U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Inspection Report 50-244/95-17 License: DPR-18

Facility: R. E. Ginna Nuclear Power Plant
Rochester Gas and Electric Corporation (RG&E)

Inspection: September 10, 1995 through October 21, 1995

Inspectors: P. D. Drysdale, Senior Resident Inspector, Ginna
E. C. Knutson, Resident Inspector, Ginna

Approved by: Lawrence T. Doerflein 11/20/95
Lawrence T. Doerflein, Chief Date
Reactor Projects Branch 1
Division of Reactor Projects

<u>Inspection Summary</u>:

Core, regional initiative, and reactive inspections performed by the resident inspectors during plant activities are documented in the areas of plant operations, maintenance, engineering, and plant support.

<u>Results</u>:

See Executive Summary.

EXECUTIVE SUMMARY

R. E. Ginna Nuclear Power Plant

Inspection Report No. 50-244/95-17

Operations: The plant operated at full power (approximately 97 percent) throughout the inspection period. For most of the current operating cycle, the plant has been operating with one PORV block valve closed due to minor PORV seat leakage. During this inspection period, seat leakage developed from the other PORV. An attempt to reseat the valve was not successful. The licensee plans to operate with both PORV block valves closed until the next refueling outage. Valve modifications to eliminate seat leakage are under review.

The licensee's followup actions to the loss of instrument air (IA) system event, occurring in June 1995, were thorough. The cause of the failure could not be definitely established; however, the licensee determined that inadequate insertion of the piping into the union was a contributing factor. Non-destructive examination of other soldered joints in the IA system indicated that inadequate piping insertion is apparently not a wide-spread problem. The operations and training departments conducted an in-depth critique of the event and thoroughly developed lessons learned. Subsequent revisions to the abnormal operating procedures were well validated on the plant simulator and now provide appropriate guidance for operation at power.

Maintenance: During a surveillance test of the turbine driven auxiliary feedwater pump, the B-steam generator flow control valve produced several spike increases in flow. Operators subsequently took control of the valve and cycled it more fully open. When returned to automatic control, no further valve oscillations were experienced and the test was completed. This problem was not entered into the corrective action system until the inspector discussed the event with licensee management. The cause of the oscillations was subsequently determined to be a defective valve position controller. The anomalous condition noted during testing of a safety system was not adequately evaluated by the licensee. Additionally, operations department intervention during testing, to attempt to correct an anomalous condition, was not specified in the test procedure.

After the C-service water (SW) pump was started in preparation for a routine surveillance test, a Results and Test (R&T) technician noted an acrid odor in the vicinity of the C-SW pump. Subsequent troubleshooting identified a turn-to-turn short circuit in the motor windings. Early identification of winding insulation breakdown in the C-SW pump motor was noteworthy. The R&T technician showed a high level of alertness and safety awareness by noticing and reporting the abnormal odor to control room operators. In pursuing this indication, the licensee averted an actual motor failure and the resultant challenge to one of the class 1E electrical busses. The licensee is examining options to improve long term reliability of the SW pump motors.

The licensee continued their efforts to quantify and determine the source of water leakage into the residual heat removal system pump room. Chemical and isotopic analyses of the water leaking into the room are consistent with the makeup of spent fuel pool (SFP) water, except that the concentration of cesium in the leakage water is higher than in the SFP. This may be due to a concentrating mechanism (drying and redissolving). The amount of water leakage into the room continues to be small, however, the rate of water loss from the SFP has not been quantified. A Technical Staff Request was issued to obtain engineering support in determining the evaporation rate and the total rate of water loss from the SFP.

The licensee has experienced several mechanical problems with the A-charging pump varidrive speed controller. The root cause analysis determined that the original problem stemmed from maintenance problems that occurred during a three-year overhaul that was performed during the 1995 refueling outage. Following a second overhaul, the varidrive experienced problems during speed reductions, producing smoke from the V-belts. The root cause of these problems were again determined to be maintenance-related. Corrective actions were under evaluation at the close of the inspection period.

<u>Engineering</u>: RG&E's project to replace both steam generators (SGs) during the next refueling outage is currently on schedule and preliminary construction activities are in progress. The major construction activities now completed or currently in progress are 1) two foundation pads are in place adjacent to the containment to support the Lampson crane that will be used to lift the SGs into and out of the containment; 2) A full-scale partial containment dome mock-up was completed and will be used for personnel training and to confirm construction and restoration methodologies; 3) installation of an equipment support and work platform around the containment dome is in progress; 4) modification of the containment facade structure is in progress to support a material storage enclosure and personnel comfort station adjacent to the containment dome work area; 5) construction of a containment access facility is in progress; 6) reinforcement of a portion of the underground 34.5 kilovolt power line duct bank and parking lot storm drains is ongoing; 7) construction of a new temporary storage facility for the old SGs is essentially complete; and 8) dredging of the barge slip at Bear Creek harbor is about 50% complete.

