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FACIL:50-244 Robert Emmet Ginna Nuclear Plant, Unit 1, Rochester G 05000244

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| MECREDY,R.C. | Rochester Gas & Electric Corp. |
| RECIP.NAME | RECIPIENT AFFILIATION |

SUBJECT: LER 92-005-00:on 920420,annual maintenance exam performed on both S/G "A" & "B." 242 tubes in "A" & 216 tubes in "B" S/G required C/A due to tube degradation.Caused by recurring IGA/SCC & PWSCC.S/G return to normal.W/920520 ltr.

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TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

NOTES:License Exp date in accordance with 10CFR2,2.109(9/19/72). 05000244

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ROBERT C. MECREDY
Vice President
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AREA CODE 716 546-2700

May 20, 1992

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Subject: LER 92-005, Steam Generator Tube Degradation Due to IGA/SCC, Causes Q.A. Manual Reportable Limits to be Reached
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

In accordance with 10 CFR 50.73, Licensee Event Report System, item (other), and the Ginna Station Quality Assurance Manual Appendix B, which requires that, "If the number of tubes in a generator falling into categories (a) or (b) below exceeds the criteria, then results of the inspection shall be considered a reportable event pursuant to 10 CFR 50.73," the attached Licensee Event Report LER 92-005 is hereby submitted.

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

Very truly yours,

Robert C. Mecredy

Robert C. Mecredy

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

Ginna USNRC Senior Resident Inspector

Cert NO P748978401

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NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

# LICENSEE EVENT REPORT (LER)



| FACILITY NAME (1) | DOCKET NUMBER (2) | PAGE (3) |
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| R.E. Ginna Nuclear Power Plant | 0 5 0 0 0 2 4 4 | 1 OF 0 9 |

TITLE (4) Steam Generator Tube Degradation Due To 1GA/SCC Causes Q.A. Manual Reportable Limits To Be Reached

| 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 NAMES | DOCKET NUMBER(S) |
| 0 4 | 2 0 | 9 2 | 9 2 | — 0 0 5 | — 0 0 | 0 5 | 2 0 | 9 2 | | 0 5 0 0 0 |
| | | | | | | | | | | 0 5 0 0 0 |

OPERATING MODE (9): N

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check one or more of the following)* (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) | 50.73(a)(2)(vii) | X OTHER *(Specify in Abstract below and in Text, NRC Form 366A)* |
| 20.405(a)(1)(iii) | 50.73(a)(2)(i) | 50.73(a)(2)(viii)(A) | |
| 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 | TELEPHONE NUMBER | |
|---|---|---|
| | AREA CODE | |
| Wesley H. Backus<br>Technical Assistant to the Operations Manager | 3 1 5 | 5 2 4 - 4 4 4 6 |

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 |
|---|---|---|---|---|---|---|---|---|---|
| X | A B | T B G | H 3 1 4 | 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 fifteen single space typewritten lines)* (16)

During the 1992 Annual Refueling and Maintenance Outage, subsequent to the Eddy Current Examination performed on both the "A" and "B" Westinghouse Series 44 Steam Generator (S/G), 242 tubes in the "A" S/G and 216 tubes in the "B" S/G required corrective action due to tube degradation. Two additional tubes in each S/G were stabilized and plugged as preventive action, due to anti-vibration bar (AVB) concerns.

The immediate cause of the event was that the "A" and "B" S/G tube degradation was in excess of the Ginna Quality Assurance Manual Reportability Limits.

The underlying cause of the tube degradation is a common S/G problem of a partially rolled tube sheet crevice with recurring intergranular attack/stress corrosion cracking (IGA/SCC) and Primary Water Stress Corrosion Cracking (PWSCC) attack on S/G tubing.

Corrective action taken was to either sleeve or plug the affected tubes with accepted industry repair methods.

## I. PRE-EVENT PLANT CONDITIONS

The plant was in the Cold/Refueling Shutdown condition for the Annual Refueling and Maintenance Outage. Reactor Coolant System (RCS) was depressurized and RCS temperature was approximately 65°F. Steam Generator (S/G) Eddy Current Inspection was in progress.

## II. DESCRIPTION OF EVENT

### A. DATES AND APPROXIMATE TIMES OF MAJOR OCCURRENCES:

- o April 20, 1992, 1200 EDST: Event Date and Time
- o April 20, 1992, 1200 EDST: Discovery Date and Time
- o April 21, 1992, 1330 EDST: Oral Notification made to the NRC Office of Nuclear Reactor Regulation (NRR).
- o April 26, 1992, 1742 EDST: Steam Generator repairs completed.
- o May 5, 1992: Follow-up written report sent to NRC Office of NRR.

