U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Inspection Report 50-244/92-08 License: DPR-18

Facility: R. E. Ginna Nuclear Power Plant
Rochester Gas and Electric Corporation (RG&E)

Inspection: April 19 through May 26, 1992

Inspectors: T. A. Moslak, Senior Resident Inspector, Ginna
E. C. Knutson, Resident Inspector, Ginna

Approved by: _____ 6/11/92
W. J. Lazarus, Chief, Reactor Projects Section 3B Date

INSPECTION SCOPE

Plant operations, radiological controls, maintenance/surveillance, security, emergency preparedness, engineering/technical support, and safety assessment/quality verification.

INSPECTION OVERVIEW

Plant Operations: A 46-day refueling outage was completed. Mid-loop operations were well controlled. The number two low pressure turbine rotor was replaced following identification of blading cracks. Feedwater control system oscillations are being evaluated.

Radiological Controls: The ALARA program achieved the lowest outage collective personnel exposure, 228 person-rem, in the site's 23-year operating history.

Maintenance/Surveillance: Outage activities were effectively coordinated and performed, with minimal rework.

Security: Daily security practices, including shipping/receiving area controls, were observed to be effectively implemented.

Emergency Preparedness: A FEMA-observed radiological medical emergency exercise was conducted.

Engineering/Technical Support: Corporate and site engineering actively supported planned and emergent outage tasks.

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DETAILS

1.0 PLANT OPERATIONS (71707)

1.1 Operational Experiences

The inspection period began with Ginna in refueling shutdown. Following completion of steam generator U-tube repairs, the reactor coolant system (RCS) was drained to mid-loop level on April 27th for removal of steam generator nozzle dams. RCS fill and vent was completed on April 29th and forced circulation was restored on May 2. Using heat input from the operating reactor coolant pump, an RCS heatup to just below 350°F was conducted to support RCS hydrostatic testing. This test was completed on May 4th. Following steam generator secondary side crevice cleaning, RCS heatup resumed and normal operating temperature (547°F) was reached on May 7th. Operational testing in hot shutdown included main steam safety valve and control rod drop testing. All requirements for startup were completed on May 9th, and criticality was achieved at 3:23 PM. Startup physics testing was completed on May 10th. Steam plant startup was conducted and the main generator was closed on the grid at 10:12 PM on May 10th. Power ascension was halted at 30% for approximately one day due to secondary plant water chemistry limitations, and again at 60% to support a planned off-site power distribution system upgrade. Power escalation resumed on May 16th and full power (approximately 97%) was reached at 4:00 PM on May 17th. On May 18th, automatic feedwater control system instability produced high water level in the "A" steam generator which resulted in a momentary (six second) actuation of the feedwater isolation Engineered Safeguard Feature. This transient had no significant effect on reactor power. At the close of the inspection period, the plant was operating at steady state full power. No other significant operational challenges occurred during the inspection period.

1.2 Control of Operations

Overall, the inspectors found the R. E. Ginna Nuclear Power plant to be operated safely. Control room staffing was as required. Operators exercised control over access to the control room. Shift supervisors consistently maintained authority over activities and provided detailed turnover briefings to relief crews. Operators adhered to approved procedures and understood the reasons for lighted annunciators. The inspectors reviewed control room log books for activities and trends, observed recorder traces for abnormalities, assessed compliance with Technical Specifications, and verified equipment availability was consistent with the requirements for existing plant conditions. During normal work hours and on backshifts, accessible areas of the plant were toured. No inadequacies were identified.

1.3 Control of Operations with Reduced RCS Coolant Inventory (TI 2515/113)

At the beginning of the inspection period, steam generator nozzle dams were in place to support U-tube repairs while the remainder of the RCS remained filled. Installation and removal of the nozzle dams requires that reactor vessel water level be lowered to approximately the mid-loop level. Because of the reduced coolant inventory and the resultant increased impact of loss of core cooling, control of mid-loop operations was again carefully

evaluated during removal of steam generator nozzle dams. As noted during the installation of the nozzle dams earlier in the outage, administrative requirements were effective in ensuring required systems availability and maximizing reliability of off-site electrical power while in the mid-loop condition. Effective operator training was evidenced by transition through all phases of mid-loop operations with no significant operational deficiencies.

