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RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 90-004-00:on 900416,steam generator tube degradation due to IGA/SCC causes QA manual reportable limits to be reached.
W/9 ltr.

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May 10, 1990

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

Subject: LER 90-004, 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) 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 90-004 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
Division Manager
Nuclear Production

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

Ginna USNRC Senior Resident Inspector

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NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104 EXPIRES: 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
R.E. Ginna Nuclear Power Plant	0 5 0 0 0 2 4 4	1 OF 0 7

TITLE (4) Steam Generator Tube Degradation Due To IGA/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	1 6	9 0	9 0	– 0 0 4 –	0 0	0 5	1 0	9 0		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(a)(1)(i)
- [ ] 20.405(a)(1)(ii)
- [ ] 20.405(a)(1)(iii)
- [ ] 20.405(a)(1)(iv)
- [ ] 20.405(a)(1)(v)
- [ ] 20.405(c)
- [ ] 50.36(c)(1)
- [ ] 50.36(c)(2)
- [ ] 50.73(a)(2)(i)
- [ ] 50.73(a)(2)(ii)
- [ ] 50.73(a)(2)(iii)
- [ ] 50.73(a)(2)(iv)
- [ ] 50.73(a)(2)(v)
- [ ] 50.73(a)(2)(vii)
- [ ] 50.73(a)(2)(viii)(A)
- [ ] 50.73(a)(2)(viii)(B)
- [ ] 50.73(a)(2)(x)
- [ ] 73.71(b)
- [ ] 73.71(c)
- [X] OTHER (Specify in Abstract below and in Text, NRC Form 366A)

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (AREA CODE)
Paul Gorski Mechanical Maintenance 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	MANUFAC-TURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	

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 1990 Annual Refueling and Maintenance Outage subsequent to the eddy current examination performed on both the "A" and "B" Westinghouse Series 44 Steam Generators (S/G), 75 tubes in the "A" S/G and 211 tubes in the "B" S/G required corrective action due to tube degradation. This defect population includes 28 tubes in the "B" S/G that had known defects plugged in prior outages. These tubes were unplugged for full length eddy current examination and were returned to service with a sleeve repair in the degraded region.

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.

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

I. PRE-EVENT PLANT CONDITIONS

The unit was in cold/refueling shutdown for the Annual Refueling Maintenance Outage. All fuel had been removed from the Reactor Vessel. Steam Generator eddy current inspection was in progress.

II. DESCRIPTION OF EVENT

A. DATES AND APPROXIMATE TIMES FOR MAJOR OCCURRENCES:

- o April 16, 1990, 1711 EDST: Event date and time.
- o April 16, 1990, 1711 EDST: Discovery date and time.
- o April 18, 1990, 1000 EDST: Oral notification made to the NRC Office of Nuclear Reactor Regulation (NRR).
- o April 21, 1990, 1800 EDST: Steam Generator repairs completed.
- o May 1, 1990: Followup report sent to NRC Office of Nuclear Reactor Regulation (NRR).

B. EVENT:

During the 1990 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 May 1989.

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

The examination was performed by personnel from Rochester Gas and Electric Corporation (RG&E), and Allen Nuclear Associates. All personnel had been trained and qualified in the eddy current examination method and had been certified to a minimum of Level I for data acquisition and Level II for data analysis.

The eddy current examination of the "A" and "B" steam generators was performed utilizing the Zetec MIZ-18 Digital Data Acquisition System. The frequencies selected were 400, 200, 100, and 25 KHZ.

The inlet or hot leg examination program plan was generated to provide the examination of 100% of each open (not sleeved or plugged) steam generator tube from the tube end to the first tube support. In addition, 20% of these tubes were selected and examined for their full length as recommended in the EPRI guidelines. All tubes with previous indications greater than 20% through wall (TW) depth were examined at a minimum to the location of their degradation. Approximately 20% of all open Row 1 U-bend regions were examined with the Motorized Rotating Pancake Coil (MRPC) between the #6 Tube Support Plate Hot (TSPH) and the #6 Tube Support Plant Cold (TSPC) from the cold leg side.

Results of the above inspections indicated that 75 tubes in the "A" steam generator and 211 tubes in the "B" steam generator (183 new repairs plus 28 previously plugged tubes) required corrective action.

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 review of the eddy current examination results.

F. OPERATOR ACTION:

None.

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 Quality Assurance Manual Reportable Limits.

B. ROOT CAUSE:

The results of the examination indicate that the IGA and SCC continue to be active within the tube sheet crevice region on the inlet side of each steam generator. As in the past, the IGA/SCC is much more prevalent in the "B" steam generator with 108 IGA indications and 49 SCC indications reported. In the "A" steam generator 16 IGA indications and 22 SCC indications were reported.

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

In addition to the IGA/SCC in the crevices PWSCC indications continue to be found at the roll transition.

This year there were 23 PWSCC indications in the "B" steam generator and 37 PWSCC indications in the "A" steam generator.

IV. ANALYSIS OF EVENT

The 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 (b) which states, "More than 1% 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% 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 operational or safety consequences or safety implications resulting from the steam generator tube degradation in excess of the Q.A. Manual reportable limits because:

- o The degraded tubes were identified and repaired prior to any significant leakage or S/G 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 were assured at all times.

V. CORRECTIVE ACTION

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

o Of the 75 degraded tubes in the "A" Steam Generator, 51 tubes were repaired using a Combustion Engineering welded 27" sleeve in the hot leg and these tubes will remain in service. The remaining 24 tubes were removed from active service by plugging both the hot and cold leg tube ends.

o Of the 211 degraded tubes in the "B" steam generator, 191 tubes were repaired using a Combustion Engineering welded 27" sleeve in the hot leg and these tubes will remain in service. The remaining 20 tubes were removed from active service by plugging both the hot and cold leg tube ends.

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 Station will continue careful monitoring of both primary RCS and secondary side water chemistry parameters. These water chemistry parameters will 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:

None.

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 LER's 76-008, 77-008, 78-003, 79-006, 79-022, 80-003, 81-009, 82-003, 82-022, 83-013, and 89-001.

C. SPECIAL COMMENTS:

None.