U.S. NUCLEAR REGULATORY COMMISSION
REGION I

License No.	DPR-18
Report No.	50-244/99-03
Docket No.	50-244
Licensee:	Rochester Gas and Electric Corporation (RG&E)
Facility Name:	R. E. Ginna Nuclear Power Plant
Location:	1503 Lake Road Ontario, New York 14519
Inspection Period:	April 5, 1999 through May 16, 1999
Inspectors:	P. D. Drysdale, Senior Resident Inspector C. C. Osterholtz, Resident Inspector W. A. Cook, Project Engineer
Approved by:	M. G. Evans, Chief Projects Branch 1 Division of Reactor Projects

EXECUTIVE SUMMARY

R. E. Ginna Nuclear Power Plant
NRC Inspection Report 50-244/99-03

This inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers a 6-week period of resident inspection and the results of a special review of the licensee's Year 2000 readiness project by a Region-based specialist.

Operations

A human performance error resulted in an automatic reactor shutdown from 35% power on April 23, 1999. The licensee effectively evaluated the trip for its principle root causes prior to unit restart. An event investigation was appropriately initiated to examine additional human performance issues related to the event.

The licensee effectively identified and corrected a failed reactor protection system bistable which resulted in an automatic reactor shutdown on April 27, 1999. Operator response to this plant transient was good.

Maintenance

Observed maintenance and surveillance activities were accomplished in accordance with procedure requirements, except for missing signatures in the procedure used to replace the delta temperature math module. That incident was properly entered into the licensee's corrective action system for resolution. The licensee's post-maintenance testing was adequate to demonstrate the operability of equipment prior to its return to service.

The licensee successfully replaced failed resistance temperature detectors (RTDs) in the reactor coolant system, and operations personnel performed well in conducting a controlled drain-down and refill of the reactor coolant system to accommodate the replacement. The Plant Operations Review Committee's initial recommendation to close the RTD leakage ACTION Report (99-0751) without completing a root cause determination was an example of ineffective corrective action.

Engineering

The licensee was challenged with operational problems associated with a new plant modification that caused over-temperature and over-power delta temperature setpoints to drift. The licensee effectively identified and corrected some problems, and conservatively reduced reactor power while troubleshooting was in progress. An inspection follow-up item will track the licensee's review of root cause(s) and corrective actions associated with this modification (IFI 50-244/98-03-01).

The licensee's actions were effective in reducing emergency diesel generator delta peak firing pressure and peak firing pressure to below the manufacturer's stated maximums.

Executive Summary (cont'd)

Plant Support

Overall, the licensee's radiological work and boundary controls inside containment during the recent outage were effective in maintaining personnel exposures and contaminations at reasonably low levels. The pre-outage exposure and contamination goals were slightly exceeded at the end of the outage; however, the licensee maintained an acceptable level of oversight and control in this area.

TABLE OF CONTENTS

Table of Contents (cont'd)

ATTACHMENTS

Report Details

I. Operations

O1 Conduct of Operations[1]

01.1 General Comments (Inspection Procedure (IP) 71707)

The inspectors observed plant operations to verify 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. The inspectors conducted ongoing verifications of engineered safety feature (ESF) system operability, verifications of proper control room and shift staffing, verification that the plant was operated in conformance with the improved technical specifications (ITS) and appropriate action statements were implemented for out-of-service equipment, and verification that logs and records accurately identified equipment status or deficiencies.

O2 Operational Status of Facilities and Equipment

O2.1 Summary of Plant Status (71707)

The plant was in MODE 6 refueling operations at the beginning of the inspection period. On April 9, 1999, refueling operations were completed, the primary loops were refilled, and MODE 5 was entered. On April 12, 1999, the licensee issued a four-hour notification to the NRC in accordance with 10 CFR 50.72 upon discovering that the containment recirculation fan cooler (CRFC) moisture separator vanes were installed backwards, placing the plant in an unanalyzed condition (see section O8.2).

On April 13, 1999, the licensee issued a four-hour notification to the NRC in accordance with 10 CFR 50.72 after an automatic start of the B-emergency diesel generator (EDG) inadvertently occurred during restoration from a functional test of the safety injection system (see section O8.3). Also on April 13, the licensee redrained the primary coolant loops back to six inches to replace three new resistance temperature detectors (RTDs) installed during the recent outage that had not performed as anticipated (see section E2.1). During the primary system hydrostatic test on April 16, 1999, the replaced RTDs were observed to be leaking and were subsequently seal welded to stop the leakage. The plant entered MODE 4 operations on April 18, and the plant heatup continued. The plant entered MODE 3 on April 19.

