U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Inspection Report 50-244/95-15 License: DPR-18

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

Inspection: July 30, 1995 through September 9, 1995

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Approved by: W. J. Lazarus, Chief, Reactor Projects Section 3B — 9/29/95 Date

INSPECTION SCOPE

Plant operations, maintenance, engineering, plant support, and safety assessment/quality verification.

INSPECTION EXECUTIVE SUMMARY

Operations

The plant operated at full power (approximately 97 percent) for the majority of the inspection period. During this period, a lightning strike caused a momentary loss of two of the four class 1E electrical buses. Corrective maintenance had been performed on the associated emergency diesel generator (EDG) earlier that day; although administratively inoperable at the time of the event, licensee management had earlier made the decision to defer acceptance testing and to place the EDG in automatic standby due to impending storm conditions. As a result, the EDG was available during the event. The EDG started and loaded normally, and all other engineered safety features equipment functioned as required. The event had no effect on reactor power. The operators responded well to the event. Management's decision to make the EDG available prior to performance of acceptance testing was prudent.

Later in this period, the shift supervisor directed that the reactor be manually tripped from approximately 70 percent reactor power due to a secondary plant transient that was caused by the loss of power to a main circulating water (CW) pump. All engineered safety features equipment functioned as required and operators promptly stabilized the plant in hot shutdown. The shift supervisor's decision to manually trip the reactor was appropriate. Operator response to the transient and reactor trip was good. Management effectively integrated resolution of CW pump motor failure and other secondary plant technical issues with plans for plant restart. The licensee's decision to also inspect the unaffected CW pump was prudent and led to identification of a malfunction that would otherwise have gone undetected.

Maintenance

As a result of maintenance activities to correct low fuel oil pressure on the B-emergency diesel generator, the licensee determined that the replacement fuel pump model that was specified in the vendor manual required modification to produce the desired discharge pressure. Although the required value of fuel oil pump discharge pressure is engine-specific, the licensee issued an interim notice to the NRC of the problem in accordance with 10 CFR Part 21. RG&E had obtained spare replacement pumps through a commercial procurement process and later dedicated them for nuclear service. All of the pumps that had shown reduced performance when installed on the EDG had been successfully tested by RG&E prior to being released to the spare parts system, raising questions regarding the adequacy of commercial grade dedication testing. The licensee indicated that a review of the procurement and dedication processes will be conducted as part of the root cause evaluation.

During routine surveillance testing of the undervoltage (UV) protection system, the practice of testing a malfunctioning indicating light by using a nearby "working" indicating light bulb produced an alarm indication of a possible loss of undervoltage protection for safeguards bus 17. The B EDG was started and connected to bus 17 to assure the continued availability of undervoltage protection during troubleshooting. The cause was subsequently determined to be that the indicating light in question had been tested with an

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incompatible light bulb, which caused a fuse in the indicating light circuitry to blow. The undervoltage protection function of the cabinet had not been affected, and therefore, operability of the UV cabinet had not been lost as a result of this event.

Corrective action included listing all the various indicating lamps of all 480 VAC UV relay and control cabinets under the applicable equipment identification numbers (EINs). The licensee also conducted training on this incident for all Results and Test Department personnel to review its causes and corrective actions. The inspector considered that the potential for a problem similar to this incident existed in other electrical maintenance and test areas. The licensee subsequently agreed to review the standard practices for troubleshooting indicator light faults and to expand training on the event to include I&C technicians and electrical maintenance personnel.

The licensee has experienced water leakage into the residual heat removal system pump room for several years. The inleakage is currently too small to represent any operational safety concern for the plant. All potential sources of the leakage have not yet been positively identified by the licensee. Ground water appears to be entering the room; however, RG&E's analyses also indicate the presence of boric acid and radionuclides that are also present in the spent fuel pool. The licensee is removing the scale buildup from the area of inleakage to facilitate better collection. The inspector will continue to monitor licensee efforts to quantify the water inleakage rate and to positively identify all sources.

The licensee is planning to replace both steam generators (SGs) during the next refueling outage, currently scheduled to begin in March 1996. The inspectors are monitoring the licensee's preparations for SG replacement and are reviewing the effectiveness of RG&E's project management controls, the safety evaluations prepared for the project, and the engineering specifications associated with the new SGs. During this period, the NRR and Region I-based Project Managers visited the site to obtain an overview of the SG replacement project (SGRP) and to coordinate inspection plans. No areas of concern were identified during review of the SGRP. The licensee was well prepared to address questions posed during the discussions.

Engineering

During a routine start of the D-service water (SW) pump, the motor began to emit smoke and flames. The licensee determined that the likely cause of failure was a phase-to-ground short circuit that developed due to vibration-induced degradation of the winding insulation. A similar failure had occurred with this motor several years earlier, and the motor had been refurbished by a vendor and returned to service. On this occurrence, the licensee decided to replace the motor with a commercial-grade motor that had been intended for installation and dedication during the next refueling outage. The licensee initiated an internal engineering action to review all of the known new motor characteristics, to determine the extent of dedication testing necessary, and

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to perform 10 CFR 50.59 safety evaluations of any necessary plant modifications necessary to perform dedication testing during plant operations.