Through observation of operations, review of records and procedures, and interviews with operations and management personnel, the inspector assessed that licensee preparations for maintaining reliable decay heat removal during mid-loop operations were thorough and were effectively implemented during the 1992 refueling outage.

1.4 Pressurizer Safety Valve Loop Seal Drain Line Isolation Valve Leakage

The post-outage maintenance RCS hydrostatic test commenced on May 3rd. Several leaking mechanical joints were identified upon initially reaching test pressure, among which was the inlet flange to pressurizer safety valve 434. The leaking flange bolts were successfully tightened and the hydrostatic test was completed on May 4th.

On May 8th, while performing operations procedure O-1.1, "Plant Heatup from Cold Shutdown or Refueling Shutdown to Hot Shutdown," in preparation for reactor startup, annunciator F-18, "Pressurizer Safety Valve Outlet High Temperature 145°F," on the main control board alarmed. The purpose of this alarm is to alert operators to the possibility that one of the two pressurizer safety valves has actuated or has developed excessive seat leakage. Operators checked supporting indications and found that they were consistent with the cause being seat leakage from pressurizer safety valve 434; specifically, the digital valve position indication (VPI) for valve 434 indicated open 0.010 inch (had indicated open 0.006 inch in cold shutdown), and pressurizer relief tank (PRT) pressure and temperature were slowly increasing. The shift supervisor was concerned that these might be indications of imminent valve actuation. With RCS pressure still 500 psi below the design lift setpoint, actuation would indicate a gross mechanical problem with the valve. Based on this concern, the shift supervisor initiated RCS cooldown and depressurization.

Subsequent review of system configuration revealed that RCS leakage through the 3/4" loop seal drain line for valve 434 would produce indications similar to those of safety valve seat leakage. Although this drain line contained two shut valves, the proximity of these valves to the safety valve inlet flange made it conceivable that they had been inadvertently cracked open during the hydrostatic test repair work. Additionally, the validity of the digital VPI as an indication of leakage from valve 434 came into doubt as RCS pressure was reduced. PRT pressure and temperature stabilized, indicating that the leak had slowed or stopped; however, valve 434 VPI still indicated 0.010 inches open. The licensee decided to stabilize RCS temperature and pressure while the loop seal drain line isolation valve positions were verified.

Access to the valves in question required removal of the pressurizer cubicle missile shield. During this work and throughout the subsequent investigation, the licensee restricted containment access to only those involved in determining the source of the leakage. In checking shut the loop seal drain line isolation valves, approximately 1/8 turn was gained on each valve. The equivalent valves in the other pressurizer safety valve loop seal drain line were also checked shut, with no additional stem motion.

The Plant Operations Review Committee subsequently reviewed the event and concluded that leakage past the RCS safety valve loop seal drain line isolation valves had been corrected and recommended a cautious resumption of RCS heatup and pressurization. Hot shutdown conditions were achieved at 3:00 PM on May 8th, with no further indications of leakage.

The inspector was in the Control Room at the onset of this event and observed that operator response was prompt, professional, and appropriately conservative. Licensee evaluation of event classification and notification criteria was appropriate and correct. Corrective action and final resolution were deliberate and carefully considered.

