U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Report No. 50-244/86-06

Docket No. 50-244

Licensee No. DPR-18 Priority -- Category C

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

Facility Name: R. E. Ginna Nuclear Power Plant

Inspection at: Ontario, New York

Inspection Conducted: March 1, 1986 through March 31, 1986

Inspectors: W. A. Cook, Senior Resident Inspector, Ginna
T. K. Kim, Resident Inspector (Trainee), Ginna

Reviewed by: J E Beall 4/22/86
J. E. Beall, Project Engineer, Date
Reactor Project Sect. No. 2A, DRP

Approved by: Robert M Gallo 4/22/86
R. M. Gallo, Chief, Reactor Date
Project Section No. 2A, DRP

Inspection Summary:
Inspection on March 1, 1986 through March 31, 1986 (Report No. 50-244/86-06)
Areas Inspected: Routine, onsite, regular, and backshift inspection by the resident inspectors (151 hours). Areas inspected included: plant operations; licensee action on previous findings; surveillance testing; maintenance; startup physics testing; CILRT review; review of periodic and special reports; and inspection of accessible portions of the facility during plant tours.

Results: In the seven areas inspected, no violations were identified. Startup physics testing was reviewed and documented in paragraph 6, (minor procedural deficiencies were identified). The containment integrated leakage rate test was reviewed and documented in paragraph 7.

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DETAILS

1. Persons Contacted

   During this inspection period, the inspectors held discussions with and interviewed operators, technicians, engineering and supervisory level personnel.

2. Licensee Action on Previous Inspection Findings

   a. (Closed) Unresolved Item (82-08-01) During the review of the 1982 Containment Integrated Leakage Rate Test (CILRT), the inspector determined that the containment liner had channels attached to each seam weld. These channels were used during construction for acceptance testing of the liner welds. As shown on detailed fabrication drawings, all liner weld channels were sealed with a threaded plug. The NRC position is that all containment liner weld channels will be vented to the containment atmosphere during the CILRT, unless the licensee has demonstrated that the weld channels will maintain their integrity when subject to the loading conditions of a Design Basis Accident (DBA). The subject weld channels were inaccessible at the time of the 1982 inspection due to insulation covering the containment walls and lack of easy access to the uninsulated containment dome surfaces. The licensee committed to initiate an engineering evaluation and prepare a justification for CILRT testing without vented weld channels.

      The inspector reviewed the licensee's engineering evaluation of the containment liner weld channels conducted in accordance with Engineering Work Request No. 3714. The purpose of the evaluation was to determine the capability of the weld channels to maintain their structural integrity under DBA conditions. The licensee determined that the weld channels were installed in conformance with ASTM A36 and that the channels were attached with 1/4 inch continuous fillet welds. Weld procedures and welders were qualified to the ASME Boiler and Pressure Vessel Code. Nondestructive testing of the weld channels included liquid penetrant and radiography. In addition, the channel weld connections were tested to 69 psig and examined using a soap bubble test and/or mixture of air and freon. Any detectable leaks were properly repaired and retested. The licensee concludes that: original field testing of the weld channels to pressures exceeding containment design pressure by 15 percent; installation of the weld channels in accordance with the same quality control standards as the containment liner; and satisfactory completion of periodic leak rate testing in accordance with Technical Specifications provide reasonable assurance that the containment liner weld channels will maintain their integrity during a DBA. The inspector had no further questions.

b. (Closed) Violation (85-04-01) During an earlier inspection, the inspectors determined that preventive maintenance on station valves listed in Administrative Procedure A-1020, "Valve Preventive Maintenance Program", was not being properly performed. In addition, periodic reviews of A-1020 by maintenance supervision to track program progress were not conducted. The licensee's response to this violation is documented in RG&E letter dated September 25, 1985. Procedure A-1020 was revised by the licensee to reflect a more practical and workable valve preventive maintenance program. The inspector reviewed the revised procedure and discussed it with the Maintenance Manager. No discrepancies were noted. In addition, the licensee has created a new maintenance position, Preventive Maintenance Coordinator/Planner, who will act to improve the scheduling of all routine maintenance on station. The inspector verified this position was filled, effective January 6, 1986, by an experienced member of the maintenance staff and discussed the responsibilities of this new position with the individual. In the long term, the licensee intends to develop and implement a computerized Maintenance Information System. The inspector discussed progress in developing this computerized program with licensee representatives. Full implementation of this computerized system is not expected until December 1987. This item is closed.

