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FACIL:50-244 Robert Emmet Ginna Nuclear Plant, Unit 1, Rochester G 05000244
AUTH.NAME AUTHOR AFFILIATION
BACKUS,W.H. Rochester Gas & Electric Corp.
MECREDY,R.C. Rochester Gas & Electric Corp.
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 93-002-00:on 930404,during 1993 SG eddy current exam, d determined 1% of total tubes in SG A & B degraded.Caused by IGA & IGSCC within tube sheet crevice region.Tubes welded using tube sheet sleeve.W/930504 ltr.

DISTRIBUTION CODE: IE22T COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 10
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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EXTERNAL:	EG&G BRYCE,J.H	2	2	L ST LOBBY WARD	1	1
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ROBERT C. MECREDY
Vice President
Ginna Nuclear Production

TELEPHONE
AREA CODE 716 546-2700

May 4, 1993

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

Subject: LER 93-002, Steam Generator Tube Degradation Due To IGA/SCC, Causes Quality Assurance 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 93-002 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 # P237 846 831

IE22 1/1

NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WTH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4) Steam Generator Tube Degradation Due To IGA/SCC, Causes Quality Assurance 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	0 4	9 3	9 3	— 0 0 2 —	0 0	0 5	0 4	9 3		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 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 H	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

YES *(If yes, complete EXPECTED SUBMISSION DATE)* X NO

EXPECTED SUBMISSION DATE (15): MONTH DAY YEAR

ABSTRACT *(Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines)* (16)

During the 1993 Annual Refueling and Maintenance Outage, subsequent to the eddy current examination performed on both the "A" and "B" Westinghouse Series 44 Steam Generators, 122 tubes in the "A" steam generator and 171 tubes in the "B" steam generator required corrective action due to tube degradation.

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

The underlying cause of the tube degradation is a common steam generator 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 steam generator tubing. (This event is NUREG-1022 (X) cause code)

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

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

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 64°F. Steam Generator (S/G) eddy current examination was in progress.

II. DESCRIPTION OF EVENT

A. DATES AND APPROXIMATE TIMES OF MAJOR OCCURRENCES:

- o April 4, 1993, 1800 EDST: Event date and time.
- o April 4, 1993, 1800 EDST: Discovery date and time.
- o April 6, 1993, 1300 EDST: Oral notification made to the NRC Office of Nuclear Reactor Regulation (NRR).
- o April 7, 1993, 2128 EDST: Steam Generator repairs completed.
- o April 19, 1993: A Special Report was sent to the USNRC.

B. EVENT:

During the 1993 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 1992 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.

The initial eddy current examination of the "A" and "B" steam generators was performed utilizing a standard bobbin coil technique with data acquisition being performed with the EDDYNET Acquisition System. The frequencies selected were 400, 200, 100, and 25 KHz.

Additional eddy current examinations of the "A" and "B" steam generators were performed utilizing the Zetec 3-coil Motorized Rotating Pancake Coil (MRPC) probe to examine the roll transition region, selected crevices and support plates. The frequencies used for these examinations were 400, 300, 100, and 25 KHz.

The inlet or hot leg examination program plan was generated to provide the examination of 100% of each open unsleeved steam generator tube from the tube end through the first tube support plate, along with 20% of these tubes being selected and examined for their full length (20% random sample as recommended in the Electric Power Research Institute (EPRI) guidelines) with the bobbin coil. In addition, 20% of each type of sleeve was examined and the remaining tube examined full length. All Row 1 and Row 2 U-Bend regions were examined with the Motorized Rotating Pancake Coil (MRPC) between the #6 tube support plate hot side and the #6 tube support plate cold side from the cold leg side.

Results of the above examinations indicated that 122 tubes in the "A" steam generator required action (i.e. 121 tubes that were found to have "new" tubesheet crevice indications, and one tube that was obstructed by a foreign object.) 171 tubes in the "B" steam generator required action (i.e. 123 new repairs, plus 48 previously plugged tubes.) Corrective actions were therefore taken for 122 tubes in the "A" steam generator, and for 171 tubes in the "B" steam generator.

On April 4, 1993 at approximately 1800 EDST, with the RCS depressurized and temperature at approximately 64°F, final review of the 1993 Steam Generator eddy current examination results was completed. Results of this review indicated that 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 6, 1993, at approximately 1300 EDST oral notification was made to the NRC Office of NRR pursuant to Appendix "B" of the Ginna Station Quality Assurance Manual.

On April 19, 1993, a Special Report listing the number of tubes required to be plugged or sleeved in each Steam Generator, was reported 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 generator eddy current examination results.

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 Steam Generator examination and repair 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 103 new crevice indications reported in 1993. In the "A" steam generator, 41 new crevice indications were reported in 1993.

In 1992, 118 new crevice indications were reported in the "B" steam generator, and 34 new crevice indications were reported in the "A" steam generator. Comparison of 1992 and 1993 results does not suggest any significant change in the rate of tube degradation due to IGA/SCC.

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, Primary Water Stress Corrosion Cracking (PWSCC) at the roll transition continued to be active during the last operating cycle. This mechanism was first addressed in 1989 and this year there were 20 roll transition (PWSCC) indications in the "B" steam generator and 80 roll transition (PWSCC) indications in the "A" steam generator. These numbers include tubes that may have PWSCC in combination with IGA or SCC in the crevice.

Comparing the number of roll transition indications reported in 1992 with the number of these indications reported in 1993, results reveal that significantly fewer roll transition indications were reported in 1993. However, the number of these indications reported in 1992 was unusually high, and represents a data anomoly due to the first-time use of the MRPC technique for examining 100% of the roll transition and tubesheet crevice region. It is believed that a large number of pre-existing roll transition indications were first detected by MRPC in 1992, and had not been detected by previous standard bobbin coil techniques. The use of the MRPC probe for examining the roll transition region was continued in 1993.

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

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) below 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 (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 122 tubes repaired in the "A" steam generator, 51 tubes were repaired using a Combustion Engineering 27" welded sleeve in the hot leg, plus 62 tubes were repaired using a Babcock and Wilcox explosively welded tubesheet sleeve in the hot leg. All of the above tubes will remain in service. The remaining 9 tubes were removed from service by plugging both the hot and cold leg tube ends. A total of 194 tubes in the "A" steam generator are currently plugged and 668 tubes are sleeved.

- o Of the 171 tubes repaired in the "B" steam generator, 153 tubes were repaired using a Babcock and Wilcox explosively welded tube sheet sleeve in the hot leg. All of the above tubes will remain in service. The remaining 18 tubes were removed from service by plugging both the hot and cold leg tube ends. A total of 284 tubes in the "B" steam generator are currently plugged and 1286 tubes are sleeved.

All the above repairs on the "A" and "B" steam generators were completed on April 7, 1993 at approximately 2128 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. ADDITIONAL INFORMATION

A. FAILED COMPONENTS:

The degraded components are: Inconel 600 Mill Annealed U-Bend 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, 91-005, and 92-005.

C. SPECIAL COMMENTS:

For a more indepth report, refer to the Special Report "Summary Examination Report for the 1993 Steam Generator Eddy Current Inspection at R.E. Ginna Nuclear Power Station", Revision 1, dated April 20, 1993.

As a note of interest, RG&E has ordered new steam generators for R.E. Ginna Nuclear Power Plant to be installed in 1996.