The inspector considered that the licensee made a sound decision to replace the D-SW pump motor and restore the pump to operability in a timely manner through installed dedication testing. The new motor's performance was thoroughly modeled and reviewed, with conservative engineering assumptions and ample consideration for the existing plant design and safety evaluations.

Plant Support

During recent months, several incidents have occurred where personnel entering a radiologically controlled area (RCA) in the plant did not log into the licensee's radiation work permit (RWP) access control system, and/or did not have the secondary alarming dosimetry required for entry. None of the incidents involved entry without the primary personnel exposure monitoring devices (thermoluminescent dosimeters); however, all represented a failure of qualified radiation workers to adhere to the procedure requirements pertaining to RCA access, RWP work controls, and possession of alarming dosimetry.

Although of minor radiological consequence, these incidents represent a relatively high occurrence over a seven month period of a lack of adherence to procedure requirements for radiological work controls and access to radiation and high radiation areas. This item is unresolved pending NRC review of the licensee's human performance evaluation report and the licensee's implementation of permanent corrective actions.

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DETAILS

1.0 OPERATIONS (71707)

1.1 Operations Overview

At the beginning of the inspection period, the plant was operating at full power (approximately 97 percent). On August 3, 1995, one of the two offsite electrical power supplies deenergized due to a lightning strike. This resulted in a momentary loss of two of the four class 1E electrical buses until the associated emergency diesel generator started and assumed electrical loads. All engineered safety features equipment functioned as required and the event had no effect on reactor power.

On August 25, 1995, the shift supervisor directed that the reactor be manually tripped from approximately 70 percent reactor power due to a secondary plant transient that was caused by the loss of a main circulating water pump. All engineered safety features equipment functioned as required and operators promptly stabilized the plant in hot shutdown. Following resolution of several balance-of-plant issues, a plant startup was performed on August 26, 1995. Plant power was escalated to 48 percent and held pending repair of the failed main circulating water pump. Before this was completed, a through-wall pipe leak was discovered in a moisture separator drain line to the main condenser. A controlled steam plant shutdown was conducted on August 28, 1995, to support replacement of the affected piping. During this shutdown, repairs were also completed to the main circulating water pump. A steam plant startup was conducted later the same day and the plant returned to full power operation on August 30, 1995. There were no other significant operational events or challenges during the inspection period.

1.2 Operational Experiences

1.2.1 Loss of One Offsite Power Supply

On August 3, 1995, the B-emergency diesel generator (EDG) was declared inoperable to investigate a decreasing trend in fuel oil pressure; the maintenance activity is discussed in detail in section 2.2.2 of this report. By that afternoon, maintenance was complete and the B-EDG was prepared for operation to conduct acceptance testing. However, licensee management decided to defer testing and to align the B-EDG for normal operation due to lightning storm activity.

At 3:26 p.m. on August 3, 1995, a lightning strike occurred offsite on one of the plant's two offsite electrical power circuits (circuit 751). As a result of the transient, circuit 751 was deenergized by protective relays at offsite station 204. This resulted in a loss of power to the two class 1E 480-volt electrical busses that were being supplied by circuit 751 (busses 16 and 17). In response to the power loss, the B-EDG automatically started and reenergized the two busses. Operators responded in accordance with abnormal procedure AP-ELEC.1, "Loss of 12A and/or 12B Busses," to stabilize affected systems. All engineered safeguards equipment functioned as required and plant power was not affected by the transient.

At 4:28 p.m., the electrical distribution system was realigned such that the unaffected offsite electrical power circuit (circuit 767) was supplying all four class 1E 480-volt electrical busses. The B-EDG was shut down and returned to standby. Circuit 751 was returned to service at 4:44 p.m.; however, the circuit was not placed in service due to continued storm activity.

Based on control room observations, review of logs, and discussions with plant personnel, the inspector determined that operators had responded appropriately to the loss of circuit 751. The inspector observed good procedural adherence during plant restoration. Operator communications were formal and concise. The Control Room Foreman provided excellent oversight of the restoration activities. Additionally, licensee management's decision to defer EDG acceptance testing due to storm activity was prudent. The technical specification requirements for one EDG being inoperable had been satisfied prior to the event, and loss of one offsite power supply does not alter these requirements. A four-hour non-emergency report was made to the NRC as required by 10 CFR 50.72.

