U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket No.	50-244
Report No.	93-20
Licensee:	Rochester Gas and Electric Corporation (RG&E) 89 East Avenue Rochester, New York
Facility:	R. E. Ginna Nuclear Power Plant
Location:	Ontario, New York
Inspection:	September 12 through October 23, 1993
Inspectors:	T. A. Moslak, Senior Resident Inspector, Ginna E. C. Knutson, Resident Inspector, Ginna
Approved by:	W. J. Lazarus, Chief, Reactor Projects Section 3B — 11/12/93 Date

Inspection Scope

Plant operations, maintenance, engineering, and plant support.

EXECUTIVE SUMMARY
R. E. Ginna Nuclear Power Plant
Report No. 50-244/93-20

Operations

The plant operated at full power throughout the inspection period. There were no significant operational events. The Plant Operations Review Committee was responsive in reviewing maintenance issues that affected plant operations, and demonstrated a strong safety perspective in resolving issues.

Maintenance

Maintenance to repair a service water leak inside containment generated questions with respect to technical specification requirements for containment integrity. In attempting to repair a service water leak in the "A" reactor compartment cooler, the isolation valves (V-4757 and V-4758, manual valves located outside of containment) were found to have excessive seat leakage. Due to the plant's older design, the supply and return lines to this cooler each have only one isolation valve. Maintenance isolation was established using other service water supply and return valves, and repairs were performed to the cooler and the leaking containment isolation valves. During this maintenance, containment integrity was potentially compromised, in that the cooler repairs were not leak tested prior to disassembly of the isolation valves. The licensee considered that the technical specifications which govern repair of inoperable containment isolation boundaries authorize containment to be violated for up to four hours. As a result, requirements for containment isolation boundaries are currently under review by the NRC. This item will be tracked as an unresolved item.

Engineering

Licensee investigation in response to NRC Information Notice 92-06, Supplement 1, revealed a design deficiency in the ATWS Mitigation System Actuation Circuitry (AMSAC); specifically, a power level time delay lock-in feature had not been included in the system as required by the original design specifications. The licensee evaluated the implications of this deficiency on continued safe operations. This evaluation, along with interim compensatory measures, was presented to the NRC.

Executive Summary

Plant Support

Inadequately detailed radiological sampling procedures affected plant operations on two occasions. A health physics technician inadvertently isolated a recirculation fan cooler condensate level transmitter while attempting to obtain a condensate sample in accordance with a radioactive discharge (RD) procedure. This action rendered the containment water level inventory system inoperable for approximately one day. In the second instance, during maintenance on the containment atmosphere radiation monitors, the licensee was required by technical specifications to obtain daily containment atmosphere grab samples. To ensure this requirement was met, the licensee's policy was to sample every 12 hours; however, lack of detail in the RD procedure caused the first sample to be invalid. By the time this problem was recognized, the licensee had exceeded the 24 hour sampling periodicity. The inspector assessed these problems to be indicative of a weakness with radiological sampling procedures. This item was left unresolved pending further NRC review.

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DETAILS

1.0 OPERATIONS (71707)

1.1 Operational Experiences

The plant operated at full power (approximately 98 percent) throughout the inspection period. There were no significant operational events during the inspection period.

1.2 Control of Operations

Overall, the inspectors found the R. E. Ginna Nuclear Power plant to be operated safely. Control room staffing was as required. Operators exercised control over access to the control room. Shift supervisors maintained authority over activities and provided detailed turnover briefings to relief crews. Operators adhered to approved procedures and were knowledgeable of off-normal plant conditions. The inspectors reviewed control room log books for activities and trends, observed recorder traces for abnormalities, assessed compliance with technical specifications, and verified equipment availability was consistent with the requirements for existing plant conditions. During normal work hours and on backshifts, accessible areas of the plant were toured. No operational inadequacies or concerns were identified.

