U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-244/90-06

Docket No. 50-244

License No. DPR-18

Licensee: Rochester Gas and Electric Company
49 East Avenue
Rochester, New York 14649

Facility Name: R. E. Ginna Nuclear Power Plant

Inspection At: Ontario, New York

Inspection Dates: April 16-20, 1989

Inspector: H. J. Kaplan
H. J. Kaplan, Sr. Reactor Engineer, MPS, EB, DRS — 5-11-90 date

J. Carrasco
J. Carrasco, Reactor Engineer, MPS, EB, DRS — 5-11-90 date

Approved by: Jack Strosnider
J. R. Strosnider, Chief, Materials and Processes Section, Engineering Branch, DRS — 5/18/90 date

Inspection Summary: Routine unannounced inspection on April 16-20, 1990 (Report No. 50-244/90-06).

Areas Inspected: The inservice inspection program was reviewed including the ten year plan, steam generator eddy current testing, primary and secondary water chemistry results, and the erosion - corrosion control program. The pipe supports upgrade program and activities related to Bulletin 79-02 also were reviewed.

Results: No violations or deviations were identified.

DETAILS

1.0 Persons Contacted

Rochester Gas and Electric Company

* D. Filion, Radiation Chemistry
* D. L. Filkins, Health Physics and Chemistry Manager
* A. Gillet, Erosion Corrosion Coordinator
  P. Gorski, Supervisor, Inservice Inspection
* P. Lewis, NDE Outage Coordinator
* R. C. McCredy, General Manager, Nuclear Production
* J. St. Martin, Corrective Action Coordinator
  D. Morgan, Structural Engineer
* J. Quigley, Quality Service Coordinator
* M. J. Saporito, Supervisor, Materials Engineering
* J. F. Smith, Manager, Materials Engineering
* R. Smith, Senior Vice President Engineering and Production
* K. Wachter, Inservice Inspection Engineer
* R. Watts, Director Corporate Radiation Protection
* J. Widay, Superintendent Production

NUS Corporation

W. Tono Structural Engineer

U.S. Nuclear Regulatory Commission

* D. Wessman, NRR, Director PD I-3
* A. Johnson, NRR, DRP, PD I-3
* T. Moslack, Resident Inspector
* P. O'Connel, Region 1 Radiation Specialist
* N. Perry, Resident Inspector
* J. Jang, Region 1 Senior Radiation Specialist

* Denotes those attending the exit meeting.

The inspectors also contacted other administrative and technical personnel during the inspection.

2.0 Scope

The scope of this inspection was the review and observation of activities in the following areas:

-- the ten year inservice inspection (ISI) program

-- steam generator eddy current examination

-- the erosion corrosion (E/C) control program

-- primary and secondary water chemistry results

-- seismic upgrade program and pipe supports (Bulletin 79-02)

3.0 Inservice Inspection Program Review

R. E. Ginna station is in the first period of the third ten year inspection interval at this time. This inspection interval is based on the ASME Code, Section XI, 1986 with no addenda. The last interval was completed in December 1989.

The licensee reported that of the three hundred fifty four items covering class 1, 2, and 3 components scheduled for this period approximately ninety percent have been completed with no significant problems uncovered. In the Class 1 grouping the inspector randomly selected items for review. These included the nozzle to elbow welds for steam generators "A" and "B" and sixteen 1 3/4 diameter by 9 inch long manway bolts removed from the pressurizer. A review of appropriate nondestructive examination reports indicated acceptable liquid penetrant and ultrasonic results for the nozzle welds and acceptable visual and ultrasonic results for the bolts. The inspector reviewed the NDE procedures employed for these inspections and found the records conformed to the requirements of the respective procedures. These procedures were:

-- NDE-200-1, Revision 5, Liquid Penetrant Examination

-- NDE-600-17, Revision 0, Manual Ultrasonic Testing of Piping Welds

-- NDE-600-23, Revision 0, Manual Ultrasonic Examination of Bolting.

All procedures were approved by the licensee's certified level III examiner. Although no significant problems were found in the review of the ISI program the inspectors review of the licensee's Open Item List noted that two items (200545, and 900010) were dispositioned as NCR's and required grinding and/or repair as necessary because of linear liquid penetrant and magnetic particle indications that exceeded Code requirements. With regard to item 200545 it is noted that the licensee performed a metallurgical examination using replication (in-situ metallography) to determine the nature of the magnetic particle indications. The indications were attributed to an arc strike. Repair requirements included grinding, magnetic particle inspection and ultrasonic thickness measurement.

Conclusions

The licensee has maintained adequate control of the ISI program and has completed the scheduled inspections in a timely manner.