1.2.2 Manual Reactor Trip due to Loss of a Main Circulating Water Pump

The main circulating water system supplies cooling water to the main condensers to condense exhaust steam from the two low pressure turbines. The system consists of two headers, each of which is supplied by a circulating water (CW) pump. Each header supplies a main condenser. The headers are cross-connected upstream of the main condensers to allow for reduced power operations with a single operating CW pump. After passing through the main condensers, main circulating water is returned to the lake via a common discharge canal.

At 5:41 a.m. on August 25, 1995, the B-CW pump tripped. Control room operators were alerted to the problem by the associated main control board annunciator. Turbine load was rapidly reduced to approximately 50 percent in accordance with abnormal procedure AP-CW.1, "Loss of a Circulating Water Pump." Operators then noted that turbine backpressure in the condenser associated with the failed CW pump (B-main condenser) was increasing, and that the normally equal levels in the main condenser hotwells were diverging; specifically, hotwell level in the B-main condenser was decreasing, while level in the other hotwell (associated with the operating CW pump) was increasing.

These abnormal conditions were developing because the B-CW pump had tripped rather than having been secured as part of an orderly transition to single CW pump operation. Although the plant can operate at up to 50 percent power on a single CW pump, the system must first be reconfigured; specifically, the discharge isolation valve for the pump to be secured must be closed before the CW pump is stopped. In this case, the discharge isolation valve was initially open when the B-CW pump stopped. The valve is motor operated and automatically began to close when the B-CW pump tripped; however, closure of this large valve takes on the order of minutes. As a result, cross-connected flow from the A-CW pump discharged back to the idle B-CW pump rather than being forced through the B-main condenser. The loss of cooling water caused pressure in the secondary side of the B-main condenser to increase. The two main condensers connect to a common suction header for the main condensate pumps; consequently, hotwell level in the B-main condenser dropped rapidly as condensate was either forced into the lower pressure A-main condenser hotwell or supplied to the main condensate pumps.

As level in the B-main condenser hotwell approached empty, the condensate pumps began to cavitate and discharge pressure began to drop. This translated to low suction pressure to the main feedwater pumps. Given the potential for damage to secondary plant equipment and a possible loss of main feedwater flow, the shift supervisor directed that the reactor be manually tripped. With reactor power at approximately 70 percent and decreasing, operators tripped the reactor at 5:43 a.m., approximately two minutes after loss of the B-CW pump. Plant response to the trip was normal. All safety systems and equipment responded as required. Operators promptly stabilized plant conditions in hot shutdown.

Investigation revealed that the B-CW pump breaker had tripped due to actuation of the power factor relay. The relay trip setpoint was checked and found to be correctly set. Inspection of the B-CW pump motor revealed that a diode in the synchronizing circuit had become disconnected. The licensee determined that this failure would have produced a power factor trip. Additionally, one of the mounting bolts for the baseplate that attaches synchronizing circuitry components to the motor rotating assembly was found sheared at the head. Localized minor insulation damage on the stator windings was also noted. The licensee theorized that when failure of this bolt occurred, the bolt head had struck and broken the diode conductor; as it continued out, the bolt head also produced the stator winding insulation damage. Ultrasonic testing was performed on the remaining baseplate bolts for the B-CW pump, and no indications of incipient failure were detected. Additionally, the A-CW pump was shut down for examination of the baseplate bolts, with no problems noted. Licensee efforts to determine the cause of the bolt failure were continuing at the end of the inspection period.

Following replacement of the failed bolt and diode, and repair of the stator winding insulation, an operational test of the B-CW pump was attempted. This test was unsuccessful, with the motor breaker again tripping on actuation of the power factor relay. Additional troubleshooting revealed that a module in the synchronizing circuit was defective. Following replacement of this module, the B-CW pump was tested satisfactorily and returned to service. Although no problems had previously been experienced with the A-CW pump, it also failed to start following completion of the bolt inspection. Troubleshooting revealed that the same synchronizing module that had failed in the B-CW pump was also malfunctioning in the A-CW pump. Both pump motors had been refurbished during the 1995 refueling outage, and the licensee suspected that heating during application of varnish to the rotors may have contributed to failure of the synchronizing modules. A second replacement module was not immediately available. Licensee management decided that the plant would be returned to operation, with power limited to 50 percent until the A-CW pump could be returned to service.

A reactor startup was commenced on August 26, 1995, and criticality was achieved at 1:53 p.m. The main generator was closed on the grid at 5:56 p.m., and plant power was raised to approximately 50 percent by morning of the following day. On August 28, 1995, a steam plant shutdown was performed to support repair of a through-wall pipe leak in a moisture separator reheater drain line. The reactor was maintained critical during this maintenance. Coincident with the shutdown, a replacement module for the A-CW pump motor synchronizing circuit was obtained and installed. A steam plant startup was conducted later the same day, and full power was achieved at 2:44 a.m. on August 30, 1995.