2.0 MAINTENANCE (62703, 61726)

2.1 Corrective Maintenance

2.1.1 Routine Observations

The inspector observed portions of maintenance activities to verify that correct parts and tools were utilized, 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. The following maintenance activities were observed:

- Work Order 9301360, "Replace Gasket In Inboard Mechanical Seal Flange - "C" Safety Injection Pump," performed in accordance with Maintenance Procedure (M)-37.130, "Disassembly and Reassembly of Pipe Flange Connections," revision 11, dated August 7, 1992, observed October 19, 1993

-- While establishing isolation for the maintenance, the licensee found a normally locked-closed safety injection (SI) system valve closed but with its lock disengaged. The licensee responded by generating an event report per administrative procedure (A)-25.1, "Ginna Station Event Report." Corrective action included verification lineups on safety systems outside of containment using the system operating procedure (S)-30 series, "Safeguard Systems Valves and Breakers," and verification of locked valves and breakers per administrative procedure (A)-52.2, "Control of Locked Valve and Breaker Operation." The lock in question was determined to be faulty and was replaced.

-- As a part of the maintenance, the installed flange fasteners were to be replaced. The replacement fasteners, fabricated of a different material, were to be torqued to a higher value than had been used on the original fasteners. As the fasteners were being incrementally torqued, the mechanics noted that the flange faces were not parallel. Since one was a slip-on flange (not welded to the pipe), they concluded that this condition might resolve itself as greater torque was applied. However, as torque was increased above the value used with the original fasteners, the slip-on flange began to bow. The flange relaxed when the fasteners were loosened, indicating that only elastic deformation had occurred. Engineering resolution of this condition was to reassemble the flange using the original fasteners, with approval for interim use documented in a non-conformance report (NCR 93-241).

- Work Order 9341419, "Perform CP-I-RMS-CNMT PART-R11," modify wiring for Victoreen Model 942A Universal Digital Ratemeters and P1 cable/connector per Engineering Change Notice 4068B-35, accomplished per PCN 93T-782 to CPI-MON-R11, "Calibration of Radiation Monitoring System Channel R-11 Containment Particulate," revision 5, dated October 7, 1993, observed October 21, 1993

-- The inspector noted good communications between the instrument and control (I&C) technicians and the control room operators during this maintenance.

2.1.2 Snubbers for "B" Steam Generator Found Inoperable

Hydraulic snubbers are a type of mechanical support that are used on components that are subject to thermal expansion effects. Snubbers are designed to change length in response to application of a slow force, such as is generated by expansion or contraction along a length of piping during heating or cooling. In response to a rapid acceleration, such as would be caused by a seismic event, hydraulic snubbers lock in length and provide rigid support for the associated component.

On September 29, 1993, during the conduct of maintenance procedure (M)-40, "Surveillance and Maintenance of Hydraulic Snubbers," the combined oil reservoir for "B" steam generator hydraulic snubbers SGB-3 and SGB-4 was found empty. These two snubbers were declared inoperable, placing the licensee in a 72-hour action statement per technical specification 3.13.2.

Investigation revealed that the vent plug on snubber SGB-4 was loose; oil in the area further supported that this had been the source of the leakage. The plug was tightened and the reservoir was refilled. Regular observation through use of the containment video monitor has shown no subsequent loss of oil.

Along with restoring the affected hydraulic snubbers to operable status, TS 3.13.2 requires that an engineering evaluation be performed on the supported component. The purpose of this evaluation is to determine if the component was adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

On October 1, 1993, the Plant Operations Review Committee (PORC) met to review Design Analysis DA-ME-93-127, "Evaluation of Reactor Coolant System (Due to Loss of Fluid of Snubbers SGB-3 and SGB-4)." The evaluation concluded that the reactor coolant system had not been adversely affected by inoperability of the two snubbers, in that no seismic events or rapid plant transients had occurred over the three month period since the snubbers had previously been satisfactorily inspected (and therefore known to have been operable). The evaluation also noted that loss of reserve oil would not affect snubber operation during slow transients, such as controlled power reductions. The PORC concurred with this evaluation, and SGB-3 and SGB-4 were declared operable on October 1, 1993.

The inspector attended the PORC meeting at which the engineering evaluation of the reactor coolant system / SGB-3 and SGB-4 inoperability was reviewed. Requirements for restoration of the snubbers to operable status were discussed, and the plant operating history during the period of snubber inoperability was reviewed. The inspector noted that concerns were freely discussed and pursued to resolution. The inspector concluded that the evaluation adequately addressed operability considerations, and that restoration of SGB-3 and SGB-4 to operable status was appropriate. The inspector had no additional concerns on this matter.

