U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-244/86-11

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: June 1, 1986 through July 31, 1986

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Inspection Summary:
Inspection on June 1, 1986 through July 31, 1986 (Report No. 50-244/86-11)
Areas Inspected: Routine, onsite, regular, and backshift inspection by the resident inspectors (297 hours). Areas inspected included: plant operations; licensee action on previous findings; surveillance testing; maintenance; Licensed Operator Requalification Program; onsite review committee; Quality Assurance & Inservice Testing; Safety Injection Control Valve; potential CCW overpressurization; housekeeping; and inspection of accessible portions of the facility during plant tours.

Results: In the areas inspected, one licensee identified violation was observed. (Paragraph 7)

## 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 (75-21-07) Develop procedural guidance for a backup method of orderly plant cooldown systems. Licensee guidance for a natural circulation cooldown following loss of all AC power is contained in Emergency Procedure ECH 0.0, Revision 3, dated July 9, 1986, "Loss of All AC Power". The inspector reviewed this procedure and determined that the guidance to plant operations was sufficient in both scope and detail. The inspector also verified that the licensee has conducted training on implementation of this procedure. The inspector has no further concerns. This item is closed.

b. (Closed) Follow-up Item (78-10-05) Reactor Coolant System Decontamination. During a prior inspection, this item was opened pending completion of an NRC internal review of cleaning solutions and decontamination methods. Current NRC policy calls for evaluation of decontamination procedures on a plant-by-plant basis.

Ginna uses three types of decontamination for material used in the primary system. The first process is a two step procedure which involves using an alkaline permanganate soak to condition megnetite film and an oxalic acid-ammonium citrate final bath. This process is performed for the licensee by the London Nuclear Services, Inc. of New York. The second method of decontamination involves use of special decontamination paint provided by the Imperial Coatings Co. of New Orleans and the third method of decontamination involves liquid abrasive cleaning. This method is a mechanical cleaning process using a distilled water and aluminum grit slury sprayed under pressure on the material to be decontaminated.

All three of the decontamination methods used at Ginna are accepted standards in the nuclear industry. The inspector reviewed the licensee procedure for the decontamination of the steam generator on March 28, 1983 and noted that the Safety Analysis Report made an in-depth analysis of the effects of the decontamination process. The inspector identified no deficiencies with the licensees safety analysis nor with the licensee's decontamination procedures. This item is closed.

c. (Closed) Unresolved Item (78-17-01) This item was opened during an inspection of actions taken by the licensee in regards to I&E Circular 78-08, "Environmental Qualification of Safety-Related Components". The circular requested the licensee to review several concerns and take action on those that were applicable to Ginna station. At the time of this earlier inspection, only one concern, had been identified as being applicable to Ginna. On December 11, 1978, by letter dated the same date, notified the NRC that no further concerns were applicable. All actions required by the licensee are completed. This item is closed.

d. (Closed) Unresolved Item (78-17-04) Acceptability of non-qualified insulation sleeves. I&E Circular 78-08, "Environmental Qualification of Safety-Related Components", Item 11, identified concerns over the qualification of Monticello Electrical Splices, specifically, the heat shrink sleeving used for lug-type cable splices. The licensee committed in a letter to the NRC dated December 11, 1978, to replace all suspect sleeves with Raychem Thermofit Sleeving which was qualified to IEEE Standard 383. The inspector reviewed the procedures used and the documentation of the resplicing and resleevings. The inspector verified that all suspect sleeves were replaced. This item is closed.

e. (Closed) I&E Bulletin 78-BU-12, Atypical weld material in Reactor Pressure Vessel Welds. Bulletin 78-12 described the use of weld wire that failed to meet all chemical properties in welds of twelve identified reactor pressure vessels. Use of the atypical weld material in vessel weldments results in the reactor pressure vessel having higher than normal nil-ductility transition temperature characteristics which in turn requires more conservative pressure/temperature operating limits.

