U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT

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Region I

Report No. 50-244/81-09

Docket No. 50-244

License No. DPR-18 Priority -- Category C

Licensee: Rochester Gas and Electric Corporation

89 East Avenue

Rochester, New York, 14649

Facility Name: R. E. Ginna Nuclear Power Plant

Inspection at: Ontario, New York

Inspection conducted: May 1-31, 1981

Inspectors: R. P. Zimmerman 7/14/81
R. P. Zimmerman, Senior Resident Inspector — date signed

date signed

date signed

Approved by: H B Kister 7/16/81
H. B. Kister, Chief, Reactor Projects Section 1C, Division of Resident & Project Inspection — date signed

Inspection Summary:

Inspection on May 1-31, 1981 (Report No. 50-244/81-09)
Areas Inspected: Routine, onsite, regular, and backshift inspection by the resident inspector (74 hours). Areas inspected included plant operating records; surveillance testing; IE Bulletin response; periodic and special reports; Licensee Event Reports; licensee action on previous inspection findings; and accessible portions of the facility during plant tours.
Results: No items of noncompliance were identified during this inspection.

Region I Form 12
(Rev. April 77)

DETAILS

1. Persons Contacted

   The below listed technical and supervisory level personnel were among those contacted:

   W. Backus, Operations Supervisor
   J. Bodine, QC Engineer
   L. Boutwell, Maintenance Supervisor
   W. Dillion, Supervisor of Nuclear Security
   C. Edgar, I & C Supervisor
   D. Filkins, Supervisor Health Physics and Chemistry
   D. Gent, Results and Test Supervisor
   G. Larizza, Technical Engineer
   R. Morrill, Training Coordinator
   T. Meyer, Nuclear Engineer
   J. C. Noon, Assistant Plant Superintendent
   C. Peck, Operations Engineer
   B. Quinn, Health Physicist
   B. A. Snow, Plant Superintendent
   S. Spector, Maintenance Engineer

   The inspector also interviewed and talked with other licensee personnel during the course of the inspection.

2. Licensee Action on Previous Inspection Findings

   (Closed) Noncompliance (244/81-03-01): The licensee held a radiation emergency drill on February 18, 1981. Inspector comments following witnessing of the above drill are included in IE Inspection Report 81-04. The licensee stated in the April 10, 1981 letter responding to the item of noncompliance, that insufficient manpower and plant workload was the primary reason for not meeting the required drill date. To preclude recurrence, three individuals have been added to the training department complement.

3. Review of Plant Operations

   a. General

      Throughout the reporting period, the inspector reviewed plant operations associated with the annual refueling, modification, and maintenance outage, major activities in progress included 'sleeving' 16 tubes in the 'B' Steam Generator; radiography of the 'A' Reactor Coolant Pump bowl longitudinal welds following full core unload; fire protection system upgrade; and seismic support modifications.

b. Shift Logs and Operating Records

Operating logs and records were reviewed against Technical Specifications and administrative procedure requirements. Included in the review were:

Control Room Log	-	daily during control room surveillance
Daily Surveillance Log	-	daily during control room surveillance
Shift Supervisor's Log	-	daily during control room surveillance
Plant Recorder Traces	-	daily during control room surveillance
Plant Process Computer Printout	-	daily during control room surveillance
Station Event Reports	-	5/1/81 through 5/31/81

The logs and records were reviewed to verify that entries are properly made; entries involving abnormal conditions provide sufficient detail to communicate equipment status, deficiencies, corrective action restoration and testing; records are being reviewed by management; operating orders do not conflict with the Technical Specifications; logs and event reports detail no violations of Technical Specification or reporting requirements; logs and records are maintained in accordance with Technical Specification and administrative procedure requirements.

c. Plant Tour

1. During the course of the inspection, tours of the following areas were conducted:

   -- Control Room

   -- Containment

   -- Auxiliary Building

   -- Intermediate Building (including control point)

   -- Service Building

   -- Turbine Building

   -- Diesel Generator Rooms

   -- Battery Rooms

-- Screenhouse

-- Yard Area and Perimeter

2. The following observations resulted from the tours:

   a. Monitoring instrumentation. Process instruments were observed for correlation between channels and for conformance with Technical Specification requirements.

   b. Annunciator alarms. Various alarm conditions which had been received and acknowledged were observed. These were discussed with shift personnel to verify that the reasons for the alarms were understood and corrective action, if required, was being taken.

   c. Shift manning. Control room and shift manning were observed for conformance with 10 CFR 50.54 (K), Technical Specification, and administrative procedures.

   d. Radiation protection controls. Areas observed included control point operation, posting of radiation and high radiation areas, compliance with Radiation Work Permits and Special Work Permits, personnel monitoring devices being properly worn, and personnel frisking practices.

   e. Equipment lineups. Valve and electrical breakers were verified to be in the position or condition required by Technical Specifications and plant lineup procedures for the applicable plant mode. This verification included control board indications daily and field observations made during routine plant tours.