c. (Closed) Inspector Follow-up Item (85-07-03) During the 1985 Refueling Outage, the inspector noted that numerous containment evacuation alarms were automatically sounded by the source range nuclear instruments generating spurious "High Flux at Shutdown" signals. The source range "High Flux at Shutdown" signals resulted from welding activities inside containment causing voltage spikes in the source range detector outputs. Consequently, workers inside containment were desensitized to the containment evacuation alarms, followed by public address system announcements to disregard the same.

   During the 1986 outage, the inspector determined that the licensee blocked the "High Flux at Shutdown" containment evacuation signals on both source range nuclear instruments. In addition, Operator Aid tags were affixed to the bistable selector switches to identify the reason for the source range nuclear instrument function being blocked and to alert control room operators to manually initiate a containment evacuation alarm, if conditions warranted. This item is closed.

d. (Open) Unresolved Item (85-09-01) During the 1985 review of the licensee's Startup Physics Testing Program, rod bank worth predictions were significantly lower than measured values. The licensee suspected Westinghouse computer software fuel performance modeling problems because predicted RCC bank boron end point calculations were reasonably accurate. The same anomaly with inaccurate rod bank worth predictions was observed during Cycle 16 physics testing in March 1986. The licensee plans to meet again with Westinghouse representative to resolve this discrepancy.

3. Review of Plant Operations

a. During this reporting period, the inspectors reviewed the licensee's activities in completing the 1986 Refueling and Maintenance Outage and returning the unit to 100% power operations. The reactor was returned to criticality at 6:45 A.M., March 21 in accordance with the low power physics testing program. The unit was synchronized with the grid at 11:31 A.M., March 22. Full power operation was achieved on March 27, 1986.

On March 28, 1986, an Auxiliary Operator making his routine morning plant tour discovered a feedwater/steam leak upstream of the B Main Feedwater Pump (MFP) suction relief valve. The leak was from a crack in the welded penetration for the one inch suction relief line and could not be isolated without securing the B MFP. Unit power was reduced to below 50% and the B MFP was taken out of service. Station maintenance personnel made the necessary repairs and the B MFP was returned to operation by 7:30 P.M., March 28. The unit was returned to 100% power by 4:25 A.M., March 29, 1986. The inspectors witnessed portions of the load reduction and maintenance activities. No violations were identified.

b. During the inspection, accessible plant areas were toured. Items reviewed include radiation protection and contamination controls, plant housekeeping, fire protection, equipment tagging, personnel safety, and security.

The inspector conducted frequent tours of containment to review preparations for the return to power operations and the general clean-up activities. Containment integrity requirements were reviewed and determined satisfactory. No discrepancies were noted.

On March 10, 1986, the inspector observed ALARA Committee Meeting 86-11. The purpose of the meeting was to review work to be performed on letdown line valves 311C and 311F inside the regenerative heat exchanger locked high radiation area. The inspector observed that the committee members attending the meeting exceeded minimum quorum requirements. Additional shielding, prefabrication work performed outside the high radiation area, and alternative welding techniques to minimize the nondestructive examination requirements were reviewed by the committee and implemented to reduce the workers' man-rem exposure. The inspector determined that of the total estimated 14.6 man-rem (34 man-hours) to complete the job, including shielding installation and removal, only 3.91 man-rem were expended. The significantly reduced exposure was attributed to lower dose rates than expected after shielding was installed.

