ACCELERATED DOCUMENT DISTRIBUTION SYSTEM

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9404280277 DOC.DATE: 94/04/22 NOTARIZED: NO DOCKET #
FACIL:50-244 Robert Emmet Ginna Nuclear Plant, Unit 1, Rochester G 05000244
AUTH.NAME AUTHOR AFFILIATION
MARTIN,J.T. Rochester Gas & Electric Corp.
MECREDY,R.C. Rochester Gas & Electric Corp.
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 94-006-00:on 940323,both "A" & "B" steam Westinghouse Series 44 steam generators,required corrective action due to tube degradation.Cause was PWS.Corrective action:tubes were repaired.W/940422 ltr.

DISTRIBUTION CODE: IE22T COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 9
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

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD1-3 PD	1	1	JOHNSON,A	1	1
INTERNAL:	AEOD/DOA	1	1	AEOD/DSP/TPAB	1	1
	AEOD/ROAB/DSP	2	2	NRR/DE/EELB	1	1
	NRR/DE/EMEB	1	1	NRR/DORS/OEAB	1	1
	NRR/DRCH/HHFB	1	1	NRR/DRCH/HICB	1	1
	NRR/DRCH/HOLB	1	1	NRR/DRIL/RPEB	1	1
	NRR/DRSS/PRPB	2	2	NRR/DSSA/SPLB	1	1
	NRR/DSSA/SRXB	1	1	REG FILE 02	1	1
	RES/DSIR/EIB	1	1	RGN1 FILE 01	1	1
EXTERNAL:	EG&G BRYCE,J.H	2	2	L ST LOBBY WARD	1	1
	NRC PDR	1	1	NSIC MURPHY,G.A	1	1
	NSIC POORE,W.	1	1	NUDOCS FULL TXT	1	1

NOTE TO ALL "RIDS" RECIPIENTS:

PLEASE HELP US TO REDUCE WASTE! CONTACT THE DOCUMENT CONTROL DESK, ROOM P1-37 (EXT. 504-2065) TO ELIMINATE YOUR NAME FROM DISTRIBUTION LISTS FOR DOCUMENTS YOU DON'T NEED!

FULL TEXT CONVERSION REQUIRED
TOTAL NUMBER OF COPIES REQUIRED: LTTR 27 ENCL 27

R I D S / A D D S

R I D S / A D D S

ROCHESTER GAS AND ELECTRIC CORPORATION • 89 EAST AVENUE, ROCHESTER N.Y. 14649-0001

ROBERT C. MECREDY
Vice President
Ginna Nuclear Production

TELEPHONE
AREA CODE 716 546-2700

April 22, 1994

U.S. Nuclear Regulatory Commission
Document Control Desk
Attn: Allen R. Johnson
Project Directorate I-3
Washington, D.C. 20555

Subject: LER 94-006, 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 94-006 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 Senior Resident Inspector

IE22
1/1

NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	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.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
R.E. Ginna Nuclear Power Plant	05000244	1 OF 8

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 NAME	DOCKET NUMBER
03	23	94	94	--006--	00	04	22	94	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	N	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)
POWER LEVEL (10)	000		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
			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	TELEPHONE NUMBER (Include Area Code)
John T. St. Martin - Director, Operating Experience	(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	TBG	H314	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)

During the 1994 Annual Refueling and Maintenance Outage, subsequent to the eddy current examination performed on both the "A" and "B" Westinghouse Series 44 steam generators, 164 tubes in the "A" steam generator and 134 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 Station Quality Assurance Manual reportable 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 NRC Performance Indicator System Cause Code 5.8.4.3 and NUREG-1022 Cause Code (B).

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

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	94	-- 006 --	00	2 OF 8

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

I. PRE-EVENT PLANT CONDITIONS

The plant was in the cold/refueling shutdown condition for the 1994 Annual Refueling and Maintenance Outage. The Reactor Coolant System (RCS) was depressurized and RCS temperature was approximately 66 degrees F. Steam Generator (S/G) eddy current examination was in progress.

II. DESCRIPTION OF EVENT

A. DATES AND APPROXIMATE TIMES OF MAJOR OCCURRENCES:

- March 23, 1994, 1500 EST: The number of degraded S/G tubes was known to exceed Ginna Station Quality Assurance (QA) Manual reportable limits, based on completion of "A" and "B" S/G hot leg and "A" S/G cold leg inspection program. Event date and time.

- March 23, 1994, 1500 EST: Discovery date and time.

- March 24, 1994, 1300 EST: Oral notification made to the NRC Office of Nuclear Reactor Regulation (NRR) that the number of degraded S/G tubes exceeded QA Manual reportable limits.

- March 26, 1994, 1800 EST: All eddy current programs completed, and the evaluation of the 1994 inservice inspection of S/G tubes completed.

- March 27, 1994, 2235 EST: S/G repairs completed.

- April 8, 1994: A Special Report was sent to the USNRC, reporting the number of tubes plugged or sleeved in each S/G.