2.1.3 Service Water Leak from "A" Reactor Compartment Cooler

During a routine at-power containment entry on September 29, 1993, the licensee identified a service water (SW) leak from the "A" reactor compartment (RC) cooler heat exchanger. The leakage rate was estimated to be 0.25 gallons per minute (gpm). The leak was isolated by shutting the service water supply and return block valves for the "A" RC cooler, V-4757 and -4758, respectively. These local manual valves are 2½-inch butterfly valves and are the first valves outside of containment penetrations P201 and P209, respectively. Although classified as containment isolation boundaries by technical specification (TS) 3.6, they are not subject to 10 CFR Appendix J requirements. The two ("A" and "B") RC coolers do not perform safety-related functions, but are supplied by the safety-related portion of the service water system.

A containment entry was conducted on September 30, 1993, to repair the "A" RC cooler SW leak. The source of the leak was a crack in one of the U-bends in the ½-inch piping that makes up the heat exchanger. The repair was to be made by cutting off the U-bend and plugging the two remaining ends. Although there was no leakage from the crack at the commencement of work, leakage resumed as cutting progressed. The heat exchanger drain was opened in an attempt to reduce leakage at the work site to the point that cutting could continue. The cooler drained at a rate of 10-15 gpm, as indicated by the "A" containment sump level. When the rate of drainage did not slow over time, the licensee concluded that one or both of the isolation valves, V-4757 and -4758, were leaking. At 1:20 p.m., the licensee declared these valves inoperable and entered TS 3.6.3.1. This limiting condition for operation (LCO) provides three action statements in the case that a containment isolation boundary is inoperable; specifically:

a. Restore each inoperable boundary to operable status within 4 hours, or;

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, one closed manual valve, or a blind flange, or;

c. Be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

To isolate V-4757 and -4758, the SW block valves for both the "A" and "B" RC coolers, V-4625/-4626 (supply), and V-4624 (return) were shut. This action stopped the drainage from the "A" RC cooler and confirmed that V-4757 and/or -4758 were leaking. The licensee considered that action statement 3.6.3.1.b had been satisfied, and exited the LCO at 2:32 p.m., September 30, 1993. Temperature monitoring of components affected by the loss of both RC coolers (primarily the excore nuclear instrument detectors) demonstrated that this condition would be acceptable for short term operation.

Repair of the "A" RC cooler SW heat exchanger was completed on September 30, 1993. The following day, the licensee planned to disassemble V-4757 and -4758 for inspection and repair. The licensee considered that the one hour and 12 minutes spent in TS 3.6.3.1 on September 30 counted against the four hours allowed by action statement 3.6.3.1.a to restore V-4757/-4758 to operable status. Therefore, the licensee concluded that they still had two hours and 48 minutes in which valve repairs could be performed under this action statement. The licensee entered action statement 3.6.3.1.a at 8:40 a.m., October 1, 1993, when maintenance on V-4757/-4758 commenced. The licensee remained in this action statement until leak tightness of the affected mechanical joints was demonstrated by an operational pressure test. Upon satisfactory completion of this test at 10:29 a.m., October 1, 1993, the licensee exited action statement 3.6.3.1.a. The time spent in this action statement on October 1 was one hour and 49 minutes; the total time spent in this action statement since V-4757/-4758 had been declared inoperable was three hours and one minute.

The licensee entered action statement 3.6.3.1.b when valves V-4624/5/6 were shut upon completion of the operational pressure test. The licensee remained in this action statement for the remainder of V-4757/-4758 acceptance testing. This testing consisted of a combined seat leakage test, and was performed in accordance with Refueling Shutdown Surveillance Procedure (RSSP)-2.8, "C.V. Reactor Compartment Cooling "Unit A" and "B" Service Water Valves Leak Check," revision 4, PCN 93T-763, dated March 13, 1992.

The licensee conducted repairs to V-4757/-4758 before repairs to the "A" RC cooler had been tested. The licensee indicated that the closed system does not constitute a containment boundary, and that testing the cooler repairs before work on the valves gained them nothing as far as compliance with TS. The licensee considers that action statement 3.6.3.1.a authorizes four hours to repair an inoperable containment isolation boundary. The licensee believes that because of their older design, there are places where only a single containment boundary exists, and that TS 3.6.3 authorizes 4 hours to restore the boundary. Regardless of the interpretation of TS 3.6.3, it would have been prudent to test the cooler repairs prior to starting work on the isolation valves.