Bulletin 78-12 required the licensee to submit information on the Ginna pressure weldments as well as descriptions of the procedures used to verify conformance to specification. This information was submitted to the NRC for the licensee by the manufacturer of the Ginna pressure vessel, Babcock and Wilcox, on March 13, 1979. The Babcock and Wilcox report concluded that the Ginna reactor pressure vessel was in conformance with specification. This item is closed.

f. (Closed) Follow-up Item (81-22-57) On-site Emergency rescue team Guidance. During an earlier inspection of the licensee's emergency plans, improvements in specific and detailed instructions for establishing, the emergency rescue team communications, radiation protection requirements and interfaces with on-site medical treatment facilites were identified. These improvements were accepted by the licensee and incorporated into station procedure SC-233, "Search and Rescue Operations". The inspector reviewed

Revision 4 to the procedure dated October 3, 1984 and verified that the additional guidance satisfactorily addressed the identified concerns given as a result of the earlier inspection. This item is closed.

g. (Closed) Violation (82-12-01) Failure to implement adequate housekeeping controls. This violation resulted from an apparent laxity in overall housekeeping practices observed during an earlier inspection. The corrective actions taken in response to this violation included an immediate plant cleanup and a review of administrative controls on housekeeping. As a result of this review, improvements were made in the licensee surveillance program to insure housekeeping problems were identified before conditions deteriorated to an unacceptable condition. However, these improvements proved ineffective, in that three years later, in inspection report 50-244/85-21, it was noted that licensee housekeeping practices were still lacking.

In response, in part, to repeated inspector concerns over housekeeping, the licensee has instituted a new housekeeping policy (see paragraph 12 of this report). As a result there has been a significant improvement in station cleanliness as well as management and station personnel attitude towards it. Based upon this demonstrated improvement, this item is closed.

h. (Open) Follow-up Item (82-21-02) Residual Heat Removal (RHR). subbasement flood protection. During a prior inspection it was noted that when the drain line to the Auxiliary Building sump became plugged, water in the line would backup and spill into the RHR subbasement. This leakage path was noted to potentially reduce the overall flood protection for the RHR pumps.

The licensee has identified the source of the blockage to be a one inch pancake type orifice installed in the drain line. The licensee intends to change the orifice configuration to a nozzle type which will enable dirt and debris to be passed for later collection from the sump pit. This item will remain open pending installation of the new orifice.

i. (Open) Follow-up Item (83-03-01) Installation of suction gauges on safety-related pumps. During an inspection, it was noted that the suction pressure for the RHR, Safety Injection (SI), and the Containment Spray (CS) Pumps were calculated from the Refueling Water Storage Tank (RWST) level. ASME Boiler and Pressure Vessel Code, subsection IWP-4212, "Pressure Tap location", states, pressure taps shall be located in a section of a flow path that is expected to have reasonably stable flow as close as practical to the pumps. The inspector noted that in the testing of the RHR pumps the test path had the potential for affecting the pump suction pressure with the pump running.

The licensee obtained inservice suction pressure measurements using existing drain line taps for both the SI and CS pumps and determined that no significant differences existed between calculated RWST suction pressure valves and indicated gauge suction pressure. The licensee, therefore, does not intend to install a permanent suction pressure gauge for those pumps. For the RHR system, the licensee now utilizes a test pressure gauge connected to the pump suction drain line to obtain suction pressure during inservice testing, and has issued an Engineering Work Request to install permanent suction gauges for the RHR pumps for use during monthly and ISI surveillances. This item will stay open pending installation of the pressure gauges.

j. (Closed) Unresolved Item (83-17-01) Status and schedule of Inservice Inspection Program. During a prior inspection of the licensee's Inservice Insepction Program, an inspector was unable to verify whether the licensee had properly implemented the requirements of Technical Specification 4.2.1, Inservice Inspection Program, due to the lack of available documentation. The inspector noted that the licensee did not maintain a current inspection status or schedule. In addition, there was a discrepancy between the Ginna Station Quality Assurance Manual, Appendix B and Technical Specifications 4.2.1.1 and 4.2.1.2 in defining when the ten year inspection interval commenced for Group B and C components.

