U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Report No. 50-244/85-26

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: December 1, 1985 through December 31, 1985

Inspector: W. A. Cook, Resident Inspector, Ginna

Reviewed by: J. R. Stair, Reactor Engineer, Reactor Project Sect. No. 2C, DRP — 1-13-86 Date

Approved by: J. C. Linville, Chief, Reactor Project Section No. 2C, DRP — 1/13/86 Date

Inspection Summary:

Inspection on December 1, 1985 through December 31, 1985 (Report No. 50-244/85-26)

Areas Inspected: Routine, onsite, regular, and backshift inspection by the resident inspector (89 hours). Areas inspected included: plant activities during routine power operations; licensee action on previous findings; surveillance testing; maintenance; IE Circular No. 80-15 follow-up; LER review; and inspection of accessible portions of the facility during plant tours.

Results: In the seven areas inspected, one violation was identified. A violation of administrative controls for maintenance-related activities is discussed in paragraph 5. A follow-up review of licensee response to IE Circular 80-15 is discussed in paragraph 6.

DETAILS

1. Persons Contacted

During this inspection period, the inspector interviewed and talked with operators, technicians, engineering and supervisory level personnel.

2. Licensee Action on Previous Inspection Findings

(Closed) Inspector Follow-up Item (79-SP-03) In 1978, the operation of containment purge and venting systems became a generic concern. The major issue is the operability of the large, butterfly-type purge valves in the event of a design basis loss of coolant accident. The licensee's interim response to this generic concern was reviewed and found acceptable by the NRC staff as documented in the Safety Evaluation Report transmitted by the Crutchfield to Kober letter, dated June 21, 1984. The inspector verified that the licensee's commitment to maintain the 48 inch containment purge valves closed, while the reactor is critical, is being adhered to. In addition, the inspector verified that a licensee commitment to limit containment vent system operation to a total of 90 hours per year is being observed. No discrepancies were noted.

To permanently resolve the containment purge and vent systems issue, the licensee plans to modify the 48 inch purge system so that it will only be capable of being used when the reactor is in cold or refueling shutdown. The installation of a new mini-purge system will permit limited purging of containment while the reactor is critical. Modification work is tentatively scheduled for the 1986 Annual Maintenance and Refueling Outage.

(Closed) Inspector Follow-up Item (81-24-03) During an earlier inspection period, the performance of a lamp test on Fire Protection System Satellite Station A (SSA) caused the circuit breaker for the SSA to trip open. The lamp test was conducted as part of a functional test of SSA following modifications to several associated fire suppression spray/sprinkler systems. It was determined that during steady state conditions SSA current readings are approximately 2.4 amps. The SSA circuit breaker is designed to trip at 2.5 amps. Functional testing indicated that an extended satellite station lamp test draws sufficient additional current flow to trip the SSA circuit breaker.

The inspector determined that a subsequent review by the licensee and supplier resulted in no significant system modifications to the satellite station design. A minor change was made to the SSA to reduce the total number of lamps being tested at one time, thereby reducing the total current being drawn. However, the licensee no longer performs a SSA lamp test. Instead, each module alarm lamp is tested periodically during the individual system surveillance test.

(Closed) Inspector Follow-up Item (79-SP-02) This item administratively tracks generic Unresolved Safety Issue (USI), A-47, "Safety Implications

of Control Systems". The purpose of reviewing USI, A-47 is to determine whether failures in control systems typically used during normal startup, shutdown and on-line power operations could significantly affect the safety of the plant. Final staff resolution of this unresolved safety issue is still pending. Any recommendations made by the NRC staff will be tracked and reviewed separately in a future report. This item is administratively closed.

(Closed) Inspector Follow-up Item (80-09-01) On July 29, 1980, the turbine control valve started to go full open for an undetermined cause. The control valves opening caused automatic control rod movement in response to lowering Reactor Coolant System (RCS) average temperature. Operators took manual control of the turbine and rod control system and stabilized the reactor plant. The licensee closely monitored turbine control valve operation and conducted troubleshooting of the Electrohydraulic Control (EHC) system to attempt to repeat the sequence of events. No apparent cause for the control valve perturbation could be identified.