The inspector considered that the licensee's action to manually trip the reactor following loss of the B-CW pump was appropriate. Through discussions with licensee personnel, review of archived plant data, and attendance of the post trip review meeting, the inspector concluded that operators had responded well to the reactor trip and that the plant had responded normally. No technical specification requirements had been violated, and no technical issues related to the reactor plant were identified. A four-hour non-emergency report was made to the NRC as required by 10 CFR 50.72. Licensee management effectively integrated resolution of the B-CW pump motor failure and other secondary plant technical issues with plans for plant restart. The licensee's decision to also inspect the A-CW pump was prudent and led to identification of a malfunction that would otherwise have gone undetected.

2.0 MAINTENANCE (62703, 61726)

2.1 Maintenance Activities

2.1.1 Routine Observations

The inspector 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, adequate measures were in place to ensure personnel safety and prevent damage to plant structures, systems, and components, and to ensure that equipment operability was verified upon completion of post maintenance testing. The following maintenance activities were observed:

- Placement of the new D-SW pump motor (observed August 15, 1995), motor power cable junction box modification (observed August 21, 1995), and motor/pump commercial grade dedication and acceptance testing (observed August 24, 1995).

The inspector concluded that the above activities were performed in a well controlled manner and that the maintenance craft actions to install the new pump motor and to perform the required acceptance test represented good quality performance.

2.2 Surveillance and Testing Activities

2.2.1 Routine Observations

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:

- PT-12.2, "Emergency Diesel Generator B," observed August 6, 1995
- PT-36M-D, "Standby Auxiliary Feedwater Pump C - Monthly," observed August 16, 1995
- PT-12.1, "Emergency Diesel Generator A," observed September 5, 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.

Additionally, during an inspection of the B-EDG following the performance of PT-12.2 on September 7, 1995, the inspector noted that the prelubricating oil pump was not running. This pump runs at all times when the EDG is shutdown in standby, and provides engine lubrication during startup and shutdown. The inspector reported the problem to the shift supervisor, and the B-EDG was declared inoperable. Troubleshooting revealed that the pump motor start relay had failed. This suggested that the motor may never have started after the EDG had been secured at the conclusion of PT-12.2. The relay was replaced and the EDG was returned to service later that day. As corrective action, the licensee is modifying PT-12.1 and -12.2 to include a verification that the prelubricating oil pump is running after the EDG is secured.

2.2.2 B-Emergency Diesel Generator Low Fuel Oil Pressure and 10 CFR 21 Report

On August 3, 1995, the B-emergency diesel generator (EDG) was declared inoperable to investigate a decreasing trend in fuel oil pressure. Normal fuel pressure is 38-45 psig, but pressure had decreased in July 1995 to the licensee's alert range alert limit of 35 psig. In August 1995, pressure decreased below the action limit of 32 psig at full diesel load. The licensee completed maintenance on the fuel oil system the same afternoon by replacing system filters, check valves, the pressure regulating/relief valve, and flexible fuel lines. The inspector witnessed the post maintenance acceptance testing and observed that fuel oil pressure remained below the action limit with the diesel fully loaded. During the test, the licensee's attempts to adjust the pressure higher were not successful. Consequently, the licensee replaced the fuel pump with a new spare from stock and retested the diesel. No pressure improvement was observed with the new pump. The licensee again replaced that pump with another new spare pump, and repeated the test, but no improvement in fuel pressure resulted.

The licensee held discussions with the diesel vendor to confirm that the fuel pump was the correct model (Tuthill Model 2CF-CC) and to investigate other possible causes for the low pressure. However, the vendor technical manual indicates that fuel pump performance can be improved by removing shims from the pump cover gasket. The licensee investigated this option and determined that end play in both of the replacement pumps was larger than the minimum specified in the vendor manual, and larger than the end play in another spare pump known to have good pressure. Cover gasket shims were removed from the replacement pumps to reduce end play, and the diesel was retested. The test was successfully completed when fuel pressure was restored to 55 psig at no load and 44 psig at full load.

On August 28, 1995, the licensee issued an interim notice to the NRC of the problem in accordance with 10 CFR Part 21, indicating that a full written report would be forthcoming by September 25, 1995. The preliminary information supplied by RG&E indicated that the fuel pump was an exact replacement (by model number) for the original pump that the licensee had tested to meet the performance requirements established by the original manufacturer. The safety concern noted by RG&E that warranted a 10 CFR Part 21 report was that the diesel may not have been able to maintain its design loading with reduced fuel pressure from similar pumps on EDGs at other nuclear facilities. Although the diesel manufacturer does not specify a minimum fuel pressure for operability, the actual lower limit varies from diesel to diesel and had not been established for the EDGs at Ginna. RG&E obtained their spare replacement pumps through a commercial procurement process and later dedicated them for nuclear service with a shop test using the pumps' original performance specifications. All of the pumps that showed reduced performance when installed on the EDG had been successfully tested by RG&E prior to being released to the spare parts system. The reason for this discrepancy has not yet been determined.