The inspector observed numerous Plant Operation Review Committee (PORC) meetings during this inspection period. Of the meetings attended, the purposes were to review plant conditions and readiness to return to normal power operations. Of primary interest to the PORC members was the review and acceptance of completed station modifications. The inspector observed that PORC review of modification turnover packages was adequate.

No violations were identified.

c. Inspector tours of the control room this inspection period included reviews of shift manning, operating logs and records, equipment and monitoring instrumentation status.

The new Plant Process Computer System (PPCS) was not fully operational prior to plant start-up on March 21, 1986. The incore flux mapping capabilities were made functional for the physics testing program. Completion of the PPCS modification is expected later this year. The inspectors verified that Control Room operators were taking appropriate compensatory action for the plant computer being out of service.

d. Safety system valves and electrical breakers were verified to be in the position or condition required for the applicable plant mode as specified by Technical Specifications and plant lineup procedures. This verification included routine control board indication review and conduct of a partial systems lineup check of the Safety Injection Accumulators on March 18, 1986, and the Standby Auxiliary Feedwater System and the 1A Motor Driven Feedwater Pump on March 24, 1986.

No unacceptable conditions were noted.

4. Surveillance Testing

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

b. The inspector witnessed the performance of a portion of the following tests:

Refueling Shutdown Surveillance Procedure (RSSP)-7.0, "Control Rod Drop Test", Revision 9, dated 5/2/85, performed on March 14, 1986.

Periodic Test (PT)-32, "Reactor Trip Logic Test A or B Train", Revision 21, dated 4/4/86, performed on March 13, 1986.

Station Modification Procedure (SM)-3698.4, "Reactor Trip Bypass Breaker Modification Testing", Revision 0, dated 3/10/86, performed on March 13, 1986.

PT-17.4, "Control Room Radiation R-36, R-37, R-38 and Toxic Gas Monitor Operability Test", Revision 2, dated 2/11/86, performed on March 31, 1986.

PT-6.4, "Excore/Incore Recalibration ", Revision 12, dated 3/1/86, performed on March 25, 1986.

PT-7.0, "Hydro Test of Reactor Coolant System", Revision 32, dated 3/8/85, performed on March 18, 1986.

PT-2.3, "Safeguards Valve Operation", Revision 42, dated 1/16/86, performed on March 20, 1986.

PT-2.5.2, "Air Operated Valves, Quarterly Surveillance (Valves 112B and 112C)", Revision 8, dated 1/16/86, performed on March 20, 1986.

PT-3, "Containment Spray Pumps and NaOH Additive System", Revision 39, dated 2/5/86, performed March 17, 1986 on the 1B containment spray pump.

No violations were identified.

5. Plant Maintenance

a. During the inspection period, the inspector observed maintenance and problem investigation activities to verify: compliance with regulatory requirements, including those stated in the Technical Specifications; compliance with administrative and maintenance procedures; required QA/QC involvement; proper use of safety tags; proper equipment alignment and use of jumpers; personnel qualifications; radiological controls for workers protection; and reportability as required by Technical Specifications.

b. The inspector witnessed portions of the following maintenance activities:

Repairs performed on the 1B containment spray pump in accordance with Maintenance Procedure (M)-11.14, "Inspection and Maintenance of Ingersoll-Rand Pumps", Revision 13, dated 12/20/85, observed March 13 and 14, 1986.

Repair of the suction relief valve (#4022) to the 1B Motor Driven Auxiliary Feedwater Pump in accordance with M-37.38.1, "Safety and Relief Valve Inspection and Maintenance for Valve No. 4022", Revision 6, dated 2/1/84, observed on March 24, 1986.

Calibration of the Analog Rod Position Indication System in accordance with Calibration Procedure (CP)-2, "Calibration and/or Maintenance of the Rod Position Indication System at Hot Shutdown", Revision 2, dated 2/24/85, observed on March 20, 1986.

Steam Generator crevice cleaning operations conducted in accordance with Operations Procedure (O)-10, "Crevice Cleaning", Revision 20, dated 3/26/86, observed on March 14, 1986.