It is not clear that the intent of TS 3.6.3 was to authorize continued plant operation with containment integrity violated. As a result, requirements for containment isolation boundaries are currently under review by the NRC. Therefore, this item is unresolved (50-244/93-20-01).

2.2 Surveillance Observations

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:

- Performance Test (PT)-2.1M, "Safety Injection System Monthly Test," revision 9, dated January 22, 1993, observed September 15, 1993

- PT-12.1, "Emergency Diesel Generator 1A," revision 71, procedure change notice (PCN) 93T-742, dated August 24, 1993, observed September 24, 1993

- PT-2.2Q, "Residual Heat Removal System - Quarterly," revision 4, dated February 18, 1993, observed October 7, 1993

- PT-16Q-B, "Auxiliary Feedwater Pump B - Quarterly," revision 10, dated July 22, 1993, observed October 12, 1993

-- The inspector noted that motor operated valve MOV-4008 (MDAFW pump B discharge valve) would not remain in manual unless the declutch lever was continuously depressed. Although this does not affect the operability of the valve, it indicates misadjustment or improper assembly of the motor operator declutch assembly. A trouble card (9301531) was generated to document the condition and initiate corrective action.

- PT-36Q-C, "Standby Auxiliary Feedwater Pump C - Quarterly," revision 8, dated October 14, 1993, observed October 21, 1993

The inspector determined through observing this testing 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.

2.3 Quality Assurance Audit of Maintenance Activities

On September 29, 1993, the inspector attended the Quality Performance Department meeting addressing the results of a Maintenance Department audit. The inspector concluded that the audit had appropriate scope and depth to assess the overall quality by which maintenance activities are managed and performed, including procedure adherence, prioritization of shop

activities, management of work order backlog, verification of technician qualifications, and resolution of past audit findings. Action items were assigned for the deficiencies identified to assure timely resolution.

The inspector concluded that the licensee's Quality Performance organization continues to improve its effectiveness as a management tool by conducting a comprehensive performance based audit of maintenance activities.

3.0 ENGINEERING (71707, 92701)

3.1 AMSAC Design Deficiency

In response to an NRC concern addressed in NRC Information Notice 92-06, Supplement 1, "Reliability of ATWS Mitigation Systems and other NRC-Required Equipment Not Controlled By Plant Technical Specifications," dated July 1, 1993, the licensee provided an evaluation for NRC staff review of a design deficiency identified in the ATWS Mitigation System Actuation Circuitry (AMSAC). The design deficiency is the absence of a power level time delay lock-in feature, that was not incorporated into the original AMSAC equipment by the manufacturer as specified by Westinghouse. The requirement for this feature was found to be missing by the RG&E engineering staff subsequent to discussions with Westinghouse and affected utilities, and through review of the supporting documentation of the AMSAC generic design package (WCAP 10858-P-A, Revision 1) that transmitted the final AMSAC design package to the NRC.

In the RG&E correspondence to the NRC dated October 22, 1993, the licensee assessed the safety implications of the deficiency, identified the in-place compensatory measures, and provided a schedule for incorporating this function into the system design. This modification would require bench testing of a new variable timer lock-in module (AMSAC SPEC 200 MICRO module PY-400B), validation of the proposed software changes, and testing of all functions performed by this module. The licensee projects that the modification could be expeditiously completed if the engineering staff determines that plant stability would be unaffected during AMSAC modification and testing.

3.2 Licensee Action on Previous Inspection Findings

3.2.1 (Closed) Unresolved Item (50-244/91-201-12) Licensee is to Establish an Engineering Basis for Low Pressure Setpoints in the Service Water System Header

In response to this concern, the licensee completed a formalized engineering analysis (DA-92-ME-0116) which documented the alarm setpoints and operator actions for two and three pump operation. This analysis justified that the 40 psig value is sufficient to ensure that adequate service water flow would be provided to the required loads under accident or transient conditions. Plant procedure changes have been completed to reflect the following settings:

2 pump operation

55 psig	Administrative limit
50 psig	Computer alarm setting: verify one pump in each loop operating; continue leak inspection; verify flow to CCW heat exchangers; check operation of temperature control valves; initiate SW loop separation if necessary.
45 psig	Start 3rd SW pump; initiate controlled shutdown if pressure not stabilized
40 psig	Trip the reactor; proceed to E-0

3 pump operation

60 psig	Administrative limit
55 psig	Computer alarm setting; verify 3 pumps operating; continue leak inspection; initiate SW loop separation if necessary; initiate controlled shutdown if pressure not stabilized
40 psig	Trip the reactor; proceed to E-0

The inspector had no further questions on this matter.