The inspector discussed with the licensee the potential for procurement or commercial grade dedication issues that may have contributed to the degraded fuel pump performance. The licensee shared these concerns and indicated that a review of the procurement and dedication processes will be conducted as part of the root cause evaluations performed through the action report process that was used to pursue the pump pressure problem.

2.2.3 Bus 17 Undervoltage Cabinet Indicating Light Failure

On August 9, 1995, the licensee was performing a regularly scheduled surveillance test (PT-9.1.17), "480 VAC Undervoltage Protection," on safeguards bus 17. During the test, Results and Test Department technicians noted that the lamp for the "125 VDC Normal" indication on the undervoltage (UV) relay cabinet panel was not lit as expected. A technician put a spare bulb into the socket, which lit up momentarily, but then blew out. The technician then used an adjacent working bulb from the "120 VAC Normal" indicator on the cabinet panel to again "test" the socket. This bulb also blew out and caused a DC control voltage failure alarm for the bus 17 UV cabinet to annunciate in the plant's control room.

The technicians and plant operators immediately notified their supervision of this condition, who in turn discussed the situation with the Plant Production Superintendent, the Operations Manager, and the System Engineer. Loss of UV protection on safeguards bus no. 17 for greater than 1 hour requires an entry into technical specification LCO 3.0.1, and immediate initiation of a plant shutdown. The Plant Superintendent concluded that the operability of the UV cabinet was not assured under the existing conditions and determined that the B-EDG should be started and connected to bus 17 as a conservative action to assure the continued availability of normal power and undervoltage protection to bus 17. An entry into the action statement for TS 3.0.1 was therefore not required.

The licensee generated action report AR 95-0249 to review the causes of this incident and to initiate maintenance actions to troubleshoot and repair the faulty indicator lamp problems. It was determined that the control room trouble alarm resulted from a blown cabinet fuse (F-1) after arcing in the AC socket was caused by the installation of a DC bulb. The F-1 fuse conducts 120 VAC from instrument bus 16 to the bus 17 UV cabinet and energizes the panel indication lamps only, without affecting the 480 VAC UV protection function of the cabinet. Operability of the UV cabinet was therefore not lost as a result of this event. However, the F-1 fuse was replaced, the proper bulb was installed in the "125 VDC Normal" socket, and PT 9.1.17 was completed satisfactorily before the B-EDG was secured and normal power restored to the bus 17 UV cabinet.

The licensee dispositioned AR 95-0249 by listing all the various indicating lamps of all 480 VAC UV relay and control cabinets under the applicable equipment identification numbers (EINs). This was done to assure that bulbs drawn from spare parts and used to replace faulted bulbs on UV cabinets would have the required electrical characteristics confirmed by Procurement Engineering. The licensee also conducted training on this incident for all Results and Test Department personnel to review its causes and corrective actions. The inspector discussed with the licensee the apparent fact that it was not unusual for I&C technicians and electrical maintenance technicians (both at RG&E and industry-wide) to "borrow" light bulbs from adjacent locations to test whether indicator light problems were "bulb-related" or "socket-related." The licensee agreed that the potential for a problem similar to this incident existed in other electrical maintenance and test areas, and stated that a review would be performed and any appropriate changes made to the standard practices for troubleshooting indicator light faults. Training on the bus 17 UV cabinet incident was subsequently planned for I&C technicians and electrical maintenance personnel to preclude this practice.

2.3 Water Leakage Into the Residual Heat Removal System Pump Room

The licensee has experienced water leakage into the residual heat removal (RHR) system pump room (located in the sub-basement of the Auxiliary Building) for several years. Water has been leaking from the seam between the top of the RHR pump room west wall and the bottom of the Auxiliary Building basement floor. The leakage rate is very low, i.e., less than 0.1 gallon per day; however, as a result of the long term leakage, hard scale deposits (several inches thick) have formed on the wall. The inleakage is currently too small to represent any operational safety concern for the plant.

All potential sources of the leakage have not yet been positively identified. by the licensee. Ground water appears to be entering the room; however, RG&E's analyses also indicate the presence of boric acid and radionuclides that are also present in the spent fuel pool. The licensee is removing the scale buildup from the area around the seam to facilitate better collection of the water before it flows down the wall. The inspector will continue to monitor licensee efforts to quantify the water inleakage rate and to positively identify all sources. (IFI 50-244/95-15-01).

3.0 ENGINEERING (71707, 37551)

3.1 Service Water Pump Motor Failure and Subsequent Upgrade

On August 9, 1995, the licensee performed a normal biweekly transfer of operating service water (SW) pumps to place the D-SW pump (one of four installed) in service and to remove the B-SW pump (both connected to the same power supply). Approximately 10-15 seconds after starting the D-SW pump, the auxiliary operator stationed at the pump noticed smoke and flames emanating from the motor and immediately notified the control room operators to secure the pump. The pump was effectively secured, and the smoke and flames at the motor ceased. No apparent effects resulted from this event in other plant equipment connected to the same power supply, and no apparent damage resulted to any equipment located adjacent to the D-SW pump. The Ginna technical specifications do not require entry into an LCO action statement after the loss of one service water pump, or until one complete train of service water is lost. By design, only one service water pump and train is required for accident mitigation at Ginna.