As a result of the concerns identified in this earlier inspection, the licensee has amended Technical Specifications (amendment 5) and revised Appendix B of the Quality Assurance Manual to bring both into agreement with regards to starting date of the ten year inspection intervals. In addition, the licensee has implemented a computerized tracking system for the Inservice Examination Plan which allows licensee personnel to determine inspection status as well as the inspection schedule for all components up until December 1, 2009.

The inspector reviewed a printout of the inservice inspection status as of the spring 1986 refueling outage and noted that the computerized system allowed easy determination of the inspection status of any component. The inspector had no further concerns. This item is closed.

k. (Closed) Violation (84-05-02) Failure to provide an adequate procedure to control inspection status. This violation was noted during an inspection of the licensee's Inservice Inspection Program. The inspector noted that the licensee had no formal method of insuring that plant modifications, applicable to the Inservice Inspection Program, would be added to that program, as required by 10 CFR 50, Appendix B, Criteria XIV.

As a result of this violation, the licensee created procedure QR&S 1004, "Incorporating Ginna Station Modifications into the Inservice Inspection Program." This inspector reviewed the guidance contained with the procedure and verified that it provided guidance requiring the review of all plant modifications for possible inclusion into the Inservice Inspection Program. This item is closed.

l. (Closed) Follow-up Item (84-14-07) Impact of collateral duty assignments on effectiveness of training program. During a prior inspection of the licensee training program, it was noted that expected long term corrective actions taken in response to that inspection would result in additional collateral duties being assigned to the training staff, and as a result, impact negatively on the training programs effectiveness.

In response to the inspector's concerns, the licensee reviewed the assignment of collateral duties within the training department and concluded that five additional instructors would be necessary to satisfactorily carry out the required long term corrective actions. The licensee has hired the additional personnel and based on interviews with the training staff, the inspector concluded that instructors collateral duties no longer interfer with their training assignment. This item is closed.

m. (Closed) Follow-up Item (84-28-01) Written tests for Emergency Preparedness Program. During a prior inspection of the Emergency Preparedness Program, it was noted that no written tests were given to determine if personnel had attained the level of knowledge necessary to perform the assigned emergency duties.

As a result of this finding, the licensee revised its emergency training program to include written exams to be given at the conclusion of each training session. The inspector reviewed the content of the exam series given by the licensee and verified that they adequately tested class participants on the information disseminated during the course. The inspector had no further questions. This item is closed.

n. (Closed) Follow-up Item (85-14-01) Analysis of steam generator and spent fuel pool water samples by the Brookhaven National Laboratory. During inspection 85-14, duplicate samples were drained from the B steam generator and the spent fuel pool to allow statistical evaluation of the licensee analysis process. The results of the licensee analysis were submitted to the Brookhaven National Laboratory where the duplicated sample was analyzed. Statistical evaluation of the two samples by Brookhaven indicated the licensee sample results were acceptable. The inspector reviewed the Brookhaven analysis and concurred with its evaluation. The inspector has no further questions. This item is closed.

o. (Closed) Follow-up Item (86-06-01) During the 1986 refueling outage an inspector noted deficiencies in Periodic Test Procedure (PT)-34.0, "Startup Physics Test Program", revision 13, dated March 7, 1986. Specifically, the inspector was concerned with lack of clear documentation of Critical Boron concentration calculations, lack of guidance for deriving the valve of the Moderator Temperature Coefficient and lack of independent review of the test procedure when completed.

