010

U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

**ATTENTION:** Document Control Desk

**SUBJECT:** **R.E. Ginna Nuclear Power Plant**
Docket No. 50-244

**LER 2010-002 Rev. 1, Unanalyzed Condition due to Leakage of Residual Heat Removal Pump Suction Relief Valves**

The attached Licensee Event Report (LER) 2010-002 Rev. 1 is submitted under the provisions of NUREG 1022, Event Reporting Guidelines. There are no new commitments contained in this submittal. Should you have any questions regarding the information in this letter, please contact Mr. Thomas Harding at (585) 771-5219.

Very truly yours,

Eric A Larson

Attachments: (1) LER 2010-002 Rev. 1

cc: W.M. Dean, NRC
D.V. Pickett, NRC
Resident Inspector, NRC (Ginna)

IE22
NRR

WPLNRC-1002371

# ATTACHMENT 1

## LER 2010-002 Rev. 1

NRC FORM 366 (10-2010) U.S. NUCLEAR REGULATORY COMMISSION

# LICENSEE EVENT REPORT (LER)

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



| 1. FACILITY NAME | 2. DOCKET NUMBER | 3. PAGE |
|---|---|---|
| R.E. Ginna Nuclear Power Plant | 05000 244 | 1 OF 5 |

**4. TITLE**

Unanalyzed Condition due to Leakage of Residual Heat Removal Pump Suction Relief Valves

| 5. EVENT DATE | | | 6. LER NUMBER | | | 7. REPORT DATE | | | 8. OTHER FACILITIES INVOLVED | |
|---|---|---|---|---|---|---|---|---|---|---|
| MONTH | DAY | YEAR | YEAR | SEQUENTIAL NUMBER | REV NO. | MONTH | DAY | YEAR | FACILITY NAME | DOCKET NUMBER |
| | | | | | | | | | | 05000 |
| 09 | 09 | 2010 | 2010 | - 2 - | 1 | 12 | 03 | 2010 | FACILITY NAME | DOCKET NUMBER 05000 |

**9. OPERATING MODE:** 1

**10. POWER LEVEL:** 100%

**11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §:** *(Check all that apply)*

| | | | |
|---|---|---|---|
| ☐ 20.2201(b) | ☐ 20.2203(a)(3)(i) | ☐ 50.73(a)(2)(i)(C) | ☐ 50.73(a)(2)(vii) |
| ☐ 20.2201(d) | ☐ 20.2203(a)(3)(ii) | ☐ 50.73(a)(2)(ii)(A) | ☐ 50.73(a)(2)(viii)(A) |
| ☐ 20.2203(a)(1) | ☐ 20.2203(a)(4) | ☑ 50.73(a)(2)(ii)(B) | ☐ 50.73(a)(2)(viii)(B) |
| ☐ 20.2203(a)(2)(i) | ☐ 50.36(c)(1)(i)(A) | ☐ 50.73(a)(2)(iii) | ☐ 50.73(a)(2)(ix)(A) |
| ☐ 20.2203(a)(2)(ii) | ☐ 50.36(c)(1)(ii)(A) | ☐ 50.73(a)(2)(iv)(A) | ☐ 50.73(a)(2)(x) |
| ☐ 20.2203(a)(2)(iii) | ☐ 50.36(c)(2) | ☐ 50.73(a)(2)(v)(A) | ☐ 73.71(a)(4) |
| ☐ 20.2203(a)(2)(iv) | ☐ 50.46(a)(3)(ii) | ☐ 50.73(a)(2)(v)(B) | ☐ 73.71(a)(5) |
| ☐ 20.2203(a)(2)(v) | ☐ 50.73(a)(2)(i)(A) | ☐ 50.73(a)(2)(v)(C) | ☐ OTHER Specify in Abstract below or in NRC Form 366A |
| ☐ 20.2203(a)(2)(vi) | ☑ 50.73(a)(2)(i)(B) | ☐ 50.73(a)(2)(v)(D) | |

**12. LICENSEE CONTACT FOR THIS LER**

| FACILITY NAME | TELEPHONE NUMBER (Include Area Code) |
|---|---|
| Thomas Harding, Licensing Director | (585) 771-5219 |