On April 19, 1999, a reactor startup was commenced and the plant entered MODE 2 operations (reactor critical) at 2:23 a.m. on April 20. Also on April 20, a small steam leak on the A-train main steam isolation valve occurred. The licensee generated an ACTION Report (99-0806) and was able to seal weld the valve flange to stop the leak. The plant

---

Topical headings such as O1, M8, etc., are used in accordance with the NRC standardized reactor inspection report outline. Individual reports are not expected to address all outline topics.

entered MODE 1 at 5:06 a.m. on April 22. On April 21, the licensee issued a four-hour notification to the NRC in accordance with 10 CFR 50.72 after an inadvertent automatic start of the turbine driven auxiliary feedwater (TDAFW) pump occurred. The licensee generated an ACTION Report (99-0794) to address the issue and commenced an event investigation for the root cause. The inspectors noted that this event resulted in a minor temperature decrease with the RCS moderator temperature coefficient still slightly positive. The licensee indicated that a Licensee Event Report (LER) would be issued sometime in May 1999.

On April 23, 1999, an automatic reactor trip occurred with the plant at approximately 35% power when instrumentation and control (I&C) technicians inadvertently pulled fuses from the wrong nuclear instrument channel after another channel had been defeated for setpoint adjustments (see section O2.2). Control room operators appropriately entered emergency operating procedures (EOPs) and stabilized the plant in MODE 3. The licensee issued a four-hour report to the NRC in accordance with 10 CFR 50.72. The trip was uncomplicated and a reactor startup was performed the same day. The plant entered MODE 2 operations at 5:30 p.m. that afternoon, and MODE 1 was entered at 8:42 p.m. that evening.

On April 27, 1999, an inadvertent automatic reactor trip occurred from approximately 90% power when a bistable failure occurred in reactor protection system (RPS) channel 3 while RPS channel 2 was already in a tripped condition for calibration (see section O2.3). Control room operators appropriately entered EOPs and stabilized the plant in MODE 3, and the licensee issued a four-hour notification to the NRC in accordance with 10 CFR 50.72. The failed bistable was replaced and a reactor startup was commenced on April 29. The plant entered MODE 2 operations at 3:35 a.m., and MODE 1 was entered at 6:25 a.m. that day.

On May 3, 1999, the licensee made a four-hour notification to the NRC in accordance with 10 CFR 50.72 after entering improved technical specification (ITS) 3.0.3 upon determining that three of four overpower and overtemperature delta temperature instruments were inoperable due to an AC ripple current (see section E2.1). Power was reduced from approximately 90% to 21%. ITS 3.0.3 was exited at 4:47 a.m. on May 4 after the licensee replaced the Foxboro bistables with NUS bistables in channels 1 and 3.

On May 5, 1999, while at 97% power and performing a load increase to 100% power, control room operators received an automatic turbine runback from the overpower delta temperature function for approximately two seconds. The operators reduced power to approximately 94.5% and issued a four-hour notification to the NRC in accordance with 10 CFR 50.72. The licensee determined that the runback was caused by overpower and overtemperature delta temperature bistables setpoint drift. Reactor power was administratively limited to 98.5%, to increase the margin to a runback, while the setpoints were adjusted and additional troubleshooting was performed. The plant remained at 98.5% power through the end of the inspection period while the licensee continued to evaluate the overpower and overtemperature delta temperature problems.

O2.2 Reactor Trip from 35% Power During Nuclear Instrument Trip Setpoint Adjustments

a. Inspection Scope (71707)

The inspectors reviewed the licensee's response and follow-up actions for an automatic reactor trip due to a personnel error.

b. Observations and Findings

During the post-outage power ascension on April 23, 1999, with the plant at approximately 35% power, instrument and control (I&C) technicians were making preparations to adjust the trip setpoints in power range nuclear instrument (NI) channel 2 (N42). Channel N42 was initially defeated, but after a brief work interruption, the technician inadvertently went into the cabinet for power range channel N43 and removed its instrument power fuses. This resulted in two out of four channels providing signals to the RPS logic and resulted in an automatic reactor trip.

The licensee initiated ACTION Report 99-0801 and a "Level II" human performance evaluation to examine the causal factors related to the event. The inspectors reviewed the post-trip evaluation report presented to the plant operations review committee (PORC) and observed that the trip was adequately evaluated and understood by the licensee before the plant was returned to power. The report concluded that the trip occurred primarily as a result of poor self-checking on the part of the technician, and inadequate labeling on the nuclear instrumentation drawers. The licensee considered that an event investigation was necessary to further examine human performance errors associated with the trip. Final corrective actions were still being evaluated at the end of the inspection period, and the licensee indicated that a Licensee Event Report (LER) would be issued.

c. Conclusions

A human performance error resulted in an automatic reactor shutdown from 35% power on April 23, 1999. The licensee effectively evaluated the trip for its principle root causes prior to unit restart. An event investigation was appropriately initiated to examine additional human performance issues related to the event.

02.3 Reactor Trip from 90% Power Following Spurious Bistable Failure in the Reactor Protection System

a. Inspection Scope (71707)

The inspectors reviewed the licensee's response and follow-up actions for an automatic reactor trip due to an RPS bistable failure.

b. Observations and Findings

On April 27, 1999, with the plant at approximately 90% power, I&C technicians were performing calibration checks of RPS channel 2 when a spike on the channel 3 over-temperature delta temperature (OTΔT) module occurred. This resulted in the two of four RPS logic being satisfied and the automatic reactor shutdown. The inspectors observed control room operators enter emergency procedure E-0, "Reactor Trip or Safety Injection," and successfully stabilize the plant in MODE 3. The licensee generated an ACTION Report (99-0821) to investigate the trip. The inspectors reviewed and agreed with the post-trip evaluation report which was presented to the PORC.