4.0 Steam Generator Eddy Current Inspection

The licensee established an inspection plan prior to the examination for both the "A" and "B" steam generators. The hot leg plan included 100% examination of tubes from the tube end to the first support plate. In addition 20% of these tubes were inspected full length. These 20% were as recommended by EPRI guidelines for the number of tubes to be inspected. All tubes with previous indications greater than 20% through wall were examined at the location of the indication. All row 1 and 2 U-bend regions were examined with the motorized rotating pancake coil (MRPC) between the number 6 hot and cold top support plates. The results of these inspections are shown in Table 1.

TABLE 1

Defect Type	Steam Generator		Total
	"A"	"B"	
PWSCC (Primary Water Stress Corrosion Cracking)	37	23	60
Wastage	0	1	1
SCC (Stress Corrosion Cracking)	22	49	71
IGA (Intergranular Attack)	16	108	124
Other	0	2	2
	75	183	258

Based on these results the licensee sleeved or plugged tubes as shown in Table 2.

TABLE 2

Disposition	Steam Generator		Total
	"A"	"B"	
Sleeved*	51	163	214
Plugged	24	20	44

* In addition, 28 tubes which had been previously plugged were sleeved.

The eddy current examination was performed using the Zetec Miz-18 Digital Data Acquisition System at frequencies of 400, 200, 100, and 25 kHz operated in both the differential and absolute modes. The examination was performed using standard 0.740 or 0.720 inch O.D. bobbin coil probes with smaller diameter coils used for the smaller radius U-bends and dented regions.

The tubes in the Ginna Station's steam generator are partially rolled into the tubesheet and seal welded, leaving approximately 19 inch crevice in the 22 inch tubesheet. From hot functional testing in November 1969 until the secondary water treatment was converted to all volatile in November 1974 the secondary water chemistry control was phosphate buffering. During this period an aggressive environment was concentrating in the sludge saturation zone and the crevices of the tubesheet. The current problems were likely initiated during this period of the plant operation.

The licensee had determined that tubes with defects in this region were to be sleeved whenever practical. To assure that the insertion and welding of the sleeves was acceptable the licensee developed a means to insert preformed sleeves into the tubes on the periphery of the steam generators. These sleeves were formed to allow for the curvature of the bottom head and were straightened as they were inserted. Autogenous welding was then performed from the inside of the sleeve. A mockup was extensively used to train the operators in each of the operations required for sleeve insertion and welding. The operations required for the sleeving process included marking the locations to be sleeved, verifying these locations, cleaning the inside of the tubes, cleaning the area to be welded on the existing tube end, inserting the sleeve, welding the top of the sleeve to the existing tube from the inside, seal welding the sleeve end to the tube at the tubesheet, and inspecting both welds. The licensee had designed the operations to minimize personnel exposure by automation as much as practical, monitoring by the use of cameras, and controlling the functions from a remote location. The sequence of the operations was also designed to minimize the number of tool changes, i.e. cleaning was performed on all tubes that could be reached by the fixture, followed by sleeve insertion etc. The inspector reviewed the ultrasonic inspection records for the automated tube sheet to sleeve welds as performed by Combustion Engineering. The inspector noted that where faulty welds occurred as in row #15 - column #86 because of lack of fusion, the weld had been rewelded satisfactorily as indicated by a review of the UT-video records for the subject weld. The sleeve material was Inconel 690 as verified by a review of the mill test reports. The licensee stated that the plug material was also Inconel 690.

Conclusions

The licensee performed a 100% inspection of all active tubes from the tube end to the first support in the hot leg of both steam generators to assure that the true condition of the steam generators was determined. In addition a 20% sample of the tubes was inspected full length and the row 1 and row 2 small radius U-bends were inspected using the Rotating Pancake Coil. The scope and methods of inspection appeared to appropriately consider the problem areas in the Ginna steam generators. It is also noted that the licensee had performed a QA audit of the eddy current activities of the steam generators in July 1989 (Report No. 89-10/05) and found no deficiencies.

5.0 Data Analysis

Data analysis was performed by two parties independently. One party used the Zetec Computer Screening (CDS) system with a manual review of the CDS calls. The other party performed the analysis manually using the Zetec DDA-4 Digital Data Analysis System. The results of these analysis were then compared for differences using the TUBAN computerized data management system. Differences were resolved by one or more Level III individuals. In addition, all tubes reported by either or both teams as requiring repair were reviewed by one or more Level III individuals. The removal of a tube from the repairable list required the concurrence of at least two Level III individuals.

Conclusions

The inspector was satisfied that the method of analyses was satisfactory to assure that defective tubes were identified.