<u>Plant Support</u>:

Several recent incidents occurred where personnel entered a radiologically controlled area (RCA) to perform work in the plant, but did not log into the radiation work permit (RWP) access control system, and/or did not have the secondary dosimetry required for entry. These incidents represented a failure of radiation workers to adhere to procedures pertaining to RCA access. The licensee initiated a Human Performance Evaluation System (HPES) review of these incidents. The report identified several "passive barriers" that contributed to a breakdown in radiation worker adherence to RCA access requirements. The licensee concluded that an "active barrier" was necessary. A locking turnstile barrier was installed at the RCA access point. The HPES also recommended ongoing reviews for improper RCA access. These reviews revealed several additional instances of improper RCA access during October 1995 that involved radiological protection technicians. The licensee's efforts to correct the conditions associated with this problem are still ongoing.

Safety Assessment/Quality Verification:

The inspector attended numerous PORC meetings. All meetings were formally conducted under the direction of the PORC chairman. Each Committee member performed independently and concerns raised by individual members prompted further evaluation by the full Committee. The PORC members considered the revised agenda format and review process to be more efficient and allowed them to focus more effectively with plant safety concerns.

The inspector attended a quarterly NSARB PORC Sub-committee meeting during the inspection period. The Sub-committee reviewed all PORC business for completeness and accuracy, and separately evaluated all PORC decisions. The Sub-committee members also compared all PORC decisions against the licensee's existing NRC commitments. The Sub-committee fully reviewed their agenda items and effectively evaluated PORC activities for the current quarter.

The inspector identified some events which were not being evaluated by the licensee's corrective action process. The licensee's audits also identified inconsistencies between departments with the threshold for writing Action Reports. The licensee revised the governing procedure for the corrective action program to improve the reporting and tracking process for Action Reports.

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DETAILS

1.0 SUMMARY OF FACILITY ACTIVITIES

The plant operated at full power (approximately 97 percent) throughout the inspection period. There were no significant operational events or challenges during the inspection period.

2.0 PLANT OPERATIONS (INSPECTION PROCEDURE (IPs) 71707)[1]

The inspectors observed plant operation and verified that the facility was operated safely and in accordance with licensee procedures and regulatory requirements. This review included tours of the accessible areas of the facility, verification of engineered safeguards features (ESF) system operability, verification of proper control room and shift staffing, verification the unit was operated in conformance with technical specifications and that appropriate action statements for out-of-service equipment were implemented, and verification that logs and records were accurate and identified equipment status or deficiencies.

2.1 Power Operated Relief Valve Leakage

On October 10, 1995, operators observed evidence that minor seat leakage had developed from power operated relief valve (PORV) PCV-430. Over a period of several hours, temperature of the downstream piping (normally constant at about 100 degrees Fahrenheit (°F) increased to the alarm setpoint of 145°F, and pressurizer relief tank (PRT) indications of pressure, temperature, and level increased. In response, operators closed the associated PORV block valve, which stopped the leakage to the PRT. The other PORV, PCV-431C, began leaking shortly after the 1995 refueling outage, and therefore, its block valve was previously closed. Although plant operation with both PORV block valves closed is not prohibited by technical specifications, it is not desirable because operator action is required to reinstate the automatic pressure reducing function of the PORVs.

On October 12, 1995, the licensee performed a leak rate test on PCV-430 by opening its associated block valve and measuring the change in PRT level. Following data collection, the block valve was reclosed. The PORV leakage rate was calculated to be approximately 0.01 gallons per minute. Later that day, the valve was cycled open and closed with the block valve closed, in an attempt to flush the seating surfaces and reseat the PORV. Testing the following day demonstrated that the leak rate had not been significantly affected. At the end of the inspection period, the licensee planned to continue to operate with both PORV block valves closed until the 1996 refueling outage. The licensee is also evaluating a possible modification of these valves (for example, the use of different materials for the seats and discs) based, in part, on vendor and industry experience with leaking PORVs.

The inspector observed control room operations during the PCV-430 leak tests. Licensee management was present in the control room and provided oversight

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[1]The NRC inspection manual procedure or temporary instruction that was used as inspection guidance is listed for each applicable report section.

during the tests. Key parameters were selected and trended on the plant computer, and operator attention was appropriately focused on these indications during the tests. The inspector concluded that the leak testing that was performed on PCV-430 was well controlled.