1.5 Initial Criticality and Startup Physics Testing

As sequenced by periodic test (PT) 34.0, "Startup Physics Test Program," reactor startup in accordance with PT-34.1, "Initial Criticality, and ARO (all rods out) Boron Concentration," began on the morning of May 9th. The reactor trip breakers were closed at 9:07 AM. This initial startup attempt was aborted at 10:19 AM due to a blown fuse in the shutdown rod cluster control (RCC) bank rod control circuitry. The fuse was replaced and startup again commenced at 11:00 AM. Unlike a normal reactor startup, in which RCS boron concentration is adjusted prior to startup to achieve criticality as a result of control rod withdrawal to a specific configuration, initial criticality is achieved by first establishing RCS boron concentration high enough to maintain the reactor shutdown with all control rods withdrawn, and then diluting to achieve criticality with control rods already fully withdrawn. This provides the very gradual approach to criticality that is required to adequately monitor parameters for startup physics testing.

Criticality was achieved at 3:23 PM on May 9th. The startup physics test program consisted of determination of ARO boron concentration, measurement of the moderator temperature coefficient, control rod bank worth measurement, and determination of control rod bank boron end point concentration. Startup physics testing was completed at 5:24 AM on May 10th. All physics parameters were in close agreement with predicted values and were well within their allowable limits.

The inspector observed portions of the startup physics test program. The RG&E reactor engineer maintained excellent control of operations associated with this testing. Operations personnel were knowledgeable of test requirements and precautions. No deficiencies were noted.

1.6 Steam Generator Feedwater Flow Oscillations

On the morning of May 18th, control room operators noted excessive oscillations in steam generator feedwater flow with the advanced digital feedwater control system (ADFCS) operating in automatic. Troubleshooting revealed that the oscillations could be eliminated by taking manual control of both feedwater regulating valves (FRVs); manual control of either FRV while the other remained in automatic control reduced the magnitude of the oscillations, but did not eliminate them. Initial corrective actions consisted of lubrication of the valve stem guide mechanisms to minimize the possibility of mechanical sticking.

At 1:37 PM on May 18th, the "B" FRV failed in the fully open direction. The ADFCS was operating in automatic and no system maintenance or troubleshooting was in progress. Operators were alerted to the problem when the standby condensate pump started automatically due to low condensate header pressure as a result of the increase in feedwater flow. Increased feedwater flow to the "B" steam generator (SG) and resultant lower feedwater header pressure combined to decrease flow to the "A" SG. Consequently, the "A" FRV also received an open signal soon after the onset of the transient. Operator response was to place both FRVs in manual control and to restore normal SG levels. Although successful in doing so, level in the "A" SG momentarily exceeded the high level feedwater isolation setpoint of 67% for approximately six seconds during the recovery. The transient had no significant effect on reactor power.

Additional troubleshooting revealed that the magnitude of feedwater flow oscillations could be reduced by operating with the FRV bypass valves fully open. In effect, this decreases the gain signal applied to the feedwater flow portion of the system by reducing flow through the FRVs, thereby decreasing the magnitude of the oscillations without affecting plant power. At the end of the inspection period, the cause of the feedwater flow oscillations was still under investigation.

The inspector concluded that prompt, correct operator action had successfully mitigated a transient which otherwise would likely have resulted in a reactor trip. The inspector will continue to monitor licensee troubleshooting and corrective actions.

2.0 RADIOLOGICAL CONTROLS (71707)

2.1 Routine Observations

The inspectors periodically confirmed that radiation work permits were effectively implemented, dosimetry was correctly worn in controlled areas and dosimeter readings were accurately recorded, access to high radiation areas was adequately controlled, and postings and labeling were in compliance with procedures and regulations. Through observations of ongoing activities and discussions with plant personnel, the inspectors concluded that radiological controls were conscientiously implemented. No inadequacies were identified.

2.2 Outage ALARA Review

The total personnel exposure received during the 1992 refueling outage was the lowest in Ginna's operating history. Total outage exposure was 228 person-rem, as opposed to the outage goal of 305 person-rem. The most significant contribution to this lower than expected total exposure was made by development of an improved robotic sleeve manipulator system for use in SG U-tube repairs. This hybrid system, developed by RG&E personnel using equipment from the two SG repair contractors, significantly reduced the number of SG end-bell entries required for sleeve installation. Exposure for this portion of repairs was reduced from an estimated 50.6 person-rem to 35.962 person-rem.