The licensee performed troubleshooting on RPS channel 3 and concluded that the spike resulted from a failed bistable in the OTΔT module. I&C technicians were able to repeat the bistable failure during shop bench testing. The bistable was replaced and a plant startup was successfully performed on April 29, 1999. The licensee indicated that an LER would be generated for this event.

c. Conclusions

The licensee effectively identified and replaced a failed reactor protection system bistable which resulted in an automatic reactor shutdown on April 27, 1999. Operator response to this plant transient was good.

O8 Miscellaneous Operations Issues

O8.1 (Closed) Licensee Event Report (LER) 1999-003: Two Valves Declared Inoperable Results in Condition Prohibited by Technical Specifications

The licensee submitted LER 1999-003, on March 31, 1999, in response to a March 1, 1999 event in which the required torque to initiate valve disc closure ("breakaway torque") for the two main steam non-return check valves (CVs) was greater than the torque specified in test procedure PT-2.10.15, "Main Steam Non-Return Check Valve Closure Verification." The maximum breakaway torque allowed by PT-2.10.15 was 600 foot pounds (ft-lbs). The actual breakaway torque for the check valves was approximately 700 ft-lbs for CV-3518 and 900 ft-lbs for CV-3519. ITS Limiting Condition for Operation (LCO) 3.7.2.E.1 required immediate entry into LCO 3.0.3 if one or more valves in a flowpath from each steam generator were inoperable. Consequently, the licensee entered ITS LCO 3.0.3 at 5:34 p.m. on March 1. The plant was already in the process of shutting down and was in MODE 3 at the time of discovery. At 8:18 p.m. on March 1, the plant entered MODE 4 and LCO 3.0.3 was no longer applicable.

The licensee's engineering staff later concluded that the as-found torque values were acceptable, since the packing had been changed during the recent outage and required a greater breakaway torque to provide for less valve vibration and a better shaft seal. The engineering calculations showed that sufficient moment existed for the valves to shut during a design basis steam line break inside containment. The LER described the root cause of the event to be the change in material and methodology for repacking

these valves. The inspectors reviewed the calculations and had discussions with system engineers during an in-plant review of the LER. Corrective actions listed in the LER included establishing a new torque reference value in accordance with the ASME Code, and revising the applicable design analysis for the new torque acceptance criteria. In addition, the LER indicated that the licensee would revise PT-2.10.15 once a new reference value for breakaway torque could be established. The inspectors considered that the LER adequately described this event and proposed sufficient corrective actions to address check valve performance under design basis conditions. The licensee intended to submit the details of additional follow-up actions in a supplement to the LER. This LER is closed.

O8.2 (Closed) LER 1999-004: Containment Recirculation Fan Moisture Separator Vanes Incorrectly Installed Results in Unanalyzed Condition

LER 1999-004 was submitted to the NRC on May 12, 1999, following the licensee's discovery that the moisture separator louvers downstream from the cooling coils in all four containment recirculation fan cooler units (CRFCs) were installed backwards during their original fabrication in the 1960s. The licensee confirmed that the CRFC manufacturer installed the louvers backwards prior to their installation at the site. The unit was shutdown in MODE 5 at the time of discovery and the CRFCs were not required to be operable.

Analysis demonstrated that following a design basis loss of coolant accident (LOCA), the backwards configuration would have removed most of the entrained moisture in the CRFC air flow before it reached the high efficiency particulate air (HEPA) filters or charcoal filter beds. The licensee could not precisely determine the moisture removal efficiency in that configuration. However, the results of a preliminary qualitative analysis indicated that the fan coolers would have still performed their intended design function. That conclusion was based upon the fact that overall moisture removal in the CRFCs is more dependent upon the cooling coil design, and that additional media were present in the flow path which would also remove entrained moisture upstream of the HEPA and charcoal filters. Also, the length and spacing of the moisture separator louvers has a larger effect on their efficiency, and the backwards configuration still provided a tortuous flow path. The inspector considered that the licensee's preliminary basis for CRFC operability was reasonable. Prior to startup from the refueling outage, the licensee modified all four CRFCs to reconfigure the louver vanes in accordance with the original design. Additional analysis was ongoing and the licensee intended to provide a conclusive evaluation of past operability in a future supplement to the LER. This LER is closed.

O8.3 (Closed) LER 1999-005: Under Voltage Signal on Safeguards Bus During Testing Results in Automatic Start of "B" Emergency Diesel Generator

LER 1999-005, dated May 13, 1999, documented an event that occurred on April 13, when the B-emergency diesel generator (B-EDG) automatically started upon sensing an under-voltage condition on safeguards bus 17. The unit was shutdown in MODE 5 and the licensee was performing Refueling Shutdown Surveillance Procedure RSSP-2.1A,

"Safety Injection Functional Test Alignment/Realignment." Bus 17 normally supplies power to two safety-related service water pumps, the electric fire water pump, and miscellaneous non-safety related electrical loads.