2.2 Instrument Air System Leak Followup

On June 7, 1995, a leak occurred in the instrument air (IA) system piping inside containment due to failure of a soldered pipe union. Details of this event were presented in inspection report 50-244/95-12. The leak resulted in a sustained loss of IA to primary system components inside containment, which presented operators with several significant operational challenges. Reactor coolant system (RCS) pressure control was complicated by loss of operability of the pressurizer spray valves and PORVs. RCS inventory control became a concern because letdown was isolated by fail-closed, air operated valves, but charging was still required to provide seal injection to the reactor coolant pumps. Operators averted a high pressurizer level reactor trip by performing a power reduction while a temporary repair of the IA leak was completed; however, procedural guidance for conducting this operation was limited.

The licensee conducted an in-depth review of the loss of IA event. The cause of the solder joint failure could not be definitively established because the affected portions were disturbed during efforts to perform a temporary repair of the leak. However, the licensee found the amount of pipe that had been inserted into the failed union was less than the minimum allowable length. To determine if similar conditions existed elsewhere in the IA system, the licensee performed ultrasonic inspections of randomly selected soldered piping joints outside of containment. All of the 21 joints that were inspected had satisfactory insertion. A similar examination of soldered joints inside containment will be performed during the 1996 refueling outage.

The licensee also examined the need to revise or enhance operating procedures based on experience gained during the event. Based on comments from a post-event critique conducted by the operations and training departments, and thorough review of the event by operations department personnel, abnormal procedures AP-IA.1, "Loss of Instrument Air," and AP-TURB.5, "Rapid Load Reduction," were revised. Changes to AP-IA.1 included:

- A caution was added noting that operability of the pressurizer spray valves and PORVs is affected by loss of IA to containment, and that this should be considered if load decrease is required.

- A step was added directing operation of pressurizer heaters as necessary to control pressurizer pressure; manual control of pressurizer heaters is necessary because the pressurizer spray valves are air operated and fail closed.

- A step was added to verify adequate cooling to the nuclear instrument detectors; temperature monitoring is required because the associated ventilation dampers are air operated and fail closed.

- A caution was added noting that it is permissible to operate reactor coolant pumps for limited periods without seal injection; this allows charging pumps to be secured if necessary to control pressurizer level.

Changes to AP-TURB.5 similarly addressed manual control of pressurizer heaters and securing coolant charging as necessary in the event that IA to containment is isolated. In addition, the procedure directs operators to trip the reactor if pressurizer pressure or level, or steam generator level, is approaching its trip setpoint and cannot be controlled. These procedure revisions received extensive review during their development, and were well validated on the plant simulator.

The inspector considered that the licensee's followup actions to the loss of IA event were thorough. Non-destructive examination of other soldered joints in the IA system indicated that inadequate piping insertion is apparently not a wide-spread problem. Revisions to AP-IA.1 and AP-TURB.5 addressed reactor coolant system inventory and pressure control while IA in containment is isolated and provide adequate guidance for continued operation at power.

3.0 MAINTENANCE (IPs 62703, 61726)

3.1 Maintenance Activities

3.1.1 Routine Observations

The inspectors observed portions of plant maintenance activities to verify that the correct parts and tools were utilized, the applicable industry code and technical specification requirements were satisfied, the procedures were adequate and properly implemented, adequate measures were in place to ensure personnel safety and prevent damage to plant structures, systems, and components, and equipment operability was verified upon completion of post maintenance testing. The following maintenance activities were observed:

- Work Order 19501305, B-Auxiliary Feedwater Pump Breaker Amptector Setpoint Adjustment, observed September 19, 1995
- Installation and acceptance testing of a new instrument air compressor, observed at various times over the course of the inspection period

The inspectors concluded that the above activities were performed in a well controlled manner.

3.2 Surveillance and Testing Activities

3.2.1 Routine Observations

The inspectors observed portions of surveillances to verify proper calibration of test instrumentation, use of approved procedures, performance of work by qualified personnel, conformance to limiting conditions for operation (LCOs), and correct system restoration following testing. The following surveillances were observed:

- Periodic Test PT-16M-B, "Auxiliary Feedwater Pump B - Monthly," observed September 19, 1995

- Instrument Calibration Procedure CPI-TRIP-TEST-5.40, "Reactor Protection System Trip Test/Calibration for Channel 4 (Yellow) Bistable Alarms," observed October 2, 1995

- CPI-CR-NH3-MON-235, "Calibration of Control Room HVAC Foxboro Ammonia Analyzer," observed October 18, 1995

- PT-17.4, "Control Room Radiation R-36, R-37, R-38 and Toxic Gas Monitor Operability Test," observed October 18, 1995

- PT-16Q-A, "Auxiliary Feedwater Pump A - Quarterly," observed October 19, 1995

The inspector determined through observing the above surveillance tests that operations and test personnel adhered to procedures, test results and equipment operating parameters met acceptance criteria, and redundant equipment was available for emergency operation.