B. EVENT:

During the 1994 Annual Refueling and Maintenance Outage, an eddy current examination was performed in both the "A" (EMS01A) and "B" (EMS01B) 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 1993 examination.

The examination was performed by personnel from Rochester Gas and Electric (RG&E) and ABB Combustion Engineering. 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 examinations of the "A" and "B" S/Gs were 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" S/Gs 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 S/G 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 was examined full length. All Row 1 and Row 2 U-Bend regions were examined with the 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 164 tubes in the "A" S/G required repair (21 new repairs by plugging and 143 new repairs by sleeving). 134 tubes in the "B" S/G required repair (31 new repairs by plugging and 103 new repairs by sleeving). Corrective actions were therefore taken for 164 tubes in the "A" S/G, and for 134 tubes in the "B" S/G.

On March 26, 1994, at approximately 1800 EST, with the RCS depressurized and temperature at approximately 66 degrees F, final review of the 1994 S/G eddy current examination results was completed. Prior to completion of this review (on March 23, 1994, at approximately 1500 EST), it was evident that more than one percent of the total tubes inspected were 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 QA Manual.

On March 24, 1994, at approximately 1300 EST, oral notification was made to the NRC Office of Nuclear Reactor Regulation pursuant to Appendix B of the QA Manual.

On April 8, 1994, a Special Report listing the number of tubes required to be plugged or sleeved in each S/G, was reported to the NRC, pursuant to Appendix B of the QA 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:

This event was apparent during the review of the "A" and "B" S/G eddy current examination results.

F. OPERATOR ACTION:

The Control Room operators were notified that the number of degraded tubes exceeded the reportable limits of the QA Manual, and that the NRC (NRR) had already been notified by corporate staff. The 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 examination and repair supervision.

G. SAFETY SYSTEM RESPONSES:

None

III. CAUSE OF EVENT

A. IMMEDIATE CAUSE:

The immediate cause of the event was the "A" and "B" S/G tube degradation was in excess of the QA Manual reportable limits.

B. ROOT CAUSE:

The results of the examination indicate that Intergranular Attack (IGA) and Intergranular Stress Corrosion Cracking (IGSCC or SCC) continue to be active within the tubesheet crevice region on the inlet side of each S/G. As in the past, IGA/SCC is much more prevalent in the "B" S/G with 74 new crevice indications reported in 1994. In the "A" S/G, 42 new crevice indications were reported in 1994.

In 1993, 41 new crevice indications were reported in the "A" S/G, and 103 new crevice indications were reported in the "B" S/G. In 1994, 42 new crevice indications were reported in the "A" S/G, and 74 new crevice indications were reported in the "B" S/G. Comparison of 1993 and 1994 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 environment 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 120 roll transition (PWSCC) indications in the "A" S/G and 66 roll transition (PWSCC) indications in the "B" S/G. These numbers include tubes that may have PWSCC in combination with IGA or SCC in the crevice.

This event is NRC Performance Indicator System Cause Code 5.8.4.3, "Maintenance Equipment Failure", and 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 (Other) and the QA 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" S/Gs exceeded the criterion of (b) which states, "More than 1 percent of the total tubes inspected are degraded (imperfections greater than the repair limit)".

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 resulting from the S/G tube degradation in excess of the QA Manual reportable limits because:

- The degraded tubes were identified and repaired prior to any significant leakage or S/G tube rupture occurring.

- Even assuming a complete severance of a S/G 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:

- Of the 164 tubes repaired in the "A" S/G, 143 tubes were repaired using a 20 3/4 inch Babcock and Wilcox kinetically welded tubesheet sleeve in the hot leg. All of these 143 tubes will remain in service. The remaining 21 tubes were removed from service by plugging both the hot and cold leg tube ends. A total of 215 tubes in the "A" S/G are currently plugged and 811 tubes are sleeved.

- Of the 134 tubes repaired in the "B" S/G, 103 tubes were repaired using a 20 3/4 inch Babcock and Wilcox kinetically welded tubesheet sleeve in the hot leg. All of these 103 tubes will remain in service. 30 tubes were removed from service by plugging both the hot and cold leg tube ends. The remaining tube was previously plugged and exhibited a crack indication in the plug. The subject tube plug was removed and a new plug installed. A total of 315 tubes in the "B" S/G are currently plugged and 1390 tubes are sleeved.

All the above repairs on the "A" and "B" S/Gs were completed on March 27, 1994, at approximately 2235 EST.

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

B. ACTION TAKEN OR PLANNED TO PREVENT RECURRENCE:

The occurrence/presence of IGA, SCC, and PWSCC is a PWR S/G problem. Utilities with susceptible tubing and partially rolled crevices must deal with this recurring attack on S/G 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 S/G 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. The 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, 92-005, and 93-002.

C. SPECIAL COMMENTS:

A more in-depth final report will be submitted to the NRC, as required by the Ginna QA Manual.

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