3.3 Erosion/Corrosion (E/C) Integrated Management Team Meeting

On September 29, 1993, the inspector attended the quarterly meeting of the RG&E E/C Integrated Management Team. Attendees included management representatives from corporate engineering, site maintenance, site technical engineering, operations, chemistry, and materials inspection departments. The meeting addressed prioritization of E/C inspections, scope of inspections and replacements scheduled for the 1994 outage, and the status of small bore piping replacement. The extent of replacing portions of the extraction steam lines to the 4A and 4B feedwater heaters and the scope of radiographic inspections of the house heating steam system during the 1994 outage were also discussed.

The inspector concluded that the licensee is actively carrying out an E/C program to identify and correct potential pipe thinning problems to improve plant safety and reliability.

4.0 PLANT SUPPORT (71707)

4.1 Radiological Controls

4.1.1 Routine Observations

The inspectors periodically confirmed that radiation work permits were effectively implemented, dosimetry was correctly worn in controlled areas and dosimeter readings were accurately recorded, access to high radiation areas was adequately controlled, survey information was kept current, and postings and labeling were in compliance with regulatory requirements. Through observations of ongoing activities and discussions with plant personnel, the inspectors concluded that the licensee's radiological controls were generally effective.

4.1.2 Inadequate Radiological Sampling Procedures

Containment Water Inventory System Inoperable

On September 29, 1993, a containment entry was conducted to perform routine surveillance and inspection. Later that day, control room operators observed that the condensate collecting pan level indicator for the "D" reactor compartment recirculation fan cooler, LI-1093, did not move from the high level alarm setpoint (approximately 60 percent) when the dump valve was opened. Because both of the "A" containment sump level instruments were also inoperable, the licensee declared the containment water inventory monitoring system inoperable. Other reactor coolant system leakage detection systems remained operable as required per TS 3.1.5.1.1, and therefore, no action statement was incurred.

On September 30, 1993, the licensee again entered containment to investigate the malfunctioning "D" reactor compartment recirculation fan cooler condensate collection pan level indicator. When the level transmitter valve alignment was checked, normally-opened valve 11492 (high side shutoff for transmitter LT-1093) was found to be shut. The valve was returned to its required position, which restored the level transmitter to service. The containment water inventory monitoring system was subsequently declared operable.

Licensee discussions with personnel who had participated in the September 29 containment entry revealed that valve 11492 had been erroneously shut by a health physics technician while obtaining a fan condensate sample in accordance with radioactive discharge procedure (RD)-1.3, "Fan Condensate Method for Determining Containment Tritium Air." Although this procedure identifies the "D" containment recirculation fan cooler level indicator as the sample collection point, the valve to be operated is not specifically identified.

Containment Radioactive Particulate Sample Not Obtained During Installed Radiation Monitor Calibration

At 9:07 a.m. on October 21, 1993, the containment particulate radiation monitor (radiation monitoring system channel R-11) was declared inoperable for maintenance and calibration. Due to system configuration, this maintenance also placed the containment noble gas radiation monitor (R-12) out of service. These two channels of the radiation monitoring system fulfill the requirement of technical specification 3.1.5.1.1 for reactor coolant system leakage detectors that are sensitive to radioactivity. If both channels are inoperable, technical specification 3.1.5.1.2 requires that grab samples of the containment atmosphere be obtained and analyzed at least once every 24 hours. To ensure this requirement is met, the licensee obtains containment grab samples once every 12 hours whenever both R-11 and R-12 are inoperable.

On the morning of October 22, 1993, while attempting to obtain a containment atmosphere grab sample per RD-1.2, "Alternate Sample Point For Containment Atmosphere Sampling And Analysis When RM-10A, RM-11 and RM-12 Are Out Of Service," the health physics technician observed that the sample air flow indication had dropped to zero. The technician subsequently noted that the containment air sample isolation valve, AOV-1597, was shut. This normally-open valve had automatically closed in response to an isolation signal that had occurred the previous day as an expected condition associated with the R-11 maintenance. RD-1.3 does not address the required position of AOV-1597, nor does it identify that a containment ventilation isolation signal will preclude sampling.