   f. Equipment tagging. Selected equipment, for which tagging requests had been initiated, was observed to verify that tags were in place and the equipment in the condition specified.

   g. Fire protection. Fire detection and fire fighting equipment was observed for conformance with Technical Specifications and administrative procedures.

   h. Security. Areas observed for conformance with regulatory requirements, the site security plan and administrative procedures, included vehicle and personnel access, protected and vital area integrity, escort and badging.

i. Plant housekeeping controls. Plant conditions were observed for conformance with administrative procedures. Storage of material and components was observed with respect to prevention of fire and safety hazards. Housekeeping was evaluated with respect to controlling the spread of surface and airborne contamination.

3. Inspector Witnessing of Surveillance Tests

a. The inspector witnessed the performance of surveillance testing of selected components to verify that the surveillance test procedure was properly approved and in use; test instrumentation required by the procedure was calibrated and in use; Technical Specifications were satisfied prior to removal of the system from service; test was performed by qualified personnel; the procedure was adequately detailed to assure performance of a satisfactory surveillance; and test results satisfied the procedural acceptance criteria, or were properly dispositioned.

b. The inspector witnessed the performance of:

-- Periodic Test (PT)-32.1, Plant Safeguard Logic Test A or B Train, Revision 4, May 7, 1980, performed on May 13, 1981.

-- PT-25, Containment Post Accident Charcoal Filter By-Pass Flow, Revision 4, September 22, 1980, performed on May 6, 1981.

N[illegible] of noncompliance were identified.

4. Review o[illegible] [illegible]eriodic and Special Reports

a. Upon receipt, periodic and special reports submitted by the licensee pursuant to Technical Specification 6.9.1 and 6.9.3 were reviewed by the inspector. This review included the following considerations: the report included the information required to be reported by NRC requirements; test results and/or supporting information were consistent with design predictions and performance specifications; planned corrective action was adequate for resolution of identified problems; determination whether any information in the report required classification as an abnormal occurrence; and the validity of reported information. Within the scope of the above, the following periodic reports were reviewed by the inspector.

-- Monthly Operating Report for April, 1981.

-- Annual Report of Changes to Station Facilities and Procedures-1980.

b. The Annual Report of Changes to Station Facilities and Procedures-1980, submitted April 29, 1981, was limited in scope to procedure change notices and the associated Plant Operations Review Committee (PORC) item number. The inspector stated that 10 CFR 50.59 requires that the report contain a brief description of the changes made to the facility as described in the safety analysis report, and, tests and experiments performed which are not described in the safety analysis report. A summary of the safety evaluation performed to establish that an unreviewed safety question did not exist is also required to be submitted for each item. The licensee representative stated that a supplementary report will be prepared and submitted containing the above information.

With regard to the report submitted April 29, the inspector informed the licensee representative that it is not required that each procedure change notice reviewed by PORC be submitted in the annual report; only those changes to procedures as described in the safety analysis report.

5. Licensee Event Report (LER's)

The inspector reviewed the following LER's to verify that the details of the event were clearly reported, including the accuracy of the description of cause and adequacy of corrective action. The inspector determined whether further information was required, and whether generic implications were involved. The inspector also verified that the reporting requirements of Technical Specifications and Station Administrative and Operating Procedures had been met, that appropriate corrective action had been taken, that the event was reviewed by the Plant Operations Review Committee, and that the continued operation of the facility was conducted within the Technical Specification limit.

81-09: Abnormal Degradation of Steam Generator Tubes-May 15, 1981. (Repeat Event: 79-06, 79-22 and 80-03) Multi-frequency eddy current examination of 100% of both steam generator inlets and 25% of the outlets revealed fourteen indications attributable to intergranular attack in the tube sheet crevice region of the 'B' Steam Generator inlet. Five of the fourteen crevice indications were above the plugging criteria of 40% degradation. Additionally, one tube was identified with a wastage defect in the number 4 wedge area (periphery) of the 'B' Steam Generator inlet. Thirteen of the fourteen crevice indications were sleeved, and one was pulled for examination. The tube with a defect in the wedge area was plugged. In total, sixteen tubes were sleeved and three tubes were pulled for examination.

During extended outage periods, the licensee has instituted a crevice flushing program to remove chemical impurities (Na and PO_4) from the tube sheet crevices in an effort to minimize tube degradation.

81-10: Leaking Relief Valve in Letdown System-April 20, 1981. During cleanup of the Reactor Coolant System prior to opening the system for the annual maintenance and refueling outage, a crud buildup in the primary demineralizers resulted in an increase in backpressure, lifting relief valve 209 in the letdown system. Primary fluid which normally relieves to the Volume Control Tank, leaked through a telltale hole in the valve due to a cracked bellows. The affected area in the Auxiliary Building was evacuated. Air sample results in the vicinity indicated a total iodine activity of 4.9.2E uc/cc and a particulate activity of 4.7E-9 uc/cc. The telltale hole was plugged and the area decontaminated prior to being reopened to general access. The valve was subsequently repaired and reinstalled in the letdown system.