In response to the inspector's concerns, the licensee has issued revision 4, to PT-34 effective May 21, 1986 which includes improved guidance for Boron and Moderator Temperature Coefficient calculations as well as the requirement for an independent review of the test results. The inspector has reviewed this procedure and verified the inclusion of the above concerns. This item is closed.

3. Review of Plant Operations

a. Throughout the reporting period, the inspectors reviewed routine power operations. The reactor operated at 100% power for much of the inspection period. The inspectors reviewed the following activities:

-- On June 26, 1986, at 0130 hours, reactor power was reduced to 48% following chemistry indication of a leaking condenser tube. The leak was identified by routine secondary chemistry analysis which indicated a significant increase in the sodium cation conductivity. Main condenser water box 1A1 was isolated and eddy current tube testing was conducted. As a result of these tests, one ruptured tube was plugged as well as four additional tubes with 50% throughwall crack indications. The cause of the tube damage was attributed to vibration of the tubes caused by a loose baffle plate. Following the corrective maintenance the unit was returned to 100% power at 2300 hours.

-- On July 29, 1986, at 0351 hours, the reactor was manually tripped from 100% power following a break in the drain line from the 2A Main Steam Reheater to the 5B Feedwater Heater. All safety systems responded properly to the reactor trip with the exception of the A Feedwater Regulating Valve (FRV) which failed to shut as required on high level in the A Steam Generator. As a result of the failure of the A FRV to shut, A Steam Generator level increased to 97% at which time a high-high steam generator level feedwater isolation signal was generated and responded to by the A FRV. Overfeeding of the A steam generator resulted in a pressurizer level shrink which eventually reached a minimum level of 3%. Letdown isolation actuated as required at 11% pressurizer level. Upon feedwater

isolation, pressurizer level returned to normal and the operators stablized the plant in the hot shutdown condition.

Investigation by the licensee determined that the drain line break was a result of erosion of the pipe metal at an elbow joint downstream of the 5B Feedwater Heater isolation valve. Condensate passing through the isolation valve was accelerated through the nozzle of the 3 inch isolation valve and impacted directly on the elbow, thereby, accelerating the erosion process. Visual inspection of the elbow showed no other erosion other than that in the area of the break. The licensee is currently evaluating plant systems for similar piping configuration and will examine identified elbows using ultrasonic testing to assure similar errosion problems are not present.

The failure of the A FRV to isolate on high steam generator level was traced to a mispositioned throttle valve on the bleed line of Solenoid S2. On high steam generator level, Solenoid valve S2 is designed to open and vent off the pressurized air controlling the position of the FRV, with the throttle valve controlling the rate of bleed. Post trip investigation determined that the throttle valve was only cracked open, thereby reducing the rate of FRV control air venting and the subsequent rate of FRV closing. The inspector will review licensee determination of the cause of the mispositioned throttle valve in a later report. (86-11-01)

Following a Plant Operations Review Committee (PORC) post-trip review and the performance of minor maintenance, the reactor was returned to criticality at 0338 hours, July 30, 1986 and synchronized with the grid at 1845 hours.

While escalating to full power at 1855 hours, the reactor tripped automatically at 25% power on Intermediate Range high flux. Post-trip investigation by the licensee determined that the A reactor trip breaker Intermediate Range high flux trip blocking relays had failed to energize as designed. The blocking relays are manually energized to block the IR trip when 2 or more power range nuclear instruments indicate greater than 10% power. When energized during the startup the blocking relays for Reactor Protection train "A" appeared to fail. The startup was halted until troubleshooting was completed. During the troubleshooting process. The blocking signal indicating lamp became energized, the licensee then incorrectly verified visually that the relays were energized and continued with the startup. When reactor power was increased above 25%, the reactor tripped as designed by the opening of the A reactor trip breaker. Following the trip, the licensee removed the relays and during bench testing, determined that the relay contacts failed to shut when the relay was energized. The faulty relays

were replaced and following reactor trip system testing, and a PORC post trip review, the reactor was brought critical at 0313 hours on July 30 and resynchronized with the grid at 0805 hours.