The licensee's initial root cause analysis indicated that an internal short apparently resulted from excessive vibration of the motor windings that degraded the winding insulation and caused localized overheating to spread through the winding coils. The overheating eventually cause an internal short to ground in the motor. The incident represented a near identical electrical short and failure of the D-SW pump motor that occurred approximately two years earlier. In that instance, a short occurred in nearly the same location, but resulted in a phase-to-phase motor winding burnout. Both instances were among a series of service water pump and motor failures that have occurred at Ginna over several years. The licensee initially intended to return the motor to the original manufacturer (Westinghouse) for an in-depth root cause analysis since a problem was suspected with the materials and processes used the last two times the motor was rewound (1993 and 1994). However, the motor remains onsite pending a management decision to rewind the failed motor or purchase a new one.

The licensee did not have a spare motor that was immediately available for replacement, and a search for motors nationwide resulted in no suitable motors available in a short time period. A rewind of the failed motor would have taken a few weeks, and purchase of a new safety-related motor was expected to take several months. The licensee had previously procured an alternate replacement motor that was located in the site warehouse. However, the motor was a commercial procurement item intended for installation and dedication during the next refueling outage in March/April 1996. The licensee determined that the spare motor could be inspected, installed, and dedicated to safety-related service as a replacement for the failed motor. Since limited test data that could be validated was available from the motor vendor (U.S. Motors), and since the new motor was not identical to the failed motor (350 H.P. and 460 VAC new vs. 300 H.P. and 440 VAC old), the licensee initiated internal engineering actions to analyze all of the known new motor characteristics, to determine the extent of dedication testing necessary, and to perform 10 CFR 50.59 safety evaluations of any plant modifications necessary to connect the new motor to safeguards bus 17 and perform dedication testing during plant operations.

The licensee developed Plant Change Request 95-046 and Design Analysis DE-EE-95-129-06 to evaluate the potential effects on installed plant equipment of the D-SW pump replacement. The analysis reviewed the critical motor characteristics such as the inrush, full load, and fault currents; motor acceleration time; steady state and dynamic responses during safeguards sequencing; and degraded voltage performance during normal and accident conditions. The analysis also considered the effects of the new motor on the existing power circuit breaker, the safeguards bus 17, and the B-EDG under worst case loading scenarios. RG&E concluded hat the new motor's expected performance under accident conditions was bounded by existing safety analyses, and was based on thorough analysis that was validated through testing. The inspector considered the analysis to be in-depth, and comprehensive in addressing concerns related to the safe installation and testing of the motor with the plant at power.

After a seismic analysis concluded the new motor was acceptable for use, it was installed on the D-SW pump. Some modifications were made to the power cables and the bearing temperature instrument wiring. Complex test instrumentation was installed on the motor to obtain performance data and to permit detailed engineering analysis of the test results. Preliminary motor testing was performed with the motor uncoupled from the D-SW pump in order to obtain accurate inertial and torque reaction data, and to validate the licensee's motor performance model. The motor was then coupled to the pump and tested under normal operating conditions for an extended period. The test results indicated that the motor performed more efficiently, drew less current, and operated at a lower temperature than the old motor. The hydraulic characteristics of the D-SW pump were unchanged with the new motor.

The inspector concluded that the licensee made a sound decision to replace the D-SW pump motor and restore the pump to operability in a timely manner through dedication testing. The new motor's performance was thoroughly modeled and reviewed, with conservative engineering assumptions and ample consideration for the existing plant design and safety evaluations.

3.2 Steam Generator Replacement Project

The licensee is planning to replace both steam generators (SGs) during the next refueling outage, currently scheduled to begin in March 1996. A special

crane will be used to move the SGs into and out of containment through two openings that will be made in the containment dome. The dome will then be restored and a full-strength pressure test of the containment building will be performed. The old SGs will be stored on site in a newly constructed temporary storage facility.

The inspectors are monitoring the licensee's preparations for SG replacement and are reviewing the effectiveness of RG&E's project management controls, the safety evaluations prepared for the project, and the engineering specifications associated with the new SGs. Major construction activities for the project to date have included:

- Installation of the two concrete foundations for the SG lift crane (a Lampson Transi-Lift crane).

- Fabrication of a full-thickness containment dome mockup. This structure is sufficiently large to accommodate the actual size and geometry of one opening as it will be made in the containment dome. The construction details of this mockup (steel-reenforced concrete with a steel liner) closely simulate the actual containment dome. The mockup will be used for testing, refining, and proving excavation and reconstruction techniques, and will also be used for personnel training.