Through personal observations, attendance of planning meetings, and interviews with personnel, the inspector assessed the following factors also to have been significant to the licensee's outage ALARA effort:

-- Improved job coordination, as a result of ALARA awareness on the part of RG&E's outage planning group (Planalog), as well as good communication and cooperation between the shops. As an example, exposure due to work in the vicinity of the SGs (in-service weld inspections, "A" SG insulation removal and reinstallation, scaffolding work, and optical templating in preparation for possible SG replacement) was significantly lower than expected due to Planalog coordination of maintenance with periods when the SG secondary sides were filled. The threefold reduction in general area radiation levels that secondary side water provides had been considered in developing the original exposure goals, but coordination effectiveness was greater than had been anticipated.

-- Strong, corporate management support. Senior management was frequently present at the twice-daily outage planning meetings, and expressed support for health physics department initiatives in support of ALARA. Management commitment to ALARA was further demonstrated in authorizing a delay of up to 24 hours in high-priority SG insulation work to coincide with a period when the SG secondary side would be full.

-- Simplified administrative requirements for installation of temporary shielding. As a result, temporary shielding was installed earlier in the outage and remained in place longer.

3.0 MAINTENANCE/SURVEILLANCE (62703, 61726)

3.1 Corrective Maintenance

3.1.1 Number Two Low Pressure Turbine Rotor Replacement

During routine outage inspection of the number two low pressure (LP) turbine, non-destructive testing revealed defects in numerous rotor blades. Specifically, cracks were detected in the bases of the blades where they attach to the rotor. The number one low pressure turbine rotor was subsequently inspected, and no similar deficiencies were identified. This made replacement with the spare rotor the most expeditious option for repair of the

number two LP turbine (as opposed to in-place blade replacement, which would have been required had both rotors been in need of repair). Identified three weeks into the outage, this significant increase in work scope immediately became a parallel critical path to outage completion. However, rotor replacement proceeded without major difficulties, and was completed without significantly impacting other maintenance or outage completion. The inspector concluded that replacement of the number two LP turbine rotor had been well coordinated and demonstrated excellent cooperation between the organizations involved.

3.1.2 (Closed) Unresolved Item (50-244/92-02-01) Main Steam Isolation Valve Performance

The inspector observed portions of the following maintenance activities:

-- Work Order 9240451, Main Steam Isolation Valve 3517 Major Inspection

-- Work Order 9240452, Main Steam Check Valve 3519 Major Inspection

These activities were inspected in light of a recent slow closure of main steam isolation valve 3517 upon demand; details of this event are presented in inspection report 92-02. Additionally, the inspector observed repacking of the equivalent valves in the "B" steam header, as discussed in inspection report 92-03. The inspector noted no visual mechanical defects upon inspection of the disassembled valves, nor did the licensee identify any significant problems during the course of refurbishment. In addition, during discussions with the inspector, the valve manufacturer representative stated that he had seen no problems with the main steam isolation valve that would have caused or contributed to the aforementioned closure problem.

Both valves were repacked using a new configuration of packing material (conical cross-section, as opposed to the previously installed rectangular cross-section) under supervision of the packing manufacturer. This new configuration was designed to reduce friction on the pivot shaft and was also used to repack the equivalent valves in the "B" main steam header. Operability of the main steam isolation valves was subsequently demonstrated by cold cycling under no-flow conditions per PT-2.10.5, "Main Steam Isolation Valve Exercising," as well as by no-flow cycling at normal system operating temperature and pressure.

Based on these observations, discussions, and test results, the inspector assessed that licensee actions to address main steam isolation valve operability had been appropriate. The inspector considers the issue of material condition as it relates to valve operability to be resolved; therefore, this item is closed.