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.

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

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 systems lineup check of the Emergency Diesel Generators on June 11, 1986, and Service Water System on July 16, 1986.

No violations were identified.

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. During this inspection period, the inspectors witnessed the performance of a portion of the following tests:

Periodic Test (PT)-2.1, "Safety Injection System Test", Revision 4.0, dated February 2, 1986, performed June 16, 1986.

PT-2.2, "Residual Heat Removal System", Revision 3.7, dated November 16, 1985, performed June 20, 1986.

PT-2.7, "Service Water System", Revision 30, dated February 11, 1986, performed June 25, 1986.

PT-3, "Containment Spray Pumps and NaOH Additive Systems", Revision 42, dated April 4, 1986, peformed June 12, 1986.

PT-5.10, "Process Instrumentation Reactor Protection Channel Trip Test", Revision 29, dated March 15, 1986, performed July 15, 1986.

PT-16, "Auxiliary Feedwater System", Revision 43, dated November 1, 1985, performed June 12, 1986.

O-6.3, "Maximum Unit Power (Power Level Calorimetric)", Revision 15, dated April 11, 1986, performed July 15, 1986.

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; and reportability as required by Technical Specifications.

b. The inspector witnessed a portion of the following maintenance activities:

Calibration Procedure (CP)-15, Calibration and/or Maintenance of reactor vessel level measurement system loop A and loop B.

CP-18, Calibration and/or Maintenance of reactor vessel level measurement system loop A and loop B.

No violations were identified.

6. Containment Entries to Repair Sample Valve

On July 1, 1986, while setting up to perform a routine sample of the Reactor Coolant System (RCS), plant chemistry personnel noted difficulty in opening valve 955, Primary Coolant Sample Isolation. After numerous tries at opening the valve, the valve was finally opened and the required primary sample was drawn, however, while securing the valve lineup, valve 955 appeared to become stuck in an intermediate position. Subsequently, the plant operations staff noted an increase in unidentified RCS leakage, and a containment entry was made to inspect valve 955, which was suspected as the cause of the increased leakage. Plant personnel noted that leakage was indeed evident on the packing of the valve and that the mechanical position indicator was misaligned.

On July 2 & 3, 1986, the licensee made preparations to repair valve 955. These actions included:

-- An ALARA committee meeting to discuss expected exposure and how to reduce it.

-- Training for team personnel on a valve similar to valve 955.

-- Pre-entry team meeting to insure all personnel were knowledgeable in their required tasks.

Upon completion of the pre-entry training and briefings, a second containment entry was made. The packing on valve 955 was tightened and the positional indicator was realigned. Total exposure was for personnel was estimated to be 3.8 mrem for 12 people. However, as a result of the pre-entry workups, the actual exposure was only 3 mrem.

No violations were identified.

7. Improper Installation of Charging Pump 1A Alternate Control Circuit Power Supply

On July 28, 1986, the licensee informed the inspector that the Appendix R modification to the 1A charging pump control circuit performed on March 12, 1986, was improperly installed. The modification, in accordance with 10 CFR Part 50, Appendix R, is designed to provide an alternate DC control system power supply as well as remote operability. The modification errors were discussed as a result of concern raised by the licensee Quality Assurance department following a document review of the modification package.

The concerns identified by the licensee Quality Assurance department included inadequate and inaccurate working blueprints and inadequate installation, test and inspection procedures. Although the modification was installed incorrectly, the installation did not affect the normal operability of the charging pump.

Following identification of modification errors, the licensee developed a procedure to remove the improper installation and install, test and inspect the correct alternate power supply configuration. The procedure was approved on July 29, 1986 and installation and testing were completed on July 30, 1986.