**13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT**

| CAUSE | SYSTEM | COMPONENT | MANU-FACTURER | REPORTABLE TO EPIX | CAUSE | SYSTEM | COMPONENT | MANU-FACTURER | REPORTABLE TO EPIX |
|---|---|---|---|---|---|---|---|---|---|
| D | BP | RV | NUPRO | Y | | | | | |

| 14. SUPPLEMENTAL REPORT EXPECTED | 15. EXPECTED SUBMISSION DATE | MONTH | DAY | YEAR |
|---|---|---|---|---|
| ☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ☑ NO | | | | |

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

**On September 9, 2010, after isolation of Residual Heat Removal (RHR) pump suction relief valves for maintenance, the shift operating crew identified that the rate of level rise in the Auxiliary Building (AB) sump tank had slowed. Prior to the isolation, leakage into the AB sump tank was estimated to be approximately 6 gph. The source of the elevated leakage was attributed to the RHR pump suction relief valves. The Technical Specification program for primary coolant leakage outside of containment has a limit of 2 gph and the current site dose calculation assumes Emergency Core Cooling System leakage of 4 gph. During certain accident conditions, with the continued leakage of RHR suction relief valves, the plant would have been in an unanalyzed condition for dose. An evaluation was performed upon identification of the condition and concluded that the safety system would have been able to perform its required function and no regulatory dose limits would have been exceeded. The cause of the leakage is attributed to damage of the valve seats due to excessive cycling. Excessive cycling was caused by a surveillance test alignment which subjected these valves to pressures in excess of the lift setpoint. Corrective Actions are described in section IV.**

| 1. FACILITY NAME | 2. DOCKET | 6. LER NUMBER | | | 3. PAGE |
|---|---|---|---|---|---|
| | | YEAR | SEQUENTIAL NUMBER | REV NO. | |
| R.E. Ginna Nuclear Power Plant | 05000 244 | 2010 | - 002 - | 1 | 2 OF 5 |

NARRATIVE

## I. DESCRIPTION OF EVENT

### A. PRE-EVENT PLANT CONDITIONS:

The reactor was in Operational Mode 1 at 100% power, 2235 psig and 573 degrees F.

### B. EVENT:

A modification was implemented in January of 2009 to address a concern with the ability of the Residual Heat Removal (RHR) system suction valves (MOV-850A/B) to open when required for system alignment to Sump 'B' during the recirculation mode of accident mitigation. Specifically, redundant relief valve trains each consisting of two (2) relief valves in series were installed. RV-686G and RV-686I comprise the 'A' train and RV-686H and RV-686J comprise the 'B' train. These relief valve trains are physically connected to the RHR pumps recirculation line and discharge to a common drain line leading to the Auxiliary Building sump tank.

In March of 2009, it was identified that all four relief valves had a setpoint of 150 psig in lieu of the 150 psia identified by the modification documents. RV-686G and RV-686I were replaced with re-set valves in July of 2009 and RV-686H and RV-686J were replaced with re-set valves in March of 2010. The as-found setpoint for RV-686H was 240 psig in lieu of 150 psig as originally set (this valve was tested two more times and each time it lifted at approx 150 psig). The evaluation performed as a result of the RV-686H setpoint concern generated a work order to verify that the set point of RV-686J had not shifted.

On September 9, 2010, the isolation valve to RV-686H and RV-686J was closed in preparation for performing the set point verification of RV-686J. Following the isolation, the shift operating crew identified that the rate of level rise in the Auxiliary Building sump tank had slowed. A review of plant computer data indicated that the rate of level rise appeared to be elevated since March 2010. RV-686J was removed, and the as-found testing identified that it failed its seat leakage requirement at a pressure of 50 psig. The acceptance criteria is 122 psig. Subsequent testing of RV-686H determined that this valve failed its set point test by leaking at a pressure of 10 psig. The desired set point is 135 psig. Both valves were replaced with tested spares. These valves are designed to ensure that RHR pump suction valves can open against a differential pressure postulated for specific accident scenarios. Given that the lift setpoint was lower than design, there is reasonable assurance that these valves could still perform in an accident. However the additional flow to the auxiliary building sump tank was not accounted for in the Emergency Core Cooling System (ECCS) leakage criteria of 2 gph. The current control room and offsite dose calculation assumes a leakage of 4 gph and it is estimated that the flow rate into the sump was approximately 6 gph. If an accident condition were to occur, portions of the plant would be in an unanalyzed condition for dose.