3.2 Surveillance Observations

Inspectors observed portions of surveillances to verify proper calibration of test instrumentation, use of approved procedures, performance of work by qualified personnel, conformance to Limiting Conditions for Operation (LCOs), and correct system restoration following testing. The following surveillances were observed:

- Refueling Shutdown Surveillance Procedure (RSSP) 2.2, "Diesel Generator Load and Safeguard Sequence Test," revision 42, dated May 1, 1992, observed May 2, 1992.

- Periodic Test (PT)-7, "ISI (in-service inspection) System Leakage Test, Reactor Coolant System," revision 42, dated April 24, 1992, observed May 4, 1992.

- RSSP-7.0, "Control Rod Drop Test," revision 18, dated December 26, 1991, conducted May 8, 1992.

- PT-34.0, "Startup Physics Test Program," revision 25, dated December 26, 1991, observed May 9, 1992, consisting of:

  - PT-34.1, "Initial Criticality, and ARO (All Rods Out) Boron Concentration," revision 17, dated February 27, 1992, observed May 9, 1992.

  - PT-34.2, "Moderator Temperature Coefficient Measurement," revision 13, dated January 29, 1992, observed May 9, 1992.

  - PT-34.3, "RCC Bank Worth Measurement," revision 13, dated January 1992, observed May 9, 1992.

No unacceptable conditions were identified.

3.2.1 Main Steam Safety Valve Testing

Main steam safety valve lift setpoints are verified during each refueling outage. To conduct this testing, RCS temperature is adjusted to establish main steam pressure at slightly less than the design lift pressure of the main steam safety valves (MSSVs). A vendor-supplied air-operated lifting device is then used to provide the additional lifting force to the valve stem necessary to cause the valve to unseat (lift). Air pressure supplied to the lifting device is converted to an equivalent steam pressure which, when added to actual steam pressure, yields the lift setpoint for the valve.

The MSSVs in each steam header discharge into individual tailpieces. The valve discharge line protrudes into the tailpiece, but the two are not physically connected; the tailpiece is configured to function as a chimney rather than a pressure boundary. This configuration allows some steam to escape into the intermediate building when an MSSV lifts.

The inspector observed testing of MSSVs per RSSP 10.2, "Preparation for and Performance of Main Steam Safety Valve Test," revision 18, dated June 13, 1991, conducted May 8, 1992. At the completion of testing valves on the "A" header, the steam pressure test gauge was moved to the "B" main steam header. The inspector left the intermediate building during this work and was not present on resumption of testing. The first valve tested on the "B" main steam header lifted very abruptly and discharged an unexpected large quantity of steam into the intermediate building; in reaction to the noise and steam, people in the vicinity moved quickly away. One person was injured when he accidentally struck his head on a fixed object. The safety valve quickly reseated and the steam rapidly cleared. The injured person had received a cut on the face, but was conscious and able to walk. A medical emergency was declared and the injured person was transported by ambulance to Rochester General Hospital. There was no radioactive contamination involved. The individual was treated and released.

Subsequent investigation revealed that the "B" steam header drains upstream of the MSIV had been isolated the previous day to support unrelated maintenance. It is likely that water which accumulated in the steam header was the cause for the excessive steam release into the intermediate building. The upstream high pressure drains were placed in service and MSSV testing was completed without further incident.

In summary, MSSV operability was adequately demonstrated.

4.0 SECURITY (71707)

4.1 Routine Observations

During this inspection period, the resident inspectors verified that x-ray machines and metal and explosive detectors were operable, protected area and vital area barriers were well maintained, personnel were properly badged for unescorted or escorted access, and compensatory measures were implemented when necessary. Shipping/receiving personnel and security guards were observed to conscientiously carry out security practices for goods received at the unloading dock. No unacceptable conditions were identified.