6.0 Erosion Corrosion Control Program (IN 86-106)

The licensee initiated their erosion - corrosion program in 1983. Since that time the program has been expanded so that approximately 400 to 450 components are in the program. This includes approximately 88 components that are reexamined to establish the erosion rate and those that are approaching the replacement conditions. During this outage 316 components were inspected and 7 components were replaced due to wall thinning. The inspector reviewed the inspection records of one of the replaced components identified as a 16" straight pipe in the steam extraction system which measured .161" as compared to a .375" nominal wall thickness. The inspector determined that a nonconformance report for the subject component had been issued on 4/3/90 and dispositioned for replacement.

Conclusions

The licensee has an extensive erosion - corrosion control program that is used to monitor a large number of components in Ginna Station. The proportion of reexaminations indicates that data for trending erosion - corrosion is available for future analyses and planning. The program appears to be effectively identifying components that require replacement.

7.0 Water Chemistry - Primary and Secondary

Water chemistry data were reviewed as part of this steam generator inspection and maintenance inspection. The methods of collecting and verifying the accuracy of these data were not included in the scope of this inspection.

Reactor Coolant System Results

The Reactor Coolant System water chemistry as reported by the licensee for the last operating cycle has been stable. Normal variations occur during plant transients. The values reported are shown in Table 3.

TABLE 3

Parameter	Range	Specification
Lithium (ppm)	0.25 - 2.2	
Conductivity (uHmos)	30 - 14	
Boron (ppm)	850 - 0	
pH	5.5 - 7.7	
Oxygen (ppb)	0 - 20	100
Fluorine (ppm)	0.02 max.	0.15
Chloride (ppm)	0.1 max.	0.15

Secondary Water Chemistry Results

The secondary water system results measured in the steam generator blowdown was also stable during the the last cycle. However, this chemistry more strongly reflected the power transients experienced by the plant during this period due to hideout return in the steam generators. The values reported are shown in Table 4.

TABLE 4

Parameter	Results	Specification
Conductivity (uHmos)	0.08 - 0.15	0.15
Chloride (ppb)	1 - 5	5
Sulfate (ppb)	2 - 3	5
Sodium (ppb)	1 - 2	5
Oxygen (ppb)	5 - 10	10

Note: Not reflecting power transients

Conclusions

The stability of the water chemistry analysis indicate that the licensee has good control of the variables affecting water quality.

8.0 Loose Bolt Discovery Incident

During the inspection conducted April 16-20, the inspectors were informed of the presence of foreign material inside the reactor pressure vessel (RPV) on the core plate. This incident took place during refueling activities at the plant. The licensee provided a chronology of events as follows:

-- 3/23 Ginna station unit off line.

-- 3/29 The head of the Reactor was removed.

-- 3/30-4/02 The fuel was unloaded from the reactor and discharged in the spent fuel pool.

-- 4/03 Installed reactor head in preparation for steam generator and valve refurbishment work.

-- 4/13 Reactor head removed in preparation for reinserting fuel.

-- 4/14-4/15 Began loading fuel into the reactor on 4/14/90. After the loading of approximately ten (10) fuel assemblies into the reactor, the refueling person on the manipulator bridge, while looking into the vessel with binoculars, noticed the foreign debris setting on the core plate. The debris in question consisted of a non-magnetic bolt 3 1/2 inches long and 5/8 inches in diameter with the head missing. Also, a 3/4 inches long and 3/8 inches in diameter magnetic screw.

Following an assessment, loading of an additional nine assemblies which were located away from the debris continued while a retrieval tool was assembled. The debris were retrieved with the use of under water vice-grips and a video camera. The video camera was used to perform additional inspections since binoculars lacked adequate visibility. After inspections to verify there was no additional debris inside the RPV, the core loading was completed, and on 4/17 the reactor head was reinstalled.

The licensee contracted Westinghouse to determine the origin of the loose parts and to evaluate their significance. The licensee prepared a justification for continued operation dated April 23, along with a 10CFR 50.59 safety evaluation. The inspector reviewed these documents. The licensee's evaluations addressed two principle concerns. The first one being the fuel integrity/flow analysis associated with the potential loose parts and the second being the integrity of the RCP diffuser assembly.

In terms of the first concern, the missing head of the bolt and the socket head cap screw, Westinghouse concluded that it is too large, when intact, to pass though the bottom nozzle plate and up into the fuel assembly even if the socket head screw breaks up into smaller pieces and passes through the bottom nozzle. Westinghouse also indicated that it is unlikely that the broken pieces would migrate beyond the lower grid plate, since the individual flow holes in the grid are smaller than the bottom nozzle holes. The origin of the socket cap screw is not known.