### B. EVENT:

During the 1992 Annual Refueling and Maintenance Outage, an Eddy Current Examination was performed in both the "A" and "B" Westinghouse Series 44 Design Recirculating Steam Generators.

The purpose of the Eddy Current Examination was to assess any corrosion or mechanical damage that may have occurred during the cycle since the 1991 examination.

The examination was performed by personnel from Rochester Gas and Electric (RG&E) and Allen Nuclear Associates, Inc. (ANA). All personnel were trained and qualified in the Eddy Current Examination method and have been certified to a minimum of Level I for data acquisition and Level II for data analysis.

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

The initial Eddy Current Examination of the "A" and "B" Steam Generators was performed utilizing the Zetec 3-Coil Motorized Rotating Pancake Coil (MRPC) probe and the Zetec MIZ-18 Digital Data Acquisition System. The frequencies selected were 400, 300, 100, and 25 KHZ.

Additional Eddy Current Examinations of the "A" and "B" Steam Generators were performed utilizing the standard 0.740" or 0.720" O.D. Bobbin Coil Probe. The frequencies selected were 400, 200, 100 and 25 KHZ.

The Inlet or Hot Leg Examination Program Plan included the examination of 100% of each open unsleeved steam generator tube from the tube end to the top of the tube sheet with MRPC. Twenty percent of the tubes were selected and examined for their full length (20% random sample as recommended in the Electric Power Research Institute (EPRI) guidelines) with a bobbin coil. In addition, 20% of each type of sleeve was examined and the remainder (unsleeved portion) of the tube was examined full length. All Row 1 and Row 2 U-bend regions selected as part of the 20% random sample were examined with the MRPC between the #6 Tube Support Plate Hot Side (TSPH) and the #6 Tube Support Plate Cold Side (TSPC) from the Cold Leg Side.

Results of the above inspections indicated that 244 tubes in the "A" Steam Generator (i.e. 226 new repairs, plus 16 previously plugged tubes, plus 2 tubes stabilized for Anti-Vibration Bar (AVB) concerns) and 218 tubes in the "B" Steam Generator (i.e. 183 new repairs, plus 3 B&W explosive plugs, plus 30 previously plugged tubes, plus 2 tubes stabilized for AVB concerns) required action. Corrective actions were therefore taken for 242 tubes in the "A" S/G, and for 216 tubes in the "B" S/G. Preventive actions (for AVB concerns) were taken for 2 tubes in each S/G.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION



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

On April 20, 1992 at approximately 1200 EDST, with the RCS depressurized and temperature at approximately 65°F, final review of the 1992 S/G Inspection Eddy Current results was completed. Results of this review indicated that more than 10 percent of the total tubes inspected are degraded (i.e. imperfections greater than 20 percent of the nominal wall thickness) and more than one percent of the total tubes inspected are degraded (i.e. imperfections greater than the repair limit). Because of the above, the results of the inspection are considered a reportable event pursuant to 10 CFR 50.73 per Appendix B of the Ginna Station Quality Assurance Manual.

On April 21, 1992 at approximately 1330 EDST, oral notification was made to the NRC Office of NRR pursuant to Appendix B of the Ginna Station Quality Assurance Manual.

On May 5, 1992 a follow-up written report of the Steam Generators Inspection and Repairs was sent to the NRC pursuant to Appendix B of the Ginna Station Quality Assurance Manual.

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

None.

**D. OTHER SYSTEMS OR SECONDARY FUNCTIONS AFFECTED:**

None.

**E. METHOD OF DISCOVERY:**

The event was apparent after the final review of the "A" and "B" Steam Generators Eddy Current Examination results.

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

F. OPERATOR ACTION:

Control Room operators completed the notifications and evaluations required by the A-25.1 (Ginna Station Event Report), submitted for the event by the S/G supervision.

G. SAFETY SYSTEM RESPONSES:

None.

III. CAUSE OF EVENT

A. IMMEDIATE CAUSE:

The immediate cause of the event was that the "A" and "B" Steam Generator tube degradation was in excess of the Ginna Station Quality Assurance Manual Reportable Limits.

B. ROOT CAUSE:

The results of the examination indicate that Intergranular Attack (IGA) and Intergranular Stress Corrosion Cracking (IGSCC) continue to be active within the tubesheet crevice region on the inlet side of each steam generator. As in the past, IGA/SCC is much more prevalent in the "B" Steam Generator with 118 new crevice indications reported. In the "A" Steam Generator, 34 new crevice indications were reported.

The majority of the inlet tubesheet crevice corrosion indications are IGA/SCC of the Mill Annealed Inconel 600 tube material. This form of corrosion is believed to be the result of an alkaline environment forming in the tubesheet crevices. This environment has developed over the years as deposits and active species such as sodium and phosphate, have reacted, changing a neutral or inhibited crevice into the aggressive environment that presently exists.