4.2 Contractor Internal Audit Reveals Discrepancies in Procedures for Granting Unescorted Access

On April 24, RG&E was informed by Asea Brown Boveri Power Systems Energy Services Incorporated (ABB PSESI, contractor for steam generator U-tube repairs) that an internal audit had revealed inconsistencies in administrative requirements for granting unescorted access to their temporary employees. The same audit had determined that no discrepancies existed in their procedures for granting unescorted access to permanent employees. At the time of notification, ABB PSESI had not yet reviewed security screening records for the 13 temporary employees who were working at Ginna. As a precautionary measure, RG&E revoked unescorted access for these personnel as of midnight, April 24. This action did not

significantly affect the contractor work force effectiveness, since permanent ABB PSESI personnel retained unescorted access. RG&E subsequently conducted a review of all temporary, and 11 of 31 permanent, ABB PSESI personnel who had unescorted access during the 1992 outage. No disqualifying deficiencies were noted in RG&E's review.

The inspector determined that RG&E's response to this situation was appropriate. RG&E promptly provided courtesy notification to the NRC resident office and Region I. Follow-up action was appropriate and thorough. The inspector had no further concerns on this matter.

5.0 EMERGENCY PREPAREDNESS (71707)

5.1 Annual Emergency Radiation Injury Drill

On May 20th, RG&E conducted its annual emergency radiation injury drill. The drill was initiated with a simulated single person accident which occurred while decontaminating staging material. Drill monitors provided indications of multiple injuries such that treatment within the contaminated area was required. After site medical emergency personnel and an off-site contract physician had responded, the patient was transported by ambulance to Wayne-Newark Hospital (one of two participating area medical facilities). Off-site drill performance was evaluated by state and federal emergency response agencies.

The inspector observed the on-site portion of the drill and attended the post-drill critique. The inspector noted no deficiencies during conduct of the drill. Emergency personnel demonstrated good perspective in maintaining radiological controls consistent with the overriding priority of dealing with the medical emergency. The post-drill critique was self-critical and generated good recommendations for drill improvements, as well as for strengthening the emergency medical response team.

6.0 ENGINEERING/TECHNICAL SUPPORT (71707, 92701)

6.1 Steam Generator U-Tube Eddy Current Inspections and Repairs

At the beginning of the inspection period, steam generator U-tube eddy current inspections had been completed and had identified 226 defective tubes in the "A" SG and 183 defective tubes in the "B" SG. Defects detected by eddy current inspection typically consist of wall thinning which is the result of chemical corrosion or mechanical wear. Depending on the location and extent of the defect, corrective action for defective U-tubes can either be repair or removal from service. Repair is made by welding a concentric sleeve inside the portion of the U-tube where the defect exists; removal from service is accomplished by installing plugs into the U-tube inlet and outlet. Both of these actions reduce the heat transfer capability of the steam generator, a parameter that is important in reactor protection analysis. Therefore, heat transfer reduction due to U-tube plugging and sleeving is restricted by Technical Specification to no more than 15% per steam generator.

Advances in sleeving technology have made it possible to repair U-tube defects which once could only be corrected by plugging. Because the reduction in steam generator heat transfer capability due to a plugged tube is 25 times that of a sleeved U-tube, returning previously plugged U-tubes to service by sleeving can be used to offset heat transfer losses due to newly identified sleeving repairs. Based on historic inspection results and U-tube locations, RG&E identified 46 previously plugged U-tubes as candidates for return to service by sleeving. Full length eddy current inspection following plug removal revealed that all but one (which was subsequently replugged) met the criteria for repair by sleeving.

Corrective actions taken for steam generator defective U-tubes are summarized in Table 1. As developed in Table 2, the overall result of the steam generator U-tube work was a reduction of heat transfer capability equivalent to eight plugged U-tubes, or approximately 0.12%. The inspector concluded that the licensee complied with relevant Technical Specification criteria.