3.2.2 Turbine Driven Auxiliary Feedwater Pump Flow Control Valve Oscillations

On September 21, 1995, the inspector observed performance of periodic test PT-16M-T, "Auxiliary Feedwater Turbine Pump - Monthly." Soon after the pump started, the B-steam generator flow control valve, AOV-4298, produced several spike increases in flow (normal flow is 200 gpm, spike increases were on the order of 50 gpm). After discussion between test and operations personnel, operators took control of the valve and cycled it more fully open. When returned to automatic control, no further valve oscillations were experienced and the test was completed.

The inspector discussed the AOV-4298 oscillations with on-shift operations department personnel. Operators considered that intervention to cycle the valve during the surveillance test was acceptable because testing the operation of the flow control valves was not one of the purposes of the surveillance test. They did not consider that component or system operability was in question, because 1) there are no specific technical specification requirements for operability of the flow control valves, 2) their actions to cycle the valve had apparently eliminated the problem, and 3) the flow control valves are designed to fail safe (open).

The inspector reviewed the completed test procedure and noted that no record was made of the AOV-4298 oscillations. After the inspector discussed the valve oscillations with licensee management, the operations department initiated an abnormal condition/event report (ACTION Report 95-0315) to document the event. Subsequent investigation revealed fluctuations within the output range of the main control board controller. A work order (WO 19504134) was initiated to replace the controller, with work to be performed during a scheduled system maintenance outage or the upcoming refueling outage.

The inspector was concerned that an anomalous condition noted during testing of a safety system was not adequately evaluated by the licensee. In this instance, a defective auxiliary feedwater system flow control valve position controller would have gone undetected, despite evidence of the malfunction that was observed by both the Results and Test Group and the Operations Department. This concern is further discussed in section 6.1 of this report. Additionally, the inspector questioned the acceptability of operations department intervention, during testing, to attempt to correct an anomalous condition, when such action was not specified in the test procedure. The inspector will review licensee testing controls during a future inspection.

3.3 C-Service Water Pump Motor Incipient Failure

On October 4, 1995, operators started the C-service water (SW) pump as part of an operating SW pump exchange in preparation for a routine surveillance test. A Results and Test (R&T) technician subsequently noted an acrid odor, as produced by hot electrical insulation. The R&T technician reported this to the control room, and the observation was subsequently verified by an Auxiliary Operator. The C-SW pump was secured to perform troubleshooting.

Electrical testing indicated that an insulation breakdown occurred within one of the motor stator windings. The motor was removed and disassembled for inspection. During this process, evidence of a turn-to-turn short circuit in the windings was observed. This was the same failure mode that was suspected of having led to failure of the D-SW pump motor on August 9, 1995 (discussed in inspection report 50-244/95-15). Both the C- and D-SW pump motors were rewound by the same vendor in 1992. On this occasion, the C-SW pump motor was sent to another vendor to be rewound, with an expected turnaround time of about one month. At the close of the inspection period, the C-SW pump remained inoperable; having one SW pump inoperable does not place the licensee in a technical specification limiting condition for operation.

The inspector considered that the licensee's identification of winding insulation breakdown in the C-SW pump motor was noteworthy. The R&T technician showed a high level of alertness and safety awareness by noticing and reporting the abnormal odor to control room operators. In pursuing this indication, the licensee averted an actual motor failure and a challenge to one of the class 1E electrical busses. The licensee is examining options to improve long term reliability of the SW pump motors.

3.4 Licensee Action on Previous Inspection Findings

3.4.1 (Open), Inspector Follow-up Item (50-244/95-15-01), Water Leakage Into the Residual Heat Removal System Pump Room

As discussed in inspection report 50-244/95-15, the licensee is experiencing water leakage into the residual heat removal (RHR) system pump room. During this inspection period, the licensee continued their efforts to quantify and determine the source of the water leakage. The scale buildup on the wall has been removed to the extent necessary to install a leakage collection system. Chemical and isotopic analyses of the water leaking into the room are consistent with the makeup of spent fuel pool (SFP) water, except that the concentration of cesium in the leakage water is higher than in the SFP; the licensee considers that this may be due to a concentrating mechanism (drying and redissolving). The amount of water leakage into the room continues to be small, however, the rate of water loss from the SFP has not been quantified. Establishing the total leakage rate is important because it will indicate if SFP water is going to points other than the RHR pump room. However, precise measurement of small changes in SFP water inventory is complicated by the large volume of the tank and uncertainty in measuring evaporative losses. Technical Staff Request 95-158 was issued to obtain engineering support in determining the evaporation rate from the SFP. Licensee isotopic analyses currently show no evidence that SFP water is migrating through the ground.