Based on the Westinghouse evaluation, the licensee concluded that the 3 1/2 inches long and 5/8 inches in diameter bolt probably came from the Reactor Coolant Pump (RCP) diffuser casing adapter. This conclusion, was based on the physical configuration and dimensions of the subject bolt. The inspector reviewed a structural analysis (Westinghouse calculation #EA/S-90-63) performed to justify the continued usage of the RCP with one bolt missing in the diffuser adaptor. The calculation was performed for normal operation and seismic loads. A conservatism in the analysis was the assumption that the load normally carried by the missing bolt would be carried by only the two bolts adjacent to the supposedly missing bolt. The results of this analysis showed that the calculated stress was less than the yield strength for Stainless Steel, the designated material grade, as specified in Section III of the ASME Code.

Conclusions

The inspector did not identify any deficiencies in the licensee's activities to identify and evaluate the loose parts.

9.0 Licensee's Actions on Previous NRC Concerns

(Closed) Deviation (50-244/87-11-01) - Pipe support anchor factor of safety.

Background - During the NRC inspection conducted on May 5 - 8, 1987, the inspector determined that the licensee did not have calculations to demonstrate the factor of safety for pipe supports subjected to I.E. Bulletin 79-02.

Action item 2, of IE Bulletin 79-02, requires the licensee to verify that the concrete anchor bolts have minimum factors of safety of 4.0 for wedge type anchors and 5.0 for shell type anchors.

Licensee's actions

In response to this deviation, the licensee completed the appropriate calculations to clearly establish the factor of safety for concrete expansion anchor bolts for all large bore safety related systems not included in the program at the time of the inspection 87-11. These included Service Water (SW) System supports in the screen house and the Standby Auxiliary Feedwater (SAFW) System pipe supports.

Assessment and Conclusions

The inspector determined that the RG&E seismic pipe supports upgrade program is completed in terms of engineering calculations. These engineering calculations were performed to ensure that the factors of safety for the anchor bolts meet the requirements of IE Bulletin 79-02.

As a result of this engineering assessment the licensee is performing modifications to supports where necessary. The licensee stated that supports SWU-524, SWU-585 located in the Auxiliary Building and supports SWU-625, SWU-636 and SWU-623 located in the screen house are the only supports left to be modified. These modifications are scheduled to be completed by the end of the refueling outage. This is in accordance with RG&E's Management commitment to the USNRC Region I, as stated in RG&E letter of December 15, 1989.

During the SSFI inspection conducted on November 27 - December 8 the inspector reviewed several pipe supports on differert systems and concluded that the evaluations were being performed appropriately.

The inspector concluded that when the modifications are complete, the licensee will have satisfied their commitments with regard to Bulletin 79-02.

Based on the above, deviation 50-244/87-11-01 is closed.

(Closed) Deviation (50-244/87-11-02) Pipe supports anchor bolts were not inspected for support load changes from compression to tension.

Background

IE Bulletin 79-02, action item 4, requires that each anchor bolt used in piping system must be inspected unless it can be established that tension loads do not exist. As a result of the licensee's seismic upgrade program some of the support anchor bolts that originally were under compression were considered to be in tension.

During inspection 87-11 the NRC inspectors and the licensee noticed that five supports displayed this change in loading. This observation resulted in a deviation, since this change in loading constituted a departure from the licensee's commitment to action item 4 of IE Bulletin 79-02.

Licensee's actions

The licensee performed a detail review of the latest support information, and determined that in fact, one support of the initial group of five had no tension loads, while the remaining four supports were redesigned to use new Hilti Kwik-bolts in place of the existing shell type anchor bolts. This new bolt configuration was designed in accordance to the requirements of IE Bulletin 79-02. These supports are located on systems which were modified as part of the original scope of the Seismic upgrade program. The supports in question were: CCU-86, CCU-156, SWU-196 and SWU-198.

Assessment and Conclusions

The inspector reviewed the calculation to qualify these four supports to accommodate the bolt replacement. The review was performed to ensure that basic considerations prescribed in the bulletin were incorporated in the calculations.

Postulated seismic loads and base plate flexibility, which introduces an additional load on the bolts, were properly considered in the analyses.

Based on this review, the inspector concluded that the four supports were properly qualified and modified in accordance with the Bulletin.

Deviation 87-11-02 is closed.

10.0 Management Meetings

Licensee management was informed of the scope and purpose of the inspection at the entrance interview at the start of the inspection. The findings of the inspection were discussed with licensee representatives during the course of the inspection and presented to licensee management at the April 20, 1990 exit interview (see paragraph 1 for attendees).

At no time during the inspection was written material provided to the licensee by the inspector. The licensee did not indicate that proprietary information was involved within the scope of this inspection.