81-11: Leaking RHR Pump Seal Cooler Fitting-April 26, 1981. With the unit in the hot shutdown mode and RHR in service, a small primary leak was discovered in a fitting of the 'B' RHR Pump Seal Cooler. Following isolation of the 'B' RHR Pump, the threads of the fitting were observed to be worn and the fitting was replaced.

6. Letdown Orifice Isolation Valves - Safety Evaluation

Following a review of the diverse containment isolation relay failure event, which occurred April 10, 1981 (IE Inspection Report 81-08), the licensee determined that the letdown orifice isolation valves (200A, 200B and 202) do not receive a containment isolation signal as depicted in Figure 5.2.2-4 of the Final Safety Analysis Report (FSAR). An apparent inconsistency exists, however; in that the letdown piping in the current configuration does satisfy the definition of Class 1 piping addressed in Section 5.3.2 of the FSAR.

The licensee performed a safety evaluation to determine if failure of valves 200A, 200B and 202 to receive a containment isolation signal constituted an unreviewed safety question. The safety evaluation determined that an unreviewed safety question did not exist, based in part on the following:

-- Section 14 (Safety Analyses) of the FSAR does not include an accident analysis which assumes that the letdown orifice isolation valves close on a containment isolation signal to mitigate the consequences of an accident;

-- The letdown orifice isolation valves do receive a close signal on low pressurizer level.

-- Initiation of a safety injection signal will give a containment isolation signal which in turn isolates instrument air to containment, failing valves 200A, 200B and 202 in the closed position.

-- A pipe rupture downstream of the letdown orifice isolation valves, either inside or outside of containment would be limited by the number of valves in service and the size of the orifice diameters. The maximum orifice diameter is .274 inches and the two remaining orifice diameters are each .234 inches. Line breaks of this size in containment have been analyzed by the small break (less than 6") loss of coolant accident analysis in the FSAR.

-- A pipe rupture outside of containment, between the letdown orifice isolation valves and the letdown isolation valve outside of containment (371; shuts on a containment isolation signal) was evaluated and found to be less severe than a steam generator tube rupture as analyzed in the FSAR.

The inspector stated that in the current configuration, and as demonstrated in the April 10 event, relief valve 203 can be required to open to prevent overpressurization of the piping between the letdown orifice isolation valves and valve 371 following a containment isolation signal. The licensee representative stated that in an effort to limit challenges to relief valve 203, an engineering work request has been instituted to review and evaluate modifying the containment isolation signal logic to include the letdown orifice isolation valves. The inspector will follow licensee actions on the above work request.

7. Followup on IE Bulletins (IEB)

The inspector reviewed facility records, interviewed licensee personnel and observed facility equipment/components to verify that:

-- licensee management received and reviewed the bulletins in accordance with administrative procedures:

-- information discussed in the licensee's bulletin response was accurate:

-- corrective action was taken as discussed in the reply; and,

-- the licensee's response was within the time period required.

IEB 79-25, Failures of Westinghouse BFD Relays in Safety-Related Systems

The licensee performed an inspection to determine if the relays affected by armature sticking were in use in safety-related systems. Results of the inspection identified the subject relays in use in the Reactor Protection System and Circulating Water Pump Trip Logic Circuitry. The relays are tested each refueling outage as part of the routine surveillance program, and no failures relative to armature sticking have been noted to date. The licensee intends to maintain the existing test frequency, based on the test result history.

The inspector reviewed the following relay test results, with no relay malfunctions identified.

-- Periodic Test (PT)-14, Circulating Water Pumps - High Water Trip Logic, Revision 4, October 5, 1977, performed on March 31, 1980.

-- PT-32, Reactor Trip Logic Test 'A' or 'B' Trains, Revision 8, March 17, 1979, performed April 29, 1980 for both trains.

The inspector stated that although no relay failures had occurred to date, a mechanism should be developed to ensure that if a relay failure as described in IEB 79-25 were to occur during future surveillance testing, a re-review of IEB 79-25 would be conducted to determine the scope of necessary corrective action. The licensee representative acknowledged the inspector's comment and agreed to revise maintenance Procedure 51.1, Changing of a Reactor Protection Relay, to require an evaluation of the cause of a relay failure for similiarity to the failure mode described in IEB 79-25.

With regard to the relay style for which the possibility for an overtravel deficiency was identified, five of twenty-eight reactor trip logic relays and one of eight spares were found to have less than the minimum established overtravel of 0.02 inches. The affected relays were replaced with qualified spares in December, 1979.

8. Exit Interview

At periodic intervals during the course of the inspection, meetings were held with senior facility management to discuss the inspection scope and findings.