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 12

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
04	07	95	95	--004--	00	05	08	95	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)			
POWER LEVEL (10)	000	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
		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 - Technical Assistant	(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 1995 Annual Refueling and Maintenance Outage, subsequent to the eddy current examination performed on both the "A" and "B" Westinghouse Series 44 steam generators, 88 tubes in the "A" steam generator and 122 tubes in the "B" steam generator required corrective action due to tube, sleeve, or plug 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.

I. PRE-EVENT PLANT CONDITIONS:

The plant was in the cold/refueling shutdown condition for the 1995 Annual Refueling and Maintenance Outage. The Reactor Coolant System (RCS) was depressurized and RCS temperature was approximately 100 degrees F. Steam Generator (S/G) eddy current examination was in progress.

II. DESCRIPTION OF EVENT:

A. DATES AND APPROXIMATE TIMES OF MAJOR OCCURRENCES:

- April 7, 1995, 1700 EDST: The number of degraded S/G tubes was known to exceed Ginna Station Quality Assurance (QA) Manual reportable limits. Event date and time.
- April 7, 1995, 1700 EDST: Discovery date and time.
- April 10, 1995, 1430 EDST: 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.
- April 13, 1995, 1200 EDST: All eddy current programs completed, and the evaluation of the 1995 inservice inspection of S/G tubes completed.
- April 15, 1995, 2100 EDST: S/G repairs completed.
- April 24, 1995: A Special Report was sent to the USNRC, reporting the number of tubes plugged or sleeved in each S/G.

B. EVENT:

During the 1995 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 1994 examination.

The examination was performed by personnel from Rochester Gas and Electric (RG&E) and Asea Brown Boveri - Combustion Engineering (ABB-CE). 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.

Sleeves were inspected using the Zetec "Plus Point" probe, which allows for improved inspection capability of the parent tube behind the sleeve. Since this advanced probe is more sensitive, it also can identify volumetric indications on the sleeve inside surface.

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

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 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 88 tubes in the "A" S/G required action (13 new repairs by plugging and 75 new repairs by sleeving). 122 tubes in the "B" S/G required action (31 new repairs by plugging and 91 new repairs by sleeving). Corrective actions were therefore taken for 88 tubes in the "A" S/G and for 122 tubes in the "B" S/G.

On April 7, 1995, at approximately 1700 EDST, with the RCS depressurized and temperature at approximately 100 degrees F, final review of the 1995 S/G eddy current examination results was completed. More than one percent of the total tubes inspected were degraded (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 April 10, 1995, at approximately 1430 EDST, oral notification was made to the NRC Office of Nuclear Reactor Regulation pursuant to Appendix B of the QA Manual.

On April 24, 1995, 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.

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

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. 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.

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

B. ROOT CAUSE:

1. TUBE DEGRADATION

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 80 new crevice indications reported in 1995. In the "A" S/G, 35 new crevice indications were reported in 1995.

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 1994 and 1995 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 60 roll transition (PWSCC) indications in the "A" S/G and 32 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." The tube degradation does not meet the NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants", definition of a "Maintenance Preventable Functional Failure".

2. SLEEVE INDICATIONS

Eddy current examination of 515 ABB-CE welded sleeves was performed using a "Plus Point" probe. This examination identified eleven (11) sleeves with inside surface (ID) volumetric indications at the upper weld location. Subsequent visual examination (VT) of the upper weld joint for these 11 sleeves identified five (5) sleeves with weld "tail-off" indications, three (3) sleeves with pinholes in the weld, two (2) sleeves with blowholes in the weld, and one (1) sleeve with no weld in the upper weld zone.

The eight (8) sleeves with weld tail-off and pinhole indications were left in service since these conditions did not affect either the structural integrity or the leak tightness of the upper weld. One of the sleeves with a blowhole indication was also left in service. The blowhole was located in the upper portion of the weld. Sufficient weld material existed below the blowhole location to provide both sleeve/tube structural integrity and a leak tight weld.