The inspectors determined that during the post-test restoration, operations personnel attempted to re-energize the bus, but its normal supply breaker did not close. The plant staff initiated troubleshooting on the supply breaker and mistakenly proceeded with further restoration of equipment involved in the test. Event review identified that RSSP-2.1A required that the B-EDG be placed in lockout and that all other vital loads powered by bus 17 be de-energized. A subsequent restoration step required the B-EDG to be reset from its lockout condition. Upon execution of this step, the diesel automatically started upon sensing the under-voltage condition on bus 17. No safeguards equipment loaded onto bus 17 after the diesel started because the under-voltage condition was not cleared with the normal supply breaker still open. The licensee later determined that the supply breaker failure was related to a blown fuse in a control power circuit.

The licensee attributed this event to a cognitive personnel error on the part of the test team. When the bus 17 breaker failed, the test coordinator decided to restore as much equipment as possible while the breaker troubleshooting proceeded. The corrective actions identified in the LER included a revision to procedure RSSP-2.1A to verify that associated buses are energized prior to resetting an EDG. Corrective actions also included performing enhanced training in team communications for operations personnel. The inspector was observing the test in the control room, at the time of this event, and discussed the system configuration and plant conditions with operators. The LER adequately described the root causes and corrective actions associated with the event. This LER is closed.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Maintenance Activities

a. Inspection Scope (62707)

The inspectors observed all or portions of several maintenance work activities.

b. Observations and Findings

The inspectors observed portions of plant maintenance activities to verify that the correct parts were used; the applicable industry codes and technical specification requirements were satisfied; and adequate measures were in place to ensure personnel safety and prevent damage to plant structures, systems, and components.

- W.O. 19901038: Safety injection (SI) system relay 431I replacement (observed April 4, 1999); and troubleshooting of SI unblock circuit problems (observed on April 11, 1999).

The inspector observed the system engineer's presentation of SI system operability to the PORC on May 13, 1999. The licensee concluded that the SI function would have been available following any actuation signal, in spite of the error introduced during the modification process (SM95-092.2). The inspector reviewed the licensee's operability evaluation and agreed with this conclusion.

- W.O. 19800909: Install new containment isolation relays (observed April 11, 1999)

- W.O. 19801591: Troubleshoot/calibrate core exit thermocouples/drawer (observed April 11, 1999)

- W.O. 19901721: Replace ΔT math modules with repaired spares, check the OTΔT and OPΔT bistables, and perform ΔT span adjustments on Channel 4 (observed on May 10 and 11, 1999).

The inspector identified that the I&C procedure (CPI-TRIP-TEST-5.40) was completed and signed-off satisfactory with one page missing that contained steps for recording setpoint data and double verification signatures that the channel was properly restored to service. When the administrative controls for equipment out-of-service were subsequently cleared on the Channel 4 ΔT modules, the verification signatures were missing from the procedure. The I&C shop foreman questioned the technicians who performed the work and confirmed that the test data was taken properly, and that the double verifications were performed when the channel was restored to operability. Notwithstanding, the failure to properly document the completion of these test procedure steps is a violation of regulatory requirements. However, this failure constitutes a violation of minor significance and is not subject to formal enforcement action. The licensee generated ACTION Report 99-0876 to investigate this administrative oversight.

c. Conclusions

The inspectors concluded that observed maintenance activities were accomplished in accordance with procedure requirements, except for the replacement of the ΔT math module. That incident was properly entered into the licensee's corrective action system for resolution. The licensee's post-maintenance testing was adequate to demonstrate the operability of equipment prior to its return to service.

M1.2 Surveillance Activities

a. Inspection Scope (61726)

The inspectors observed portions of surveillance activities.

b. Observations and Findings

The inspectors observed selected surveillance tests to verify that approved procedures were in use, procedure details were adequate, test instrumentation was properly calibrated and used, technical specifications were satisfied, testing was performed by knowledgeable personnel, and test results satisfied acceptance criteria or were properly dispositioned. The following surveillance tests were observed:

- RSSP-2.1A, "Safety Injection (SI) Functional Test Alignment/Realignment," (observed on April 23, 1999)
- PT-12.2, "Emergency Diesel Generator B," (observed on April 26, 1999)
- PT-12.1, "Emergency Diesel Generator A," and PT-12.2, "Emergency Diesel Generator B," engine diagnostics (observed on May 11, 1999)
- PT-2.1Q, "SI System Quarterly Surveillance Test," (observed May 12, 1999)

c. Conclusions

The licensee's test procedures contained adequate details for accomplishing test requirements. Testing was performed by knowledgeable personnel, and test instrumentation was properly calibrated. All test results satisfied procedure acceptance criteria or were properly dispositioned, and technical specifications surveillance requirements were properly satisfied.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.2 Reactor Coolant System Resistance Temperature Detector (RTD) Leakage and Replacement

a. Inspection Scope (62707)