In summary, the licensee has characterized the water leakage into the RHR pump room as SFP water. Progress has been made and efforts are continuing to quantify the leakage. The inspector will continue to monitor the licensee's progress in addressing this concern.

3.5 A-Charging Pump Varidrive Speed Controller Problems

The chemical and volume control system at Ginna uses three high head positive displacement charging pumps for reactor coolant system inventory control and chemical addition. These pumps are not part of the emergency core cooling system, but are safety significant in that at least one is required to maintain injection flow to the reactor coolant pump shaft seals. Pump capacity is controlled by varying the speed of the pump through a mechanical "varidrive" speed controller.

A routine three-year overhaul of the varidrive speed controller for the A-charging pump was performed during the 1995 refueling outage. After approximately two months, the varidrive began to make intermittent abnormal noises. When efforts to eliminate the noise were unsuccessful, a second overhaul was performed in July 1995. During post-overhaul testing, the two neoprene V-belts in the varidrive unit began to smoke as pump speed was decreased. The pump was secured and inspection of the varidrive revealed that some of the moving parts on the lower shaft of the V-belt drive train were not operating freely. Specifically, the conical discs that form the lower pulleys for the V-belts were not sliding freely on the shaft (by changing the distance between these discs, the effective radius of the pulley is changed, and thus the output speed is varied). The varidrive was lubricated in accordance with the quarterly maintenance requirement. The pump was returned to service and allowed to run-in at constant speed. After approximately 13 hours of operation, the pump speed was decreased, and the V-belts again began to emit smoke. Inspection revealed no deficiencies with the varidrive. With mechanics observing, the pump was restarted and its speed was varied. No problems were observed with operation of the varidrive, and the pump was returned to service. However, an abnormal noise from the varidrive again developed after approximately a week of operation. The cause of the noise was determined to be a bushing that serves as a mount for one of the upper shaft bearings. A slight amount of axial play in the bushing was allowing it to rub intermittently against a casing component. Shortly after the close of the inspection period, this problem was corrected by staking the bushing to the shaft to restrict axial motion.

The licensee performed a root cause analysis of the A-charging pump varidrive failures (report M-95-004). The probable cause of the abnormal noise that led to the second varidrive overhaul was the upper and lower sliding disc bushings that were not replaced during the first overhaul. Excessive clearances between the shafts and bushings resulted in additional wear and resultant noise. The apparent cause of the smoking V-belts was the new sliding disc bushings installed during the second overhaul that were too tight on the shafts, which caused sticking and an erratic sliding motion. The licensee determined that the failure of maintenance personnel to detect these conditions was attributed to inadequate training and inadequate work instructions. The recommended corrective actions included 1) evaluating the need for modification of the post-overhaul run-in procedure; 2) more frequent inspection of sliding disc components; 3) revision of the varidrive overhaul procedure to specify replacement of the disc bushings during the three year overhaul, to record bushing-to-shaft clearances, and to better describe proper fitup and operation of the sliding discs; and 4) revision of the varidrive training course to cover proper fitup of sliding disc bushings to the shafts. The licensee was still evaluating the necessary corrective actions at the close of the inspection period.

The inspector was concerned that the series of mechanical problems with the A-charging pump varidrive were maintenance-related. The charging pump varidrive units are overhauled on a rotating basis, such that mechanics perform at least one overhaul per year. With this experience base, it is not clear why achieving proper component fitup in the A-charging pump varidrive presented so many difficulties. The inspector will continue to monitor the licensee's address of the A-charging pump maintenance problems.

3.6 Pressurizer Safety Valve Position Indication Instrument

During a routine inspection in the control room, the inspector noted that the valve position indication for pressurizer safety valve PCV-434 indicated an abnormally large deviation from zero (valve closed). The expected indication was within a few thousandths of an inch of zero; on this occasion, however, the indication was +0.010 inch. Control room operators confirmed that this was an abnormal indication, and stated that Instrument and Control (I&C) personnel would be contacted to investigate.

A trouble report was initiated which stated that PCV-434 indicated +0.012 inches and may need calibration. The shift supervisor's review of this report indicated that the indicator was still operable and that no interim compensatory action was required. A work package (Work Order 19504405) was generated to clean the display unit in accordance with instrument calibration procedure CP-I-PRZR-LVDT-434/435, "Calibration of Linear Variable Differential Transformers (LVDTs) on Pressurizer Relief Valves 434 and/or 435." Work commenced on the afternoon of October 13, 1995, at which time the PCV-434 valve position indicator was declared inoperable. This placed the licensee in a seven day action statement per technical specification 3.5.3.2.