Replacement of spare circuit breaker in position 7E on the 1A Motor Control Center in accordance with M-44.1, "Isolation and Restoration of Motor Control Center 1A", Revision 8, dated 7/24/895, observed on March 18, 1986.

No unacceptable conditions were identified.

6. Physics Testing Review - Cycle 16 Startup

The startup physics test program is conducted in accordance with Periodic Test Procedure, PT-34.0, "Startup Physics Test Program", Revision 13, dated March 7, 1986. This procedure specifies the initial conditions and prerequisites, outlines the steps of the testing sequence, references detailed test procedures, and contains the data collection sheets for the various tests. The inspector reviewed the test procedures and test results to ascertain that the startup testing was conducted in accordance with technically adequate procedures and as required by Technical Specifications. The inspector independently verified that the predicted values and acceptance criteria were obtained from the "Nuclear Design and Core Management of the R. E. Ginna Nuclear Reactor, Cycle 16", Westinghouse WCAP-11069, dated March 1986.

The inspector reviewed selected tests to verify the following:

-- the test was implemented in accordance with Cycle 16 Startup Physics Test Program;

-- the test procedures were adequate and provide precautions, limitations and acceptance criteria in conformance with Technical Specifications;

-- appropriate measures were taken if a test condition or result did not meet acceptance limits;

-- testing methods and calculations were clearly specified and adhered to; and,

-- procedure review, approval and documentation of results were in accordance with Technical Specifications and licensee administrative controls.

a. Critical Boron Measurements

The licensee determines critical boron concentrations or boron end points in accordance with test procedures PT-34.1, "Initial Criticality, and ARO Boron Concentration", and PT-34.4, "RCC Bank Boron End Point Concentration". The inspector reviewed the data and noted the following results:

Configuration	Predicted Value (ppm)	Measured Value (ppm)
All Rods Out (ARO)	1454 ± 75	1464
D IN	1371 ± 75	1388
D + C IN	1269 ± 75	1285
D + C + B IN	1169 ± 75	1201
D + C + B + A IN	1012 ± 75	1026.5

The measured critical boron concentration for a given configuration is compared to the predicted concentration. The acceptance criterion of ± 75 ppm was met in all cases.

The bank boron end point concentrations are calculated values determined from the just critical boron concentration plus any minor bank height adjustments to maintain criticality. The bank height adjustments are converted to ppm boron based on differential bank rod worths obtained from the reactivity computer. With the assistance of the Reactor Engineer, the inspector verified these calculations were accurate. Clear documentation of these calculations and results was not evident in the completed procedures provided to the inspector for review.

No violations were identified.

b. Moderator Temperature Coefficient

The Moderator Temperature Coefficient (MTC) was calculated in accordance with test procedure PT-34.2, "Moderator Temperature Coefficient Measurement". The test procedure measures the Isothermal Temperature Coefficient (ITC) and the MTC is then calculated using Figure 5.1 of Westinghouse, WCAP-11069. The measured value of ITC was -1.07 pcm/degree F, at 547 degree F and ARO. The measured value of ITC was in good agreement with the predicted value of -1.00 pcm/degree F. ITC is defined as the change in reactivity per unit change in the moderator,

cladding and fuel temperatures. Using Figure 5.1, the MTC was interpolated to be +1.5 pcm/degree F. This value meets Technical Specification requirements that MTC be less than +5 pcm/degree F below 70 percent of rated thermal power. The inspector determined that the licensee did an additional calculation of MTC with a RCC bank configuration of D + C In. This configuration and boron concentration more closely approximate power conditions. The calculated MTC was -2.8 pcm/degree F. This value gives confidence that MTC at or above 70% rated thermal power will be zero or negative as required by Technical Specification 3.1.3.1.

The inspector noted that PT-34.2 does not clearly state how the MTC is derived by measuring the ITC. Additionally, the inspector determined that the supporting computer plots and calculations are not clearly labeled or documented. The Reactor Engineer stated that these discrepancies would be corrected.