The inspectors reviewed licensee activities to replace RTDs for the determination of primary system average temperature (Tave).

b. Observations and Findings

During the past refueling outage, the licensee performed a modification (PCR 98-024) to the RPS that included relocating and replacing an RTD for channel 2, and replacing two hot leg RTDs in channel 1 (see section E2.1). Repositioning the channel two RTD made its alignment consistent with the other channels, and allowed for optimum temperature measurement across the hot leg. The new Conax RTDs in channel 1 and channel 2 replaced the Rosemount RTDs, which had a slightly deeper immersion depth in the RCS piping. On April 11, 1999, after refilling the RCS loops and restarting the reactor coolant pumps (RCPs), a channel 1 Conax RTD (TE-401) failed and the channel 2 Conax RTD (TE-402) provided intermittent resistance readings. The licensee concluded that the new

RTDs did not provide the expected indication and decided to drain the RCS back down to the six-inch level to replace all three Conax RTDs with Rosemount RTDs from stock.

On April 13, 1999, control room operators performed the RCS drain-down in accordance with Significant Infrequently Performed Evolution (SIPE) 99-04, "RCS Loop Hot Leg RTD Replacement." The procedure provided adequate guidance to perform the evolution in a controlled manner, and the drain-down was performed well. The RTDs were replaced and the RCS was refilled without incident. The removed Conax RTDs showed no visual sign of degradation and were returned to the manufacturer to determine the cause of the failure. On April 16, 1999, during performance of PT-7, "ISI System Leakage Test - Reactor Coolant System," the licensee noted that the three replacement Rosemount RTDs were weeping RCS coolant. The licensee generated an ACTION Report (99-0751) and attempted to re-torque the RTDs to their maximum allowed value (65 ft-lbs). The RTDs continued to weep, and the RTDs were seal welded on April 18 to eliminate the leakage. The inspectors noted that the PORC recommended closure of ACTION Report 99-0751 without requiring a root cause analysis for the leakage, since the RTDs were seal welded and could not be examined for thread or seal defects. The inspectors were concerned that the RCS pressure boundary leakage was common to all three replaced RTDs, and that all other RTDs installed in the RCS were not seal welded. The licensee subsequently reopened ACTION Report 99-0751 and initiated a root cause analysis.

c. Conclusions

The licensee successfully replaced failed resistance temperature detectors (RTDs) in the reactor coolant system, and operations personnel performed well in conducting a controlled drain-down and refill of the reactor coolant system to accommodate the replacement. PORC's initial recommendation to close the RTD leakage ACTION Report (99-0751) without completing a root cause determination was an example of ineffective corrective action.

III. Engineering

E2 Engineering Support of Facilities and Equipment

E2.1 (Open) Inspector Follow-up Item (IFI) 50-244/99-03-01: Reactor Protection Module Modifications for Primary System Temperature, and Primary Resistance Temperature Detector (RTD) Replacements

a. Inspection Scope (37551)

The inspectors reviewed a modifications to RPS circuit modules and RTD replacements, and their post-modification testing to confirm operability.

b. Observations and Findings

During the recent refueling outage, the licensee performed two modifications, PCR 97-026, to replace RTD input processing modules for reactor protection channels 1, 3, and 4 (channel 2 was replaced prior to the 1999 refueling outage) and PCR 98-024, to reposition and replace an RCS hot leg RTD for channel 2, and to replace two hot leg RTDs for channel 1. The module replacement was intended to allow for continuous primary system average temperature (Tave) and delta-temperature (ΔT) signal processing. The previous processor utilized a resistance bridge circuit to generate the Tave and ΔT signals. The new modification averaged the hot leg and cold leg inputs for generating a reactor protection signal. If one RTD should fail after the modification, its input could be removed from the circuit and replaced by doubling the input from an operable RTD. Repositioning the channel 2 RTD and replacing the channel 1 RTD were designed to prevent inadvertent automatic rod motion due to the effects of Tave "streaming" (see IRs 50-244/97-12 and 50-244/98-08). The safety evaluation for relocating the hot leg RTD for channel 2 supported the licensee's conclusion that this modification was within the existing design basis of the plant.

During calibration checks on the overpower delta temperature (OPΔT) and OTΔT setpoints in channel 1, on May 3, 1999, the licensee discovered that the setpoints deviated from the required value by approximately 5%. The maximum allowed deviation had been previously analyzed at 4.2%. The licensee also discovered out-of-specification setpoints on channel 3 and 4 for OPΔT and OTΔT. The licensee declared RPS channels 1, 3, and 4 inoperable, entered ITS 3.0.3, and commenced decreasing power at 10% per hour. Troubleshooting revealed that a 500kHz AC ripple of approximately 600 mV was present on top of the normal 0-10 VDC bistable output. The inspector determined from the licensee that the modification had provided for NUS modules to be installed with the existing Foxboro bistables. The licensee contacted the NUS module design representative for assistance, as the modification performed on channel 2 during the previous outage showed no similar anomalies. The licensee then decided to replace the Foxboro bistables in channels 1, 3, and 4 with NUS bistables, which included an AC filter and were considered to be more compatible with the NUS modules. The licensee replaced the bistables and exited ITS 3.0.3 on May 4, 1999. However, some AC noise was still present on the newly installed bistables.