Along with IGA/SCC in the crevices, there appears to have been a slight increase in Primary Water Stress Corrosion Cracking (PWSCC) at the roll transition during the last operating cycle. This mechanism was first addressed in 1989 and this year there were 63 roll transition PWSCC indications in "B" Steam Generator and 189 roll transition (PWSCC) indications in "A" Steam Generator. These numbers include tubes that may have PWSCC in combination with IGA and SCC in the crevice.

## IV. ANALYSIS OF EVENT

This event is reportable in accordance with 10 CFR 50.73, Licensee Event Report item (other) and the Ginna Station Quality Assurance Manual Appendix B which requires that, "If the number of tubes in a generator falling into categories (a) or (b) exceeds the criteria, then results of the inspection shall be considered a reportable event pursuant to 10 CFR 50.73." The tube degradation in the "A" and "B" Steam Generators exceeded the criterion of (a) which states, "more than 10 percent of the total tubes inspected are degraded (imperfections greater than 20 percent of the nominal wall thickness)", and the criterion of (b) which states, "more than 1 percent of the total tubes inspected are degraded (imperfections greater than the repair limit)". This repair limit is defined as, "Steam Generator tubes that have imperfections greater than 40 percent through wall, as indicated by Eddy Current, shall be repaired by plugging or sleeving."

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

There were no safety consequences or implications resulting from the Steam Generator tube degradation in excess of the Quality Assurance Manual Reportable Limits because:.

o The degraded tubes were identified and repaired prior to any significant leakage or Steam Generator tube rupture occurring.

o Even assuming a complete severance of a Steam Generator tube at full power, as stated in the R.E. Ginna Nuclear Power Plant Updated Final Safety Analysis Report (Ginna/UFSAR) section 15.6.3, (Steam Generator Tube Rupture), the sequence of recovery actions ensures early termination of primary to secondary leakage with or without offsite power available thus limiting offsite radiation doses to within the guidelines of 10 CFR 100.

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:

o Of the 244 tubes in the "A" Steam Generator, 36 tubes were repaired using a Combustion Engineering 27" welded sleeve in the Hot Leg, plus 194 tubes were repaired using a Babcock and Wilcox explosively welded tubesheet sleeve in the Hot Leg and all of the above tubes will remain in service. The remaining 14 tubes were removed from service by plugging both the Hot and Cold Leg tube ends. A total of 185 tubes in the "A" Steam Generator are currently plugged and 555 tubes are sleeved.

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

- o Of the 218 tubes in the "B" Steam Generator, 186 tubes were repaired using a Combustion Engineering 27" welded sleeve in the Hot Leg, plus 9 tubes were repaired using a Combustion Engineering 30" welded sleeve in the Hot Leg and all of the above tubes will remain in service. The remaining 23 tubes were removed from service by plugging both the Hot and Cold Leg tube ends. A total of 313 tubes in the "B" Steam Generator are currently plugged and 1134 tubes are sleeved.

All the above repairs on the "A" and "B" Steam Generators were completed on April 26, 1992 at approximately 1742 EDST.

B. ACTION TAKEN OR PLANNED TO PREVENT RECURRENCE:

The occurrence/presence of IGA, SCC, and PWSCC is a common PWR Steam Generator problem. Utilities with susceptible tubing and partially rolled crevices must deal with this recurring attack on Steam Generator tubing.

R.E. Ginna Nuclear Power Plant will continue careful monitoring of both primary RCS and secondary side water chemistry parameters. These water chemistry parameters will continue to be evaluated against accepted industry guidelines in order to minimize harmful primary and/or secondary side environments.

Degraded Steam Generator tubes shall be sleeved or plugged in accordance with the Inservice Inspection Program and Accepted Industry Repair methods.

VI. <u>ADDITIONAL INFORMATION</u>

A. FAILED COMPONENTS:

The degraded components are: Inconel Grade 600 tubes having an outside diameter of 0.875 inches and a nominal wall thickness of 0.050 inches. These tubes were manufactured by Huntington Alloy Company.

B. **PREVIOUS LERs ON SIMILAR EVENTS:**

A similar LER event historical search was conducted with the following results: The crevice indications are similar to those reported in AO-74-02, AO-75-07, RO-75-013, and LERs 76-008, 77-008, 78-003, 79-006, 79-022, 80-003, 81-009, 82-003, 82-022, 83-013, 89-001, 90-004, and 91-005.

C. **SPECIAL COMMENTS:**

For a more indepth report, refer to the "1992 Steam Generator Eddy Current Inspection" Summary Examination Report sent to the NRC May 5, 1992.