To date, the valves have been extensively tested, disassembled and inspected. New valves have been installed and have been observed to lift during surveillance testing. Evaluation has identified an unexpected consequence of the test configuration. When the isolation valve in the pump recirculation flow line is closed, the relief valves are exposed to higher pressure than would otherwise be experienced during operational conditions. It is postulated that damage to the valve seats occured due to excessive cycling of these valves during testing. This resulted in a failure of the valves to fully reseat.

NARRATIVE

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

None

D. DATES AND APPROXIMATE TIMES OF MAJOR OCCURRENCES:

01/06/2009 Relief Valves installed
07/21/2009 RV-686G and RV-686I replaced to correct setpoint
03/17/2010 RV-686H and RV-686J replaced to correct setpoint
09/09/2010 RV-686J seat leakage and RV-686H setpoint found below acceptance criteria. Replaced with new valves and returned to service.

E. OTHER SYSTEMS OR SECONDARY FUNCTIONS AFFECTED:

None

F. METHOD OF DISCOVERY:

Planned Maintenance and Surveillance testing

G. MAJOR OPERATOR ACTION:

The valves were isolated at the time of discovery. No actions were required.

H. SAFETY SYSTEM RESPONSES:

No safety systems actuated or were required to respond to this event.

II. CAUSE OF EVENT:

This event was entered into the site corrective action program (CR-2010-005530). The failure is attributed to damage of the valve seats due to excessive cycling of these valves during testing. Excessive cycling was caused by a system configuration alignment during surveillance testing which subjected these valves to pressures in excess of the lift setpoint.

III ANALYSIS OF THE EVENT:

This event is reportable in accordance with 10 CFR50.73, Licensee Events Report System under item (a)(2)(ii)(B) based on the plant being in an unanalyzed condition that significantly degraded plant safety. The event is also considered to be reportable in accordance with item (a)(2)(i)(B) based on operation or condition prohibited by Technical Specifications.

NARRATIVE

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

The valves were isolated at the time of discovery and no immediate actions were required. Since the valves were found with setpoints lower than expected, the pressure relieving function was preserved. It appears the valves could have been leaking since replacement in March of 2010. The existing analysis uses twice the programmatic limit for ECCS leakage to calculate post-accident doses. With a Technical Specification program limit of 2 gph leakage, the analysis uses 4 gph. An evaluation was performed against 10CFR50.67(b)(2)(iii) criteria for access to and occupancy of the control room as well as offsite dose under accident conditions. It was calculated that under worst case conditions, the leakage would not have resulted in exceeding regulatory limits.

An evaluation was performed and concluded that there is reasonable assurance that there would have been no loss of safety function or significant impact on ECCS inventory as a result of this condition.

Based on the above considerations, the nuclear safety consequences of this event are very low.

This event does not have any impact on NRC performance indicators.

IV CORRECTIVE ACTIONS:

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

Relief valves RV-686H and RV-686J were replaced

B. ACTION TAKEN OR PLANNED TO PREVENT RECURRENCE

The surveillance procedure will be revised to isolate RHR relief valves during testing. Improved system monitoring will be performed to ensure early identification of valve leakage if it were to occur in the future.

NARRATIVE

## V. ADDITIONAL INFORMATION:

### A. FAILED COMPONENT

RHR relief valves RV-686J and RV-686H failed to re-close following surveillance testing. The system configuration alignment during surveillance testing caused the valves to unexpectedly open.

### B. PREVIOUS LERS ON SIMILAR EVENTS

A review of Ginna events over the past five years identified no similar events

### C. THE ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) COMPONENT FUNCTION IDENTIFIER AND SYSTEM NAME OF EACH COMPONENT OR SYSTEM REFERRED TO IN THIS LER:

| COMPONENT | IEEE 803 FUNCTION IDENTIFIER | IEEE 805 SYSTEM INDENTIFICATION |
|---|---|---|
| RV-686J | RV | BP |
| RV-686H | RV | BP |

### D. SPECIAL COMMENTS

None