6.2 Licensee Action on Previous Inspection Findings

6.2.1 Probabilistic Risk Assessment Evaluations

-- (Closed) Unresolved Item (50-244/89-81-01), Evaluate Risk of Loss of Service Water to Diesel Generators

-- (Closed) Unresolved Item (50-244/89-81-03), Evaluate Residual Heat Removal (RHR) Net Positive Suction Head during Containment Recirculation

-- (Closed) Unresolved Item (50-244/89-81-08), Evaluate RHR Pump Seal Failure during Loss of Coolant Accident Scenario

The above mentioned items have been formally included in the licensee's Commitment and Action Tracking System (CAT numbers R00432, R00428, and R00424, respectively) to assure resolution. Each evaluation will be addressed in the RG&E Probabilistic Risk Assessment (PRA) which is presently under development and tentatively scheduled for completion in late 1992. Upon completion, the PRA will be submitted for NRC staff review in accordance with Generic Letter 88-20. Since the PRA will undergo a programmatic staff review, no additional resident or regional specialist follow-up is required at this time. These items are closed.

7.0 SAFETY ASSESSMENT/QUALITY VERIFICATION (71707, 90712, 90713, 92701, 40500)

7.1 Periodic Reports

Periodic reports submitted by the licensee pursuant to Technical Specification 6.9.1 were reviewed. Inspectors verified that the reports contained information required by the NRC, that test results and/or supporting information were consistent with design predictions and performance specifications, and that reported information was accurate. The following reports were reviewed:

-- Monthly Operating Report for April 1992

-- 10 CFR 21 Interim Report, Potential Inaccuracies Associated with Using the "Open Calibration" Approach to Measure the Closing Thrust of Motor Operated Valves

-- Annual Radiological Environmental Operating Report

-- Thirty (30) Day Special Report, Inoperable Fire Suppression System

No unacceptable conditions were identified.

7.2 Licensee Event Reports (LERs)

Two LERs submitted to the NRC were reviewed to determine whether details were clearly reported, causes were properly identified, and corrective actions were appropriate. The inspectors also assessed whether potential safety consequences were properly evaluated, generic implications were indicated, events warranted onsite follow-up, and applicable requirements of 10 CFR 50.72 were met.

The following LERs were reviewed (Note: date indicated is event date):

92-004 Personnel Exposure due to Radioactive Particle (April 4, 1992)

92-005 Steam Generator Tube Degradation due to IGA/SCC Causes QA Manual Reportable Limits to be Exceeded (April 20, 1992)

The inspector concluded that the LERs were accurate and met regulatory requirements. No unacceptable conditions were identified.

8.0 ADMINISTRATIVE (71707, 30702, 94600)

8.1 Backshift and Deep Backshift Inspection

During this inspection period, backshift inspections were conducted on the following dates: May 6 and 8, 1992. Deep backshift inspections were conducted on the following dates: April 25, May 2, 3, 9, and 10, 1992.

8.2 Systematic Appraisal of Licensee Performance (SALP) Management Meeting

A meeting was held on March 30th at the Ginna Training Center between RG&E and NRC management to discuss the NRC's evaluation of licensee performance for the period October 1, 1990 through January 18, 1992, as presented in the Ginna SALP report, 50-244/90-99.

8.3 Information Meeting With Local Officials

Following the SALP management meeting on April 30th, NRC representatives met with the Town Supervisor for the town of Webster, New York. During this informal meeting, the SALP process and Ginna SALP results were discussed. Responsibilities of the NRC and other general topics of interest were also discussed.