No violations were identified.

c. <u>Control Rod Worth Measurement</u>

The control rod reactivity worth measurements were performed in accordance with test procedure PT-34.3, "RCC Bank Worth Measurement". The inspector reviewed the data and noted the following results:

RCC Bank	Predicted Worth (ppm)	Measured Worth (ppm)
D	836 ± 125 (15%)	723.7 (13.4%)
C	1030 ± 154.5 (15%)	940.5 (8.7%)
B	1021 ± 153.2 (15%)	763.1 (25.3%)
D + C + B	2887 ± 375.3 (13%)	2427.3 (15.9%)
A	1615 ± 242.3 (15%)	1578.2 (2.3%)
D + C + B + A	4502 ± 585.3 (13%)	4005.5 (11.03%)

The acceptance criteria for measured individual bank integral worth is ±15% of the predicted value. Additionally, the total measured integral bank worth must be within 13% of the predicted value. If the criteria for total bank worth is not met, additional banks will be measured until the total measured bank worth is within 13% of the predicted value. Boron end points shall also be compared with the integral rod worth for the particular bank or banks in question.

The inspector noted that all of the individual bank worth measurements were on the low end of the acceptance criteria. Control bank B was out of the acceptance criteria of ±15% by having a deviation from the predicted value of 25.3%. The resultant total deviation of the first three banks measured was 15.9%. The deviation was out of the acceptance range of 13% of the total predicted worth. Since the acceptance criteria was not satisfied, the licensee measured the worth of bank A as required by procedure. The additional bank A worth brought the total bank worth deviation to within the acceptance criteria of ±13%. Bank A deviation from predicted worth was 2.3% and the resultant total deviation for all measured bank worths was 11.03%.

This is the third consecutive fuel cycle that measured integral bank worths have been lower than predicted values. The inspector discussed the bank worth results with the licensee and determined that the licensee plans to meet again with Westinghouse representatives to further discuss the apparent computer software modeling problem with the control rod bank worth predictions. The licensee stated that adequate shutdown margin still exists due to the the total measured bank worth falling within acceptance criteria. In addition, boron end point measurements, which provide a more accurate measure of rod bank worth, were very close to the predicted values.

The inspector again noted minor discrepancies with the written test procedure and with clear documentation of the supporting calculations and data. The Reactor Engineer committed to make appropriate revisions to the procedure and more clearly record test data and calculations.

In addition to the previously identified test procedure deficiencies, the inspector observed that the individual signing for review of the test procedures was often the individual responsible for conducting the test and completing the test results review. The lack of an independent review of test procedures and results does not appear consistent with site procedural review practices. The inspector will review licensee resolution of this practice and PT-34 series procedural revisions in a subsequent inspection. (86-06-01)

No violations were identified.

7. <u>Containment Integrated Leakage Rate Test Review</u>

a. <u>Documents Reviewed</u>

-- Refueling Shutdown Surveillance Procedure (RSSP)-6.0, "Containment Integrated Leakage Rate Test", Revision 14, dated 3/1/86.

-- RSSP-6.1, "Integrated Leakage Rate Test Valving Alignment", Revision 12, dated 1/28/86.

-- RSSP-6.2, "Pressurization Monitoring of Penetration Free Volumes During CILRT", Revision 6, dated 5/31/85.

-- RSSP-6.3, "Air Supply for Integrated Leak Rate Test", Revision 8, dated 12/13/85.

-- RSSP-6.4, "Integrated Leak Rate Test Instrument Integrity Check", Revision 0, dated 2/5/86.

-- Calibration Procedure (CP)-50, "Containment Atmosphere Dewpoint Instrumentation Calibration", Revision 2, dated 8/13/85.