- Construction of the Old SG Storage Facility (OSGSF).

- Rerouting of various site services and construction of several onsite support facilities.

The NRR and Region I-based Project Managers visited the site to obtain an overview of the SG replacement project (SGRP) and to coordinate inspection plans with the NRC Senior Resident Inspector. Discussions were held with RG&E and Bechtel managers, supervisors, craftworkers and quality assurance personnel on the project team. Procedures and specifications related to the project were sampled for review. Work in progress and sites of planned work were observed. From these discussions, reviews and observations, it was found that the SGRP is being conducted by a team of utility and contractor personnel, using the experiences gained through completion of similar projects, in preparation of detailed plans and procedures to accomplish the SG replacement. The work in progress showed careful attention to detail; for example, the manufacturing of the new SGs was being covered by a full time licensee quality inspector in residence at the fabrication plant.

Aspects of the project that were reviewed or observed included an overview of the project, the lifting equipment (Lampson and tower cranes), the containment dome mockup, dome cutting plans, welding training/qualification plans, the Lampson crane foundations, a video/computer-photographic tour of containment in the project affected areas, the measurement method, controls on SG fabrication, the OSGSF, SG transportation to the site, secondary side water chemistry, Quality Assurance/Quality Control involvement, the project library, and overall project control.

The NRR Project Manager obtained and reviewed several licensee safety evaluations and design criteria, including:

•	SEV 1019	Containment Structural Modifications
•	SEV 1024	DRAFT SG Rigging and Handling.
•	SEV 1021	SG Vessel, Piping, and Insulation
•	SEV 1025	Temporary Utilities, Services and Shielding
•	SEV 1018	Facilities Outside Containment
•	BWNT 77-1235965-01	Replacement Steam Generator Safety Evaluation
•	DC 10034A	Replacement Steam Generator Design Criteria - Containment Structural Modifications

The licensee is required to submit to the NRC, for review and approval, any change that constitutes an unreviewed safety question or requires a change in technical specifications (TSs). During this inspection visit, no unreviewed safety questions or required changes to TSs were identified. If the containment access is enlarged, design reviews should be coordinated with NRR/ECGB, via the Project Manager, to confirm the restored design margin. If necessary, this will be performed by NRR in the future during the licensee's structural integrity test (SIT) and integrated leak rate test (ILRT) on the reconstructed containment.

No areas of concern were identified during review of the SGRP. The licensee was well prepared to address questions posed during the discussions.

4.0 PLANT SUPPORT (71750)

4.1 Radiological Work Controls and External Exposure Monitoring in Radiologically Controlled Areas

During recent months, several incidents have occurred where personnel entering a radiologically controlled area (RCA) in the plant did not log into the licensee's radiation work permit (RWP) access control system, and/or did not have the secondary alarming dosimetry required for entry. None of the incidents involved entry without the primary personnel exposure monitoring devices (thermoluminescent dosimeters); however, all represented a failure of qualified radiation workers to adhere to the procedure requirements pertaining to RCA access, RWP work controls, and possession of alarming dosimetry. Three of the documented incidents are described as follows.

On April 21, 1995, during the last refueling outage, a qualified radiation worker entered the containment enclosure (a posted high radiation area) to inspect work on reactor coolant system piping. Upon exiting the containment approximately one hour later, he realized that he entered the containment without the required secondary dosimetry (ALNOR) and without signing in on the

applicable RWP. The radiation protection (RP) technician on duty subsequently initiated an action report (AR 95-082) to investigate the cause of this incident and to initiate the appropriate corrective actions. The worker's exposure during the containment entry was temporarily estimated from the ALNOR dosimeter reading (4 mR) of a coworker who accompanied him during his entire stay inside containment. Subsequent analysis of the worker's TLD confirmed the 4 mR exposure. RG&E's immediate actions to address this incident included temporarily restricting the individual's access to restricted areas, conducting interviews with other individuals in the same work group, reviewing the incident with all RP personnel, and posting signs at the RCP access point to remind workers of the RWP and dosimetry requirements for RCA entries.

RG&E's RP department reviewed this incident and concluded that the Technical Specification (TS) requirements (Section 6.13) were satisfied since the TSs allow for a group of individuals to use a single alarming dosimeter inside a high radiation area. Also, no work had apparently been performed outside the restrictions of the applicable RWP. However, several plant procedures require that all individuals entering restricted areas (i.e., RCAs) shall log into the RWP access control system and obtain secondary alarming dosimetry. Procedure A-1, "Radiation Control Manual," stipulates that workers must acknowledge all RWP requirements by logging into the access system and obtaining a secondary dosimeter that must be worn continuously by each individual while inside a restricted area. Also, procedures A-1.3, "Restricted Area Entry and Exit," and A-1.8, "Radiation Work Permits," both require worker acknowledgement of the radiological controls associated with their job by logging into the RWP system and obtaining alarming dosimetry prior to RCA entry. AR 95-082 was closed in August 1995, and recommended a longer term corrective action to investigate a possible link between the RWP and security access systems to prevent unauthorized access to the RCA. This was later investigated by the licensee, but was not implemented because the two systems are not physically compatible.