Upon returning to approximately 97% power, on May 5, 1999, control room operators received an automatic turbine runback on OPΔT for approximately two seconds and stabilized reactor power at approximately 94.5%. The licensee subsequently discovered that the bistable setpoints had drifted (in the conservative direction), due to a change in impedance that was induced between the modules and the bistables after the NUS bistables were installed. The setpoints were recalibrated and the licensee administratively restricted maximum power to 98.5%, while monitoring the setpoints for drift and evaluating a method for eliminating the remaining AC noise (≈50mV). The licensee held a conference call with NRC on May 7, 1999, to discuss problems with the OPΔT and OTΔT modifications. The NRC staff considered the licensee's actions to administratively restrict the maximum reactor power to 98.5% while troubleshooting was

a conservative decision. Troubleshooting activities were still in progress at the end of the report period.

c. Conclusions

The licensee was challenged with operational problems associated with a new plant modification that caused over-temperature and over-power delta temperature setpoints to drift. The licensee effectively identified and corrected some problems, and conservatively reduced reactor power while troubleshooting was in progress. An inspection follow-up item will track the licensee's review of root cause(s) and corrective actions associated with this modification **(IFI 50-244/98-03-01)**.

E8 Miscellaneous Engineering Issues

E8.1 (Closed) IFI 50-244/97-12-02: Determine Appropriate Operability Parameters for the Emergency Diesel Generators

In December 1997 the inspectors noted two emergency diesel generator (EDG) performance parameters that could have affected engine operability if the limits stated by the manufacturer (ALCO) were exceeded (see IR 50-244/97-12). The engine technical manual indicated that the difference between the maximum and minimum (delta) peak firing pressure for the entire engine should not exceed 150 pounds per square inch (psi) at full rated load. In addition, an ALCO engine design representative (Coltec Industries) stated that the maximum firing pressure for the engines at the Ginna Station should not exceed 2000 psi. The inspectors had observed an increasing trend in the delta peak firing pressures on both engines over the past three years, and in all diagnostic tests between March 1996 and December 1998 delta peak firing pressure for both engines exceeded 150 psi. Also during testing in March 1996, the peak firing pressure in one cylinder on the A-EDG had achieved 1925 psi. This inspection follow-up item was opened because the licensee did not have a formal evaluation or plan to address the effects on operability of continuously testing and operating the EDGs beyond the delta peak firing pressure limitations, or for exceeding the peak firing pressure of 2000 psi. It appeared to the inspectors that both parameters could represent an operability concern that was not recognized in the licensee's procedures, and that no instructions to operators or test personnel were provided to avoid exceeding that value.

The licensee conducted diagnostic tests of both EDGs in December 1998. The test procedures were revised to set engine load at 2000 kilowatts (KW), allowing for more consistent test data and more meaningful results for trending purposes. Testing at the rated load also kept the cylinder peak firing pressures in the 1600-1800 psi range. However, the test data from December 1998 indicated that the delta peak firing pressure still exceeded 150 psi on both engines. During subsequent discussions with a Coltec representative, the inspectors determined that exceeding 150 psi delta peak firing pressure alone would not be a sufficient basis to declare the EDGs inoperable. However, the 150 psi limit was characterized as an important specification for preserving engine life, and that the licensee should take positive actions to prevent operation above 200 psi.

During the recent refueling outage (March - April 1999), the licensee performed preventive maintenance and inspections on both EDGs. The work was conducted with the assistance of a Coltec Industries performance analyst who consulted the licensee on the existing mechanical condition of the engines, and who recommended appropriate repairs and adjustments during the maintenance. In an effort to achieve better and more consistent combustion characteristics in all of the engine cylinders, the licensee made valve clearance adjustments, fuel pump timing adjustments, and replaced several fuel injectors. The licensee revised test procedures PT-12.1 "Emergency Diesel Generator A," and PT-12.2 "Emergency Diesel Generator B," to include explicit acceptance criteria for the 150 psi limit on delta peak firing pressure and for 2000 psi on the maximum firing pressure. The post-maintenance tests showed a significant improvement in the overall balance of both engines (the maximum delta peak pressure was 70 psi on the A-EDG, and 80 psi on the B-EDG). The test data also indicated that the cylinder temperature spread across each engine was reduced, and that the maximum firing pressure did not exceed 1800 psi on either engine at full load. Following the outage maintenance and testing, the inspectors observed the diagnostic tests on both engines. The diagnostic computer data was not available at the end of the current inspection; however, the inspectors noted that engine performance appeared to improve based upon reduced engine noise and vibration levels.