In recent years, the licensee has identified a similar turbine EHC system characteristic. While operating close to full power with the EHC system in automatic, the number two turbine control valve is positioned approximately 40% open. At 40% open, the inherent throttling characteristics of the control valve result in valve hunting to maintain a constant impulse pressure. This control valve hunting results in small changes in RCS reference temperature and turbine load which ultimately results in automatic control rod movement to compensate. To dampen this effect the licensee routinely operates with the rod control system in manual and with the turbine control valve limiter reduced to approximately 40%.

3. Review of Plant Operations

a. Throughout the reporting period, the inspector reviewed routine plant operations. The reactor operated at full power for the duration of the report period. The licensee continues with preparations for the 1986 Refueling Outage scheduled to commence February 8, 1986.

b. During the inspection, accessible plant areas were toured. Items reviewed include radiation protection controls, plant housekeeping, fire protection, equipment tagging and security. On December 17, the inspectors reviewed the licensee's receipt inspection of new fuel assemblies in the Auxiliary Building. No discrepancies were noted.

c. Inspector tours of the control room this inspection period included review of shift manning, operating logs and records, and 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 partial systems lineup check of the Auxiliary Feedwater Systems on December 20, 1985.

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:

Periodic Test (PT)-3, "Containment Spray Pumps and NAOH Additive System", Revision 38, dated 10/30/85, performed on December 23,1985.

PT-12.1, "Emergency Diesel Generator 1A", Revision 22, dated 12/21/85, performed on December 6, 1985.

PT-5.30, "Process Instrumentation Reactor Protection Channel Trip Test (Channel 3 Blue)", Revision 32, dated 12/03/85, performed on December 19, 1985.

During the performance of PT-5.30, when pressurizer pressure controller PC-431k was transferred from manual to auto, both pressurizer spray valves went full open. I&C technicians had finished defeating the channel and were having licensed operators complete the switch line-up at the control board when this occurred. The operators immediately returned pressure control to manual and took manual control of both spray valves to close them. Primary plant pressure dropped to approximately 2100 psig during the ensuing pressure transient. Subsequent troubleshooting indicated the cause of the malfunction was dirty contacts in pressure controller PC-431k. The problem could not be duplicated after several switching operations. The licensee plans to replace pressure controller PC-431k during the 1986 Refueling Outage. In the interim, pressurizer spray valves are placed in manual, in the closed position, prior to switching PC-431k from manual to auto or vice versa.

5. Radiological Effluent Sampler Disconnected

On December 3, 1985, an Instrumentation and Control (I&C) technician installed a new in-line chlorine monitor in the discharge canal sample line. Chlorine monitoring is part of the Station's State Pollutant Discharge Elimination System, a New York State environmental pollution control requirement. The new monitor, (Orion Industrial, Model 1770 Chlorine Monitor), replaced an existing chlorine monitor which was no longer in service. In addition to the chlorine monitor, an in-line pH meter and a Technical Specification required radiological effluent composite sampler are connected to the same sample line. The new monitor installation required some modification to the existing sample piping due to the size and

location of the monitor. Upon completion of the piping reroute, the I&C technician restored flow through the pH meter, located upstream of the chlorine monitor, and established flow through the new monitor. Not recognizing the composite sampler as a Technical Specification required component, the technician had secured flow to the sampler and rerouted sample line flow directly to the floor drain located below the chlorine monitor.

Flow remained isolated to the composite sampler until the condition was identified by the Radiochemist on December 16, 1985. The sample line piping was rerouted and flow to the composite sampler was restored later that day. The licensee plans to report this event in accordance with Technical Specification 3.16.1.2 in the Annual Radiological Environmental Operating Report.

The inspector determined that neither the I&C technician involved nor his supervisors were aware of the discharge canal sample line radiological effluent monitoring requirements or associated equipment. In addition, no Maintenance Work Request and Trouble Report (MWR) or equivalent procedural guidance was initiated to control the installation of the chlorine monitor. An informal request between the Chemistry and Maintenance groups was made to install the monitor. Consequently, routine station review and approval channels were bypassed, in that, the new chlorine monitor was installed without the appropriate station reviews and procedural controls specified by Administrative Procedure (A)-1603, "Maintenance Work Request and Trouble Report". This is a violation. (85-26-01)

6. Follow-up on IE Circular No. 80-15

IE Circular No. 80-15 contained information on the June 11, 1980 event at St. Lucie Unit I involving a total loss of component cooling water (CCW) flow. During the subsequent natural circulation cooldown, reactor coolant system pressure and pressurizer level response indicated the formation of steam voids in the reactor vessel. This event was significant for several reasons. It demonstrated that, under seemingly normal natural circulation cooldown conditions, steam voids can form which are large enough to cause significant and rapid variations in pressurizer level. It highlighted the vulnerability of the reactor coolant pumps to a total loss of component cooling water flow due to a single failure of CCW containment isolation valves. In addition, it was an example of inadequate system alignment controls. An unanticipated discharge of reactor coolant directly to the refueling water tank resulted from an improperly positioned valve in the low pressure safety injection system.