The second blowhole was located in the lower portion of the sleeve weld. Sufficient fusion existed for weld structural integrity. However, the possibility existed for a leak path to develop from the sleeve to the secondary side of the S/G. Although the sleeve could have been accepted as a leak-limiting sleeve, as a precautionary measure, the sleeve was repaired by plugging.

The one sleeve with no upper weld was identified as being a curved sleeve originally installed in 1990. In discussion with ABB-CE it was concluded that the lack of any weld in the sleeve was a result of an equipment problem with the flexible welding tool used to weld curved sleeves. Consequently, all installed curved sleeves were examined either by a review of the Plus Point data or by performing a VT inspection to verify the presence of an upper weld on the sleeve ID. This re-examination process discovered a second curved sleeve with no upper sleeve weld.

Both curved sleeves with missing upper welds were installed in 1990, and both welds were examined and accepted by UT to confirm the existence of weld fusion. The UT examinations were performed by the same UT inspector. As a result of the failure of the UT examination to discover the lack of an upper weld, all of the 113 sleeves examined by the one Level II UT inspector (in 1990) were re-examined by a different ABB-CE Level III UT inspector in 1995. This UT re-examination process discovered an additional six (6) sleeves that had inadequate weld fusion. All eight of the sleeves discovered with either missing or inadequate weld fusion during the 1995 outage were repaired by plugging.

The condition of these 8 ABB-CE sleeves was not identified during the original installation inspection because of the inexperience of the Level II UT inspector who performed the examination. The lack of fusion indications at the weld location resulted in the UT inspector mistaking the sleeve expansion region for a weld.

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.

- Sleeve indications do not present any operational safety consequences because there was no major defect in the design, construction, or installation of the ABB-CE welded sleeve which would have resulted in a structural failure of the installed sleeve. Additionally, the lack of adequate fusion during the installation process does not prevent the installed sleeve from functioning as a leak-limiting sleeve.

- The parent tube would remain constrained by the tubesheet and the installed sleeve. Therefore, RCS pressure boundary integrity has been maintained.

- No significant increase in radiation exposure or release to the general public would have occurred. Based on the maximum calculated leakage where a sleeve has a completely missing weld, the resulting primary to secondary leakage is well below the bounding leakage for the existing Ginna S/G tube rupture accident analysis.

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 88 tubes repaired in the "A" S/G, 75 tubes were repaired using a 20 3/4 inch Babcock and Wilcox kinetically welded tubesheet sleeve in the hot leg. All of these 75 tubes will remain in service. The remaining 13 tubes were removed from service by plugging both the hot and cold leg tube ends. A total of 228 tubes in the "A" S/G are currently plugged and 885 tubes are sleeved.

- Of the 122 tubes repaired in the "B" S/G, 91 tubes were repaired using a 20 3/4 inch Babcock and Wilcox kinetically welded tubesheet sleeve in the hot leg. All of these 91 tubes will remain in service. Of the remaining 31 tubes, 29 tubes were removed from service by plugging both the hot and cold leg tube ends. The other 2 tubes were previously plugged and exhibited minor leakage indications. For these 2 tubes, the tube plugs were removed and new plugs installed. A total of 343 tubes in the "B" S/G are currently plugged and 1464 tubes are sleeved.

- All sleeves that were inspected by the inexperienced Level II UT inspector in 1990 were re-examined in 1995 by a different ABB-CE Level III UT inspector, to verify acceptable weld fusion between the sleeve and the parent tube.

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.

- To ensure that this lack of experience was limited to the 1990 installation of ABB-CE welded sleeves, the records of all of the lead UT inspectors used by ABB-CE from 1986 to 1993 at Ginna Station were reviewed. This review determined that, excluding the one Level II UT inspector used in 1990, all of the other UT inspectors used at Ginna Station had prior experience with the performance of UT examination of these sleeves.

- A review of ABB-CE's complete sleeve installation and inspection program since 1984 has shown that this Level II UT inspector was not employed by ABB-CE for any other sleeving program at any other utility.

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	95	-- 004 --	00	12 OF 12

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

VI. ADDITIONAL INFORMATION:

A. FAILED COMPONENTS:

The degraded tubes 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, 93-002, and 94-006.

C. SPECIAL COMMENTS:

A more detailed 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.