The licensee's actions to reduce EDG delta peak firing pressure and peak firing pressure to below the manufacturers stated maximums, and to resolve this item have been effective. This item is closed.

E8.2 <u>(Closed) LER 1999-001: "Deficiencies in NSSS Vendor Steamline Break Mass and Energy Release Analysis Results in Plant Being Outside its Design Basis"</u>

The licensee submitted LER 1999-001 following a notification from Westinghouse of two modeling errors in the plant's 1995 analysis for a main steamline break (MSLB) inside containment with an assumed single failure of a main feedwater regulating valve (FRV) (see IR 50-244/99-02). One error involved approximately 1000 gallons of high temperature feedwater that was not previously accounted for. The other error involved an under-estimate of the total time required to isolate the feedwater system. Both modeling errors resulted in a non-conservative impact on the actual margin to the design basis maximum containment pressure (60 psig), and preliminary analyses indicated that it could have been exceeded following the more limiting MSLB scenario. At the time of the notification, the plant was in a gradual power reduction for a planned refueling outage that commenced in March 1999. The licensee reported in the LER that there were no operational or safety consequences to the Ginna plant at the time of the event since the preliminary analysis showed that the maximum containment pressure would not be exceeded if a MSLB occurred while the power reduction continued, and if service water inlet temperature remained less than 40°F. Less than a week later, the Ginna plant was shutdown in MODE 5, and the temporary operating restrictions no longer applied.

The LER stated that further analysis was necessary to determine if the plant was actually operated outside its design basis during previous operating cycles at high power, and to

assure that the maximum containment pressure would not be exceeded if a limiting MSLB occurred during the next operating cycle. The LER remained open pending NRC review of the specific root cause of the 1995 modeling errors, the cycle 28 reload analysis, the LER supplement, and an evaluation of deficiencies in RG&E's design verification processes that did not detect the errors in the 1995 analysis.

The licensee approved safety evaluation SEV-1131, "Cycle 28 Reload," Revision 1 on April 15, 1999. The evaluation incorporated the results of a calculation and re-analysis performed by Westinghouse using corrected modeling assumptions to determine the peak containment pressure following a design basis MSLB from 102%, 70%, and 30% power. The safety evaluation concluded that the Ginna Station could be operated at full power with maximum service water temperature for the full cycle, and that containment pressure would not exceed 60.0 psig following a MSLB, providing the following operating restrictions were maintained: 1) a nominal 100% power Tave must be maintained at 561 $\pm$1.5°F throughout Cycle 28, 2) a cycle-specific shutdown margin of 2.40% delta K/K must be maintained with main feedwater supplying the steam generators, and 3) containment pressure must be maintained $\leq$1.0 psig, as required by the ITS. The Westinghouse calculation found that the most limiting MSLB with one FRV failure occurred at 102% power and resulted in a peak containment pressure of 59.84 psig. The calculation utilized a more realistic time for the rate of energy release into containment following the MSLB, which resulted in a slower increase in containment pressure and earlier pressure suppression from containment cooling. The original analysis had assumed that all of the available feedwater inventory instantly flashed into containment. The re-analysis utilized other refinements to obtain a lower initial reactor coolant system temperature and initial steam generator pressure, which achieved an acceptable peak pressure for the most limiting case.

The inspectors concluded that the licensee's safety evaluation adequately described the technical and safety bases used to justify entry into operating MODE 3 and full power operation of the plant throughout Cycle 28. The licensee indicated that further analyses by RG&E and Westinghouse were ongoing, and were expected to result in a larger margin below the design containment pressure using additional refinements for plant-specific modeling assumptions. In addition, the licensee's review of the causes for the original modeling errors was still ongoing at the end of the current inspection period. These reviews will be included in a supplement to the LER. This LER is closed.

E8.3 Year 2000 Project Readiness Review

During this inspection period, a review was conducted of Ginna's Y2K activities and documentation using Temporary Instruction (TI) 2515/141, "Review of Year 2000 (Y2K) Readiness of Computer Systems at Nuclear Power Plants." The review addressed aspects of Y2K management planning, documentation, implementation planning, initial assessment, detailed assessment, remediation activities, Y2K testing and validation, notification activities, and contingency planning. The reviewers used NEI/NUSMG 97-07, "Nuclear Utility Year 2000 Readiness," and NEI/NUSMG 98-07, "Nuclear Utility Year 2000 Readiness Contingency Planning," as the primary references for this review.

The results of this review will be combined with similar reviews of Y2K programs at other U.S. commercial nuclear power plants and summarized in a report to be issued by the NRC staff by July 31, 1999.