The technicians performing the maintenance observed oxidation and corrosion products on the electrical connectors of the instrument signal conditioner. All connectors to the signal conditioner were cleaned, after which the

indication returned to normal (-0.001 inch). Work was completed and the PCV-434 valve position indicator was declared operable at 9:00 p.m. on October 13, 1995.

Through review of the completed work package and discussions with personnel, the inspector determined that the actions taken in response to the abnormal valve position indication for PCV-434 had been adequate. Although troubleshooting was not begun until several days after the problem was identified, the position indication remained within the acceptance tolerance of the calibration procedure (± 0.020 inches) until the day that work was performed. The inspector concluded that the technical specification requirement for instrument operability was satisfied during that period. The inspector had no additional concerns on this matter.

4.0 ENGINEERING (IPs 71707, 37551)

4.1 Update: Steam Generator Replacement Project

RG&E's project to replace both steam generators (SGs) during the next refueling outage (March 1996) is currently on schedule and preliminary construction activities are in progress. Overall, the project is well managed and coordinated between RG&E and Bechtel, the principal contractor. The major construction activities now completed or currently in progress are as follows:

- Two foundation pads are in place adjacent to the containment to support the special Lampson crane that will be used to lift the SGs into and out of the containment through access holes cutout of the dome. The pads are each 4 ft thick and are made with high strength reinforced concrete.

- A full-scale partial containment dome mock-up was completed in September 1995. Throughout December and January, the mock-up will be used to confirm the methodology for containment concrete excavation and restoration, rebar removal and reconstruction, containment liner plate removal and rewelding, radiological work controls, and for personnel training.

- Installation of an equipment support and work platform ("CROWN") around the containment dome is in progress. The platform will be used to support concrete excavation, liner plate cutting and removal, and containment restoration. The platform anchors have been set into the concrete and erection of the structural steel is ongoing.

- Modification of the containment facade structure is in progress. Reinforcements to the steel framework are needed to support a material storage enclosure and personnel comfort station that will be installed adjacent to the containment dome work area.

- Construction of a containment access facility (CAF) is approximately 50% complete. This new support building will house health physics and radiological protection personnel, provide worker support facilities, and will serve as the sole control point for containment access during the

outage. Utilities have been connected and the facility is scheduled for availability by early December 1995.

- Reinforcement of a portion of the underground 34.5 kilovolt power line duct bank is ongoing. The duct bank houses the station's electrical output conductors, and lies beneath the Lampson crane and SG haul route inside the site's protected area. The existing reinforcement does not have sufficient strength to support the heavy loads that will pass over it.

- Reinforcement of storm drains beneath the main site parking lot is in progress. A 100 ft X 600 ft area of the parking lot will be used to assemble and load test the Lampson crane before bringing it inside the protected area. The existing storm drains cannot support the weight of the crane.

- Construction of a new temporary storage facility for the old SGs is essentially complete. The facility is a large bunker-type concrete enclosure with appropriate radiological controls for storing the old SGs onsite until permanent offsite disposal is arranged. The facility was designed to store the old SGs through the end of the current license period, if necessary.

- Dredging of the barge slip at Bear Creek harbor is about 50% complete. The harbor is two miles east of the site and is being dredged and upgraded to provide a docking facility for arrival of the new SGs. Fabrication of the new SGs is nearly complete and they are scheduled to arrive at Bear Creek in February 1996.

4.2 Licensee Action on Previous Inspection Findings

4.2.1 (Open), Unresolved Item (50-244/95-08-01), Motor Operated Valve MOV-4007 Failure

Inspection Report 50-244/95-08 documented the failure of MOV-4007, the "A" auxiliary feedwater (A-AFW) pump discharge valve, during a monthly surveillance test in April 1995. MOV-4007 did not open on demand under full motor torque, and the actuator motor did not trip out under the high overcurrent condition. The actuator motor had to be replaced due to damage caused from overheating. The failure was associated with an unusually high vibration level experienced in MOV-4007 that dislodged the actuator handwheel and loosened the actuator torque switch, causing it to drift to its maximum torque setting. No limiter plate was installed on the torque switch. When the valve was closed with maximum available motor torque and running inertia, it could not be reopened. RG&E has an established policy requiring the use of torque switch limiter plates on all GL 89-10 MOV torque switches to prevent overtorquing incidents. Several other MOVs in the GL 89-10 program also do not have limiter plates installed. This item was left unresolved pending completion of the root cause analysis and further NRC review of RG&E's corrective actions.

In June 1995, RG&E completed a root cause analysis (AR 95-051) of this event, which indicated that high vibration was the most probable root cause of the valve failure, since it was intense enough to loosen the torque switch set screw. The analysis also noted various inconsistencies in the licensee's MOV program documents regarding the use of torque switch limiter plates. In addition to the initial corrective actions to repair and restore MOV-4007 to operability, other actions included inspections of other similar AFW MOVs for proper torque switch settings and signs of overtorque. MOV maintenance procedures were also identified for enhancements to assure proper work practices are included for tightening torque switch set screws.