8.4 Exit Meetings

At periodic intervals and at the conclusion of the inspection, meetings were held with senior station management to discuss the scope and findings of inspections. The exit meeting for inspection report 50-244/92-08 was held on May 28, 1992 with the following individuals attending:

NAME	TITLE
NRC	
Thomas Moslak	Sr Resident Inspector
Edward Knutson	Resident Inspector
RG&E	
Terry Schuler	Manager, Operations
Andy Harhay	Manager, HP & Chemistry
Fred Mis	Health Physicist
Don Filion	Chemist
Steven Adams	Manager, Technical Support
Paul Gorski	Manager, Mechanical Maintenance
John St. Martin	Corrective Action Coordinator
Ron Jaquin	Engineer, Nuclear Safety and Compliance
Terry White	Operations
Michael Lilley	Manager, Nuclear Assurance
Robert McMahon	Quality Control Engineering

TABLE 1

STEAM GENERATOR U-TUBE CORRECTIVE ACTIONS

	Steam Generator "A"	Steam Generator "B"
U-Tubes Requiring Corrective Action		
-- Newly Identified Defects	226	183
-- Previously Plugged - Candidates for Return to Service	16	
-- Previously Plugged - Plug Repair Required	30	
-- U-Tube Stabilization Required (1)	0	3
	2	2
Total	244	218
U-Tubes Repaired		
-- Combustion Engineering welded sleeve		
-- 27" straight	36	124
-- 27" curved	0	62
-- 30" straight	0	9
-- Babcock and Wilcox explosively welded tubesheet sleeve	194	0
Total	230	195
U-Tubes Removed From Service		
-- Plugged	12	21
-- Stabilized	2	2
Total	14	23

(1) A precautionary modification to reenforce and anchor specific U-tubes which the steam generator manufacturer has identified as being susceptible to double-ended shearing; for purposes of this table, a stabilized tube is the same as a plugged tube.

TABLE 2

CHANGES IN STEAM GENERATOR HEAT TRANSFER CAPABILITY

	Steam Generator "A"	Steam Generator "B"
Pre-Outage Steam Generator Status		
-- U-Tubes Plugged	186	323
-- U-Tubes Sleeved	325	941
-- Equivalent Number of Plugged Tubes (1)	13.0	37.64
-- Percent Heat Transfer Degradation (2)	6.10%	11.06%
Outage Work		
-- Newly Sleeved Tubes	214	163
-- Equivalent Number of Plugged Tubes	8.56	6.52
-- Newly Plugged Tubes	14	22
-- Previously Plugged U-Tubes Returned to Service	16	29
-- Equivalent Number of Plugged Tubes (3)	15.36	27.84
-- Net Change in Number of Tubes (4)	-7.2	-0.68
Present Steam Generator Status		
-- U-Tubes Plugged	184	316
-- U-Tubes Sleeved	555	1133
-- Equivalent Number of Plugged Tubes	22.2	45.32
-- Percent Heat Transfer Degradation	6.33%	11.08%

(1) U-tube sleeving causes a reduction in reactor coolant flow through the tube due to reduction in the cross-sectional area. For purposes of determining the effect of sleeving on the steam generator heat transfer capability, 25 sleeved U-tubes are equivalent to one plugged U-tube. Therefore, the number of sleeved U-tubes expressed as equivalent plugged tubes is:

$$\text{Equivalent Number of Plugged Tubes} = \frac{\text{(Number of Sleeved Tubes)}}{25}$$

(2) The percent heat transfer degradation is the number of U-tubes with no flow (plugged tubes plus sleeved tubes expressed as equivalent plugged tubes) divided by the total number of U-tubes in the steam generator (3260).

$$\text{Percent Heat Transfer Degradation} = \frac{\text{(Number of Tubes Plugged)} + \text{(Number of Tubes Sleeved)}/25}{\text{Total Number of Tubes}}$$

3) Since sleeving causes a 1/25 reduction in flow from that of an unsleeved tube, a previously plugged U-tube returned to service by sleeving represent a gain of (24/25)tube.

(4) Net change in the number of tubes = [Gains] - [Losses]
= [(Previously Plugged Tubes Returned to Service)] -
[(Newly Plugged Tubes) + (Equivalent Plugged Tubes Due to Sleeving)]