As a result, the licensee Quality Assurance department commenced a review of all Appendix R modifications performed during the spring 1986 outage to verify proper installation. This item is a self-identified violation of 10 CFR Part 50, Appendix B, Criteria V; because the NRC wants to encourage and support licensee initiative for self-identification and correction of problems, the NRC will not generally issue a notice of violation for a violation that meets all of the following tests:

(1) it was identified by the licensee
(2) it fits in Severity level IV or V
(3) it was reported, if required
(4) it was corrected within a reasonable time
(5) it was not a violation that could reasonably be expected to have been prevented by the licensee's corrective action for a previous identified violation.

The above mentioned item meets these tests.

The inspector will follow up on licensee final resolution and corrective actions in a subsequent report (86-11-02)

8. Service Water Leak Inside Containment

During the performance of Periodic Test (PT)-2.2, "Service Water System" on July 29, 1986, the control room operator observed a significant decrease in the containment vessel 'A' sump pump actuation interval. An investigation by the licensee personnel was immediately initiated to determine the cause of the leakage inside containment. A Reactor Coolant System inventory balance and containment radiation dewpoint and recirculation fan cooler condensate collection systems check indicated no change. The leakage rate to sump 'A' was calculated to be approximately four to five gallons per minute and a service water leak became suspect.

At the time, the reactor was in hot standby following the trip detailed in paragraph 3. Based upon high service water flow indication, fan cooler unit D was isolated and a containment entry was made. Personnel entering containment identified the leak as coming from one of the 1/8 inch diameter threaded motor cooler drain plugs on the B recirculation fan cooler nut. The B fan cooler unit was promptly isolated and unit D returned to service.

The displaced motor cooler drain plug was replaced and the integrity of all other recirculation fan cooler drain plugs was visually verified. The B recirculating fan cooler was returned to service as required prior to the reactor startup on July 30.

Two similar events of this nature have occurred, one on February 11, 1981 as reported in Licensee Event Report No. 81-004 and again on July 1, 1985 as submitted as an enclosure to a RG&E letter to the NRC dated July 12, 1985. In all three events, it was determined that the 1/8 inch drain plugs blew out as a result of weakening due to corrosion. The drain plugs are made of carbon steel and serve as cathodic protection for the recirculation fan cooler units. Subsequent to the 1985 event, annual containment recirculation fan cooler inspections included a visual as well as physical check of plug integrity. It is apparent that these actions are not sufficient in preventing reoccurrences.

The inspector will follow-up on licensee corrective action following the submission of the leak report issued in IEB 80-24. (86-11-03)

9. ECCS Safety Injection Minimum Recirculation Flow Control Valve

IE Bulletin 86-01, issued May 23, 1986, informed all Boiling Water Reactor (BWR) licensees and BWR construction permit holders of a recently identified problem at Pilgrim in which a single failure in the residual heat removal (RHR) system minimum flow control logic could potentially disable all of the RHR pumps. A similar concern may also exist at a number of Westinghouse Pressurized Water Reactors (PWRs) involving the potential failure in the minimum recirculation flow protection configuration. IE Information Notice 85-94, "Potential for Loss of Minimum Flow Paths Leading to ECCS Pump Damage During a LOCA", also informed licensees of this potential problem.

The inspector discussed these information items with the licensee and reviewed the station safety injection (SI) pump recirculation flow configuration. Two in-line air operated recirculation flow control valves are used in the common minimum flow return line to the refueling water storage tank (RWST). These two flow control valves are interlocked with the containment sump suction isolation valves to prevent contamination of the RWST when operating in the recirculation mode following a LOCA. These valves fail closed on a loss of air supply.