-- CP-52, "Calibration and/or Maintenance of the Containment Temperature Monitoring System", Revision 2, dated 9/24/85.

b. Scope of Review

The inspectors reviewed the Containment Integrated Leakage Rate Test (CILRT) procedures listed above for technical adequacy and to ascertain compliance with Technical Specification 4.4 and 10 CFR 50, Appendix J. The inspectors noted that test procedures were in general conformance with the guidance of ANSI/ANS 56.8-1981, "American National Standard Containment System Leakage Testing Requirements". The inspectors discussed various aspects of the CILRT with licensee representatives and contract personnel conducting the test. The inspectors also reviewed the test procedure initial conditions and prerequisites, pressurization equipment setup, monitoring instrumentation placement, data recording and processing methods, and test personnel training. No discrepancies were noted.

c. Data Collection and Leakage Rate Calculation

The licensee utilized a contractor, Quadrex, to perform the data collection and processing. Quadrex used redundant micro-computer systems for automated data acquisition and real time data processing. The micro-computer systems received data input from two precision pressure gages, 24 Resistance Temperature Detectors (RTD's) and six dew cells. The computer program calculated total time leakage rates, mass point leakage rates, Least Square Fit (LSF) leakage rates and 95% Upper Confidence Level (UCL) leakage rates. Test personnel were able to view computer displays of real time data, print accumulated data in table form and obtain graphic plots of both raw data and calculated leakage rates.

d. Test Boundaries

The inspectors reviewed a random sampling of piping penetration valve lineups from test procedure RSSP-6.1. This review was to ensure that systems were properly vented and drained to expose the containment isolation valves to containment atmosphere and to full differential pressure. No problems or unacceptable lineups were identified by the inspectors.

e. Instrumentation

The inspectors verified that the instrumentation specified in the test procedures satisfied the instrument selection guidance of ANSI/ANS 56.8-1981. Instrument calibration records were reviewed and found acceptable.

f. Results

The inspectors witnessed portions of each phase of the CILRT. Initial alignments were completed and pressurization commenced at 2:10 P.M., March 8, 1986. During pressurization, a drop in pressurizer level was observed. Operations and test personnel determined that the level decrease was due to the pressurizer being vented to containment atmosphere. As containment pressure was being increased, the voids in the top of the steam generator u-tubes were being compressed and primary water being displaced from the pressurizer. A pressurizer level increase was similarly observed during the depressurization phase.

Containment pressurization was completed and the eight hour stabilization period started at 11:00 P.M., March 8. Upon establishing stable containment temperature and pressure readings, the 24 hour leakage period was commenced at 6:00 P.M., March 9. The observed leakage rate was verified to be within the .75 Lt acceptance criteria within the first few hours of the test. The controlled leakage verification test was started at 7:00 P.M., March 10. The results of the controlled leakage test were satisfactory and the containment was depressurized in accordance with the test procedure.

No violations were identified.

8. Review of Periodic and Special Reports

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 reports contained the information required to be reported by NRC requirements; test results and/or supporting information were consistent with design predictions and performance specifications; and the validity of the reported information. Within this scope, the following reports were reviewed by the inspector:

-- Monthly Operating Report for February 1986.

-- Special Report - Fire Detection Systems Inoperable Greater Than 14 Days, dated March 31, 1986.

The licensee submitted this report in accordance with Technical Specification 3.14.1.1.c. due to two fire detection zones being out of service for greater than 14 days. Fire detection system S-07, Computer Room Floor, and Zone Z-17, Computer Room Ceiling, were removed from service to facilitate the installation of the new plant process computer system (PPCS). The new PPCS configuration also required the licensee to modify the fire systems in the computer room. The inspector verified that the required compensatory actions were taken while the fire detection systems were out of service and being modified. In addition, the inspector verified that the licensee submitted an Application for Amendment to the station Operating License (request dated February 12, 1986) reflecting the changes made to the fire protection systems.

No violations were identified.

9. Exit Interview

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

Based on the NRC Region I review of this report and discussion held with licensee representatives, it was determined that this report does not contain information subject to 10 CFR 2.790 restrictions. No draft written material was provided to the licensee during this inspection.