IV. Plant Support

R1 Radiological Protection and Chemistry (RP&C) Controls

R1.1 1999 Refueling Outage Radiological Protection Controls

a. Inspection Scope

The inspectors observed the licensee's radiological protection (RP) controls inside the containment during the refueling outage.

b. Observations and Findings

The inspectors observed selected radiological boundaries the licensee established inside the containment for controlling radiological aspects of work performed, or for materials stored inside the designated boundaries within the radiologically controlled area (RCA). All boundary markers observed were clearly visible, well marked, and accurately posted and the RP technicians assigned to containment work effectively maintained access controls across boundaries. During the current refueling outage, the licensee performed inspections and replacements of reactor vessel lower internal baffle bolts, reactor vessel ten-year in-service inspections, and reactor vessel head penetration inspections which required close support from RP technicians. The inspectors observed that outage workers involved in these activities exercised appropriate radiological work practices, and that contaminated equipment was properly identified and controlled. The licensee's pre-outage estimate for total personnel exposure were slightly exceeded for the outage. However, the original exposure estimate was relatively uncertain since the unique activities inside containment could not be precisely predicted from past experience. An unexpectedly high number of personnel contamination incidents occurred during the outage, but the licensee was able to identify and decontaminate individuals prior to them exiting from the RCA. No individual ingested radiological material from these events. The licensee closely monitored exposures and contaminations on a daily basis and implemented additional controls where necessary. Overall, the licensee's radiological controls inside containment were effective in maintaining personnel exposures and contaminations as low as reasonably achievable.

c. Conclusions

Overall, the licensee's radiological work and boundary controls inside containment during the recent outage were effective in maintaining personnel exposures and contaminations at reasonably low levels. The pre-outage exposure and contamination goals were slightly exceeded at the end of the outage; however, the licensee maintained an acceptable level of oversight and control in this area.

V. Management Meetings

X1 Exit Meeting Summary

After the inspection was concluded, the inspectors presented the results to members of licensee management on May 27, 1999. The licensee acknowledged the findings presented. The licensee indicated that no materials examined during the inspection were considered proprietary.

ATTACHMENT I

PARTIAL LIST OF PERSONS CONTACTED

Licensee

T. Alexander	Operational Review Manager
P. Bamford	Reactor Engineering Manager
G. Graus	I&C/Electrical Maintenance Manager
G. Hermes	Acting Primary Systems Engineering Manager
J. Hotchkiss	Mechanical Maintenance Manager
M. Lilley	Quality Assurance Manager
R. Marchionda	Nuclear Assessment Manager
T. Marlow	Nuclear Engineering Services Manager
G. Joss	Results and Test Supervisor
R. Popp	Production Superintendent
J. Pascher	Electrical Systems Engineering Manager
T. Plantz	Maintenance Systems Manager
R. Ploof	Secondary Systems Engineering Manager
P. Polfleit	Emergency Preparedness Manager
J. Smith	Maintenance Superintendent
W. Thomson	Radiological Protection & Chemistry Manager
J. Wayland	Scheduling Manager
J. Widay	Plant Manager
T. White	Operations Manager
G. Wrobel	Nuclear Safety & Licensing Manager

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 40500:	Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 61726:	Surveillance Observation
IP 62707:	Maintenance Observation
IP 64704:	Fire Protection Program
IP 71707:	Plant Operations
IP 71750:	Plant Support
IP 92700:	Onsite Follow-up of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92901:	Follow-up - Operations
IP 92902:	Follow-up - Maintenance
IP 92903:	Follow-up - Engineering

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

IFI	99-03-01	Reactor Protection Module Modifications for Primary System Temperature, and Primary RTD Replacements

Closed

IFI	97-12-02	Determine Appropriate Operability Parameters for the EDGs
LER	1999-001	Deficiencies in NSSS Vendor Steamline Break Mass and Energy Release Analysis Results in Plant Being Outside its Design Basis
LER	1999-003	Two Valves Declared Inoperable Results in Condition Prohibited by ITS
LER	1999-004	Containment Recirculation Fan Moisture Separator Vanes Incorrectly Installed Results in Unanalyzed Condition
LER	1999-005	UV Signal on Safeguards Bus During Testing Results in Auto Start of "B" EDG

Discussed

None

LIST OF ACRONYMS USED

ALARA	As Low As Reasonably Achievable
ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
CRFC	Containment Recirculation Fan Cooler
CV	Check Valve
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure
ESF	Engineered Safety Feature
FRV	Feedwater Regulating Valve
HEPA	High Energy Particulate Air
IFI	Inspector Follow-up Item
IR	Inspection Report
ISI	Inservice Inspection
ITS	Improved Technical Specification
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
MSLB	Main Steamline Break
NRC	Nuclear Regulatory Commission

OPΔT	Overpressure delta temperature
OTΔT	Overtemperature delta temperature
PORC	Plant Operations Review Committee
psig	pounds per square inch gage
PT	Periodic Test
RCA	Radiologically Controlled Area
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RG&E	Rochester Gas and Electric Corporation
RP	Radiation Protection
RP&C	Radiological Protection and Chemistry
RPS	Reactor Protection System
RSSP	Refueling Shutdown Surveillance Procedure
RTD	Resistance Temperature Detector
SEV	Safety Evaluation
SI	Safety Injection
SIPE	Significant Infrequently Performed Evolution
Tave	Primary System Average Temperature
TDAFW	Turbine Driven Auxiliary Feedwater Pump
UFSAR	Updated Final Safety Analysis Report