The licensee has concluded that if the SI pumps were running against shut off head (small break LOCA) and concurrent loss of control air was experienced, potential pump damage may result due to inadequate recirculation cooling flow. The licensee has fabricated mechanical blocking devices for each of the valves to prevent the valves from closing and has drafted procedural controls for their use. The inspector will review the installation of the blocking devices and final resolution in a subsequent report. (86-11-04)

10. Potential Component Cooling Water Overpressurization

By letter dated July 19, 1984, the licensee was notified by Westinghouse of the potential for overpressurization of the component cooling water system. The overpressurization is possible via leakage into the Component Cooling Water (CCW) system from the higher pressure Reactor Coolant System (RCS) and the subsequent automatic closure of the CCW surge tank vent upon detection of RCS in-leakage by the in-line radiation monitor.

The licensee determined via an engineering analysis that the existing CCW system design is not susceptible to overpressurization. The inspector reviewed the licensee's evaluation; no deficiencies were identified. The most limiting system component was found to be the CCW

pump seals with a hydrostatic test pressure limit of approximately 500 psig. Under the most conservative estimates, the maximum system pressurization was calculated to be approximately 229 psig. This pressure was limited by maximum CCW pump discharge pressure, maximum system deviation head, and highest possible relief valve pressure setting.

The inspector had no further questions.

11. Requalification of Licensed Operators

A review of the requalification program for licensed operators was conducted to verify the program's conformance to the requirements of 10 CFR 55, Appendix A. Requalification training conducted in the plant specific simulator was observed. Discussions with the training staff were conducted and the following training documents were reviewed:

-- Procedure No. A-103.4, Operator Requalification Program

-- Long range requalification training schedule

-- Lesson plans

-- Student handouts

-- 1986 Cycle Requalification Examinations

-- An Operator's requalification history

-- Completed emergency procedure review assignment forms

-- Requalification lecture attendance records

-- Requalification examination grading summary sheet

-- Documentation of individual cross training, control manipulations, plant evolutions, and discussions

The inspector evaluated the simulator training scenario as well planned and challenging. The discussion lead by the simulator instructor after the scenario provided meaningful training to the operators.

INPO accreditation for training programs will not be completed until 1987. The initial self-evaluation for three nonlicensed training programs were submitted to INPO the last week in June. The submittal of the self-evaluation for all training programs is planned for August 1986. An eight person task force has been assigned to develop the programs for INPO accreditation which is considered adequate.

A long range training schedule has been prepared for the year, however, a detailed schedule of specific training topics is not prepared until two weeks before the beginning of a cycle. The length of quality of the segmented examinations have improved over the year and are adequate to evaluate operator's knowledge levels.

No violations or deviations were identified. The requalification training program is adequate to provide meaningful training to the operators and is improving due to the use of the plant specific simulator and the implementation of a training program based on learning objectives developed from a job task analysis.

12. Housekeeping

On June 16, 1986, licensee plant management implemented a maintenance organization change whereby individual personnel were given responsibility for building and grounds maintenance, in part, including housekeeping. (See paragraph 2, item g.)

Plant management agrees that poor housekeeping could be perceived as a lack of management support to quality control. Under the new program, the duty engineer, and the fire protection and safety coordinator are required to tour the plant weekly and identify items of housekeeping concern. Identified items will be tracked under the Quality Control Audit Program.

Although this policy has been in effect for only six weeks and is still waiting corporate level approval to make it permanent, there already has been a significant improvement in the cleanliness of the plant as well as personnel attitude towards good housekeeping. The inspector will follow-up on effectiveness of this program by reviewing plant cleanliness on daily tours.

13. Onsite Review Committee

The inspector observed the conduct of the Plant Operations Review Committee (PORC) meetings Nos. 86-086, 86-092, 86-094 held on July 16, 29 and 30, 1986. The inspector attended the meeting as a nonparticipant to observe the general conduct of the meeting and to verify the provisions of Technical Specifications regarding the PORC were satisfied and to determine the depth of the licensee's post trip review process. A subsequent review of the meeting minutes was conducted to confirm that the decisions and recommendations of the Committee were properly documented and acted upon.

No discrepancies were noted.

14. 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 report was reviewed by the inspectors:

-- Monthly Operating Report for May 1986.

15. 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.