On June 27, 1995, an RP technician discovered that a qualified radiation worker had entered the RCA to work in the Intermediate Building without his required dosimetry (RADOS). The worker had signed into the RWP access system, indicating he had read, understood, and would comply with the RWP requirements applicable to his work assignment. However, he subsequently entered the RCA without taking the RADOS dosimeter with him. After approximately five minutes, the RP technician on duty noticed his dosimeter still at the access control desk and paged the worker to return to the entry point. The RP technician allowed the worker to return to the RCA after giving him the dosimeter, and counselling him on the RWP and procedure requirements for possessing a RADOS when entering the RCA. The RP technician then initiated an action report (AR 95-166) to investigate the incident and to pursue corrective actions. As of this inspection, the final disposition of this AR was not complete.

On August 3, 1995, a qualified radiation worker self-identified that he had entered and worked in a high radiation area inside the Auxiliary Building. After receiving a pre-job briefing from an RP technician, the individual entered the high radiation area without his required RADOS dosimeter. Prior to entering the RCA, the individual had donned anti-contamination clothing,

but did not register into the RWP electronic access system. Upon exiting the RCA, the individual discovered his missing dosimeter and immediately discussed the situation with RP personnel and his supervisor. The worker subsequently initiated an action report (AR 95-0240) to investigate the causes for this incident and to pursue corrective actions. RP personnel were able to estimate the worker's dose from the job duration and the radiation levels in his work area. His actual exposure was later confirmed by reading his TLD. As of this inspection, the final disposition of AR 95-0240 was not complete.

On August 3, 1995, RG&E initiated a Human Performance Evaluation System (HPES) report (HPES 95-0240) to perform an in-depth root cause analysis of these three incidents and to determine the necessary corrective actions to prevent recurrences. HPES 95-0240 was not complete as of this inspection; however, the licensee's detailed comparison between the security and RWP access logs indicated that twenty instances of personnel entry into the RCA actually occurred without the required secondary dosimeter between January 1, 1995, and August 6, 1995. Most of these instances also involved a failure to log into the RWP system prior to RCA entry. The draft HPES indicated that a turnstile barrier could be installed to prevent RCA access without the necessary entries into the RWP and RADOS systems. This initiative is considered to be a positive measure by the licensee. However, the final root cause evaluation and disposition of ARs 95-166 and 95-0240, and the HPES are still ongoing.

Although of minor radiological consequence, these incidents represent a relatively high occurrence over a seven month period of a lack of adherence to procedure requirements for radiological work controls and access to radiation and high radiation areas. This item is unresolved pending NRC review of the completed HPES 95-240 report and the licensee's implementation of permanent corrective actions. (URI 50-244/95-15-02)

5.0 SAFETY ASSESSMENT/QUALITY VERIFICATION (71707)

5.1 Periodic Reports

Periodic reports submitted by the licensee pursuant to Technical Specification 6.9.1 were reviewed. The inspectors verified that the reports contained information required by the NRC, that test results and/or supporting information were consistent with design predictions and performance specifications, and that reported information was accurate. The following reports were reviewed:

- Monthly Operating Reports for July and August 1995

No unacceptable conditions were identified.

5.2 Licensee Event Reports

Two Licensee Event Reports (LERs) submitted to the NRC were reviewed and the inspector determined that the details were clearly reported, the causes were 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 LERs were reviewed (Note: date indicated is event date):

- 95-006, Loss of 34.5 KV Offsite Power Circuit 751, Due to Offsite Lightning Strike, Results in Automatic Start of "A" Emergency Diesel Generator (June 30, 1995)

- 95-007, Loss of 34.5 KV Offsite Power Circuit 751, Due to Offsite Electrical Storm, Results in Automatic Start of "B" Emergency Diesel Generator (August 3, 1995)

The inspector concluded the LERs met regulatory requirements and appropriately evaluated the safety significance of the events. LERs 95-006 and 95-007 are closed.

6.0 ADMINISTRATIVE

6.1 Senior NRC Management Site Visits

During this inspection period, three senior NRC managers visited Ginna Station. On August 16-17, 1995, Mr. James C. Linville, Chief of Reactor Projects Branch 3, toured the site and met with senior licensee management. On August 22, 1995, Mr. Thomas T. Martin, Regional Administrator for NRC Region 1, toured the site and met with senior licensee management. On August 23-24, 1995, Mr. William J. Lazarus, Chief of Reactor Projects Section 3B, toured the site and met with senior licensee management.

6.2 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 the current resident inspection report 50-244/95-15 was held on September 11, 1995.