AR 95-051 did not directly associate the missing limiter plate as a contributing cause of the valve failure, and it did not conclude that the failure would have been prevented if a limiter plate was installed. Also, the corrective actions identified by AR 95-051 did not include installation of a limiter plate on the torque switch. However, the licensee currently plans to install limiter plates on MOV-4007 and on the corresponding valve (MOV-4008) in the B-AFW train during the next refueling outage. Both MOV-4007 and -4008 are high risk significant valves that could be subjected to the same high vibrations under design-basis accident conditions. The licensee has not yet resolved all inconsistencies in the MOV program requirements regarding the use of limiter plates on Generic Letter 89-10 valves. RG&E expects the program requirements to be revised before the next refueling outage, and to have limiter plates available for installation on the appropriate MOVs. This item remains unresolved pending full NRC review of the effectiveness of all corrective actions associated with this event.

5.0 PLANT SUPPORT (IP 71750)

5.1 Licensee Action on Previous Inspection Findings

5.1.1 (Open), Unresolved Item (50-244/95-15-02), Radiological Work Controls and External Exposure Monitoring in Radiologically Controlled Areas

Inspection Report 50-244/95-15 documented several incidents that occurred in recent months where personnel entered a radiologically controlled area (RCA) to perform work in the plant, but did not log into the radiation work permit (RWP) access control system and/or did not have the secondary dosimetry required for entry. These incidents represented a failure of radiation workers to adhere to procedures pertaining to RCA access, RWP work controls, and possession of all required dosimetry. The licensee wrote several ACTION Reports to initiate root cause investigations and corrective actions. After one individual improperly entered a locked high radiation area, the licensee initiated a Human Performance Evaluation System (HPES) review (95-0240) to conduct an in-depth root cause analysis and human performance review of these incidents, and to determine the necessary corrective actions to prevent recurrences.

HPES 95-0240 was completed shortly after the end of this inspection period. The report identified several "passive barriers" that contributed to a breakdown in radiation worker adherence to RCA access requirements, such as "Rad Worker Self-Checking," "Training," and "Rad Postings." The HPES also

recommended ongoing reviews for improper RCA access. These reviews revealed several additional instances of improper RCA access during October 1995 that involved radiological protection technicians. ACTION Reports 95-0349 and -0350 were initiated to evaluate the additional incidents.

The licensee's efforts to correct the conditions associated with this problem are still ongoing. RG&E's Chemistry/Radiation Protection Department purchased a turnstile barrier for installation at the RCA access point as an active barrier to prevent personnel access before they make the necessary entries into the RWP access control system. Shortly after this inspection period ended, the turnstile was temporarily installed and it appeared to function properly. The turnstile remains locked until it electronically reads each individual's RADOS detector. After confirming that the proper log entry was made into the RWP system, the turnstile unlocks and the individual gains access to the RCA. The licensee will further evaluate the proper configuration of personnel barriers around the control point and the turnstile, and will complete a permanent installation at a later time. The licensee is also continuing to evaluate the effectiveness of the RWP system and personnel adherence to radiological and administrative work requirements. This item remains unresolved pending completion of all corrective actions and subsequent NRC review.

6.0 SAFETY ASSESSMENT/QUALITY VERIFICATION (71707)

6.1 Corrective Action Program Procedure Revision

As discussed in inspection report 50-244/95-13, the licensee instituted a single entry point corrective action program in July 1995. Interface procedure IP-CAP-1, "Abnormal Condition Tracking Initiation or Notification (ACTION) Report," (AR) is the governing procedure for the program. The licensee has conducted at least two internal audits of their corrective action system. The audits have revealed that several ARs have not been completed within the original time targets assigned and that the threshold for writing ARs was inconsistent between the various plant departments. The licensee has made some improvements in the processing of ARs, and the Ginna Production Superintendent verbally presents all new ARs to managers and supervisors at each morning meeting. However, the inspector was concerned that some significant events (not identified by the corrective action audits) were not being evaluated through the AR system. Examples included:

- Unanticipated rod motion that occurred following a momentary loss of power to instrument bus B on June 30, 1995, was not reported until the inspector discussed the occurrence with licensee management (discussed in inspection report 50-244/95-13; this event could have been reported either by an ACTION Report or an A-25.1 Event Report).

- Oscillations of valve AOV-4298 (Turbine driven auxiliary feedwater pump flow control valve) during surveillance testing on September 21, 1995, were not reported until the inspector discussed the occurrence with licensee management (discussed in section 3.2.2 of this report).