The inspector reviewed licensee response to NRC staff recommendations promulgated in IE Circular No. 80-15. Licensee records and procedures were reviewed, station personnel were interviewed and facility instrumentation and control systems were observed.

Information contained in the IE Circular appears to have been well disseminated. Licensed operators interviewed were cognizant of the potential for steam void formation in the reactor vessel head during natural

circulation cooldown and were knowledgeable of the instrumentation available and procedural requirements for proper control of a natural circulation cooldown. The inspector reviewed applicable emergency and operating procedures and verified appropriate cautions to protect against anomalous conditions were included and that specific recovery actions/procedures were incorporated. The inspector observed a demonstration by the licensee of how the newly revised Emergency Operating Procedures would step the control room operators through a scenario similar to the St. Lucie Unit I event.

The station component cooling water system design was reviewed and documented in NUREG-0821, Integrated Plant Safety Assessment, Systematic Evaluation Program (SEP). (reference: SEP Review Topic IX-3, "Station Service and Cooling Water Systems"). In addition, reactor coolant pump (RCP) seal integrity was reviewed under NUREG-0737, "Clarification of TMI Action Plan Requirements", topic II.K.3.25. (references: NRR to RG&E letter, dated July 2, 1982, and NRC Inspection Report No. 50-244/84-19). The invulnerability of the component cooling water system to single failures, which could result in either a loss of RCP cooling or loss of RCP seal integrity, was adequately demonstrated.

During the 1984 Annual Maintenance and Refueling Outage, the licensee replaced the existing core exit thermocouple (CET) system with an upgraded system which meets NUREG-0737 and Regulatory Guide 1.97, Revision 3 requirements. The new CET system display consoles, cabling, containment penetration and connectors are seismically and environmentally qualified. The system is split into two trains with separate digital scanning displays in the control room. The displays provide isolated outputs to the plant computer for normal operation and safety assessment. Three of the 39 CET's are positioned to measure temperatures in the reactor vessel head area to enhance post accident monitoring capability.

7. Licensee Event Reports (LERs)

The inspector reviewed the following LERs to verify that the details of the event were clearly reported, the description of the cause was accurate, and adequate corrective action was taken. The inspector also determined whether further information was required, and whether generic implications were involved. The inspector further verified that the reporting requirements of Technical Specifications and station administrative and operating procedures had been met; that the event was reviewed by the Plant Operations Review Committee and that continued operation of the facility was conducted within the Technical Specification limits.

85-18: "Manual Turbine and Reactor Trip Due to EHC System Problems". On September 28, 1985, the turbine and reactor were manually tripped from approximately 50% power. A service water leak in the turbine Electro-Hydraulic Control (EHC) System severely degraded the operability of the turbine intercept, stop and control valves causing the operators to take the unit off the line. This event was reviewed and documented in Inspection Report 50-244/85-21, paragraph 3.a.

85-19: "Automatic Reactor Trip Due to Trip of the 1B Condenser Circulating Water Pump". On November 25, 1985, the reactor tripped automatically due to a coincident steam flow/feed flow mismatch and low B steam generator level (30%). The cause of the automatic reactor trip was a trip of the 1B circulating water pump which resulted in a partial loss of turbine condenser heat removal capability. Operators were unable to control the subsequent condensate and feedwater transient which resulted in insufficient feedwater flow to the steam generators. This event was reviewed and documented in Inspection Report 50-244/85-24, paragraph 3.a. The licensee's long-term corrective actions to address the secondary plant systems response and operator actions to mitigate similar plant transients will be reviewed by the inspector in a future report. (85-26-02)

8. Review of Periodic and Special Reports

Upon receipt, periodic and special reports submitted by the licensee pursuant to Technical Specifications 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 reported information was valid. Within this scope, the following report was reviewed by the inspector:

-- Monthly Operating Report for November 1985.

9. Exit Interview

At periodic intervals during the inspection, 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.