At about 10:00 a.m., the technician called the control room and requested that AOV-1597 be opened. The shift supervisor had been on duty when the first grab sample was obtained, and realized that there had been no request to have AOV-1597 opened at that time. Since the containment ventilation isolation signal had been present, the first grab sample had been obtained with the containment air sample isolation valve shut and was therefore invalid. The shift supervisor realized that greater than 24 hours had elapsed since R-11 had been declared inoperable; beyond 24 hours, technical specification 3.1.5.1.2 requires that the reactor be placed in hot shutdown within the next six hours.

Operators reset the containment ventilation isolation signal and associated ESF relays, and opened AOV-1597. By 10:52 a.m., a grab sample had been obtained and analyzed, showing normal particulate activity. Power reduction had not been commenced when the problem was identified, because the shift supervisor knew that the sample would be completed well ahead of the six hour technical specification operating limit.

Discussion

The licensee determined that both of these incidents were the result of inadequately detailed radiological sampling procedures. In response, the licensee initiated human performance evaluation system (HPES) reviews of both events. At the close of the inspection period, these reviews were still in progress. The inspector considered that these two instances indicated a

weakness in radiological sampling procedures. At the end of this inspection, the NRC had not completed the review of the licensee's long term corrective action or determined if there had been previous occurrences. This item will be left unresolved pending the completion of the NRC's review (50-244/93-20-02).

4.2 Security

4.2.1 Routine Observations

During this inspection period, the inspectors verified that x-ray machines and metal and explosive detectors were operable, protected area and vital area barriers were well maintained, personnel were properly badged for unescorted or escorted access, and compensatory measures were implemented when necessary. No unacceptable conditions were identified.

4.3 Fire Protection

4.3.1 Routine Observations

The inspectors periodically verified the adequacy of combustible material controls and storage in safety-related areas of the plant, monitored transient fire loads, verified the operability of fire detection and suppression systems, assessed the condition of fire barriers, and verified the adequacy of required compensatory measures. No discrepancies were identified.

4.4 Emergency Preparedness

4.4.1 Practice Drill

On October 13, 1993, the licensee conducted a practice emergency preparedness drill in preparation for the upcoming NRC/FEMA evaluated drill. The inspector verified that conduct of this practice drill, which included a site evacuation, did not adversely affect the conduct of plant operations.

4.5 Periodic Reports

Periodic reports submitted by the licensee pursuant to Technical Specification 6.9.1 were reviewed. 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 report was reviewed:

-- Monthly Operating Report for September, 1993

No unacceptable conditions were identified.

4.6 Licensee Event Reports

A licensee event report (LER) submitted to the NRC was reviewed to determine whether details were clearly reported, causes were properly identified, and corrective actions were appropriate. The inspectors also assessed whether potential safety consequences were properly evaluated, generic implications were indicated, events warranted onsite follow-up, and applicable requirements of 10 CFR 50.72 were met.

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

- 93-004, Feedwater Control Perturbations, Due To A Secondary Side Transient, Causes Steam Generator High Level Feedwater Isolations (July 7, 1993)

The inspector concluded that the LER was accurate and met regulatory requirements. However, the inspector identified a large percentage of the cause codes classified as "other" in the recent LERs, and informed the licensee that the methodology used to determine the appropriate code should be reevaluated.

5.0 ADMINISTRATIVE (71707, 30702, 94600)

5.1 Backshift and Deep Backshift Inspection

During this inspection period, a backshift inspection was conducted on September 30, 1993. Deep backshift inspections were conducted on the following dates: September 12, 18, October 2 and 11, 1993.

5.2 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 inspection report 50-244/93-17 (initial operator licensing examination, conducted September 13-17, 1993) was held by Mr. W. Maier on September 17, 1993. The exit meeting for inspection report 50-244/93-19 (operator requalification program inspection, conducted September 19-24, 1993) was held by Mr. J. D'Antonnio on September 24, 1993. The exit meeting for inspection report 50-244/93-21 (engineering programs review, conducted October 4-8, 1993) was held by Mr. L. Scholl on October 8, 1993. The exit meeting for inspection report 50-244/93-20 was held on October 27, 1993.

5.3 NRC Staff Activities

J. Linville, Chief, Region I Projects Branch 3, visited the site on September 13 and 14, for discussions with the inspectors and utility management, and to tour the site.