- On October 4, 1995, the inspector noted that quadrant power tilt ratio (QPTR) was not being manually calculated while the QPT monitor was inoperable. Technical specification requirements for this condition were being satisfied and the data necessary to perform the calculation was being recorded; however, alarm response procedure AR-F-29, "PPCS Axial or Quadrant Power Tilt/PPCS LTOP Hi-Low Pressure," requires that QPTR be calculated every 12 hours. QPTR was subsequently calculated for the period that had been missed; however, an ACTION Report was not generated.

Shortly after this inspection period, the licensee issued revision 2 to IP-CAP-1. The revision provides an extensive list of examples to be used as guidance on the expected threshold for initiation of an ACTION Report. The revision did not go into effect before the close of the inspection period, however, the inspector considered that the proposed list of examples will establish an effective threshold for event reporting. The revised procedure appears to improve the reporting and tracking process for ARs; however, the inspector will evaluate licensee effectiveness in implementing the threshold in a future inspection as part of the routine inspection program.

6.2 Plant Operations Review Committee (PORC) and Nuclear Safety Audit and Review Board (NSARB) Meetings

During recent months, RG&E revised their PORC meetings to simplify and improve the review process, and to reorganize the format of documents related to PORC business, such as meeting minutes. Administrative procedure A-205, "Plant Operations Review Committee Operating Procedure," was revised and updated to clarify PORC member responsibilities and to reflect changes in the formal committee processes. The draft A-205 revision was submitted to the full PORC committee for review and is currently awaiting the incorporation of member comments prior to final approval.

The inspector attended numerous PORC meetings over recent months to evaluate the Committee's review of plant safety issues. All PORC meetings were formally conducted under the direction of the PORC chairman and each safety issue presented was thoroughly reviewed by the Committee. Each Committee member performed as an independent evaluator and concerns raised by individual members prompted further evaluation by the full Committee. When the PORC could not reach a satisfactory consensus on particular matters, the PORC chairman was prompt in returning issues to the plant line organizations for further evaluation and resubmittal to PORC. The PORC chairman and members considered the revised agenda format and review process to be more efficient and allowed them to focus more effectively on plant safety concerns.

On September 28, 1995, the inspector attended the quarterly NSARB PORC Sub-committee meeting that was conducted by the PORC chairman and attended by one NSARB member, one PORC member, and a plant employee (the PORC Chairman is also a full member of the NSARB). The Sub-committee reviewed all PORC meeting minutes for completeness, technical clarity and accuracy, and separately evaluated PORC decisions and followup actions for adequacy. The Sub-committee members also compared all PORC decisions against the licensee's existing NRC commitments. Minutes of the meeting were recorded and action items were initiated to followup on the Sub-committees findings and recommendations. The

inspector considered that the Sub-committee fully reviewed their agenda items and effectively evaluated PORC activities for the current quarter.

6.3 Periodic Reports

Periodic reports submitted by the licensee pursuant to Technical Specification 6.9.1 are routinely reviewed by the inspectors. The inspectors verify that the reports contain information required by the NRC, that test results and/or supporting information are consistent with design predictions and performance specifications, and that reported information was accurate. The following report was reviewed during this inspection:

- Monthly Operating Report for September 1995

No unacceptable conditions were identified.

6.4 Licensee Event Reports

A Licensee Event Report (LER) submitted to the NRC was reviewed and the inspector determined that the details were clearly reported, the cause was properly identified, and the corrective actions were appropriate. The inspectors also determined that the potential safety consequences were properly evaluated, the generic implications were indicated, events that warranted additional follow-up were identified, and the licensee met the applicable requirements of 10 CFR 50.73.

The following LER was reviewed (date indicated is event date):

- LER 95-008, Secondary Transient, Caused by Loss of "B" Condenser Circulating Water Pump, Results in Manual Reactor Trip (August 25, 1995)

The inspector concluded the LER met regulatory requirements and appropriately evaluated the safety significance of the event. LER 95-008 is closed.

7.0 ADMINISTRATIVE

7.1 Exit Meetings

At periodic intervals and at the conclusion of the inspection, meetings were held with senior station management to discuss the scope and findings of inspections. The exit meeting for inspection report 50-244/95-16 (Licensed operator requalification program evaluation, conducted September 18-22, 1995) was held by Mr. Joseph D'Antonio on September 22, 1995. The exit meeting for inspection report 50-244/95-18 (Confirmatory measurements for radioactive waste treatment and effluent and environmental monitoring programs, conducted October 2-6, 1995) was held by Mr. James Kottan on October 6, 1995. The exit meeting for the current resident inspection report 50-244/95-17 was held on October 24, 1995.