January 22, 2004

Dr. Robert C. Mecredy
Vice President, Nuclear Operations
Rochester Gas and Electric Corporation
89 East Avenue
Rochester, New York 14649

SUBJECT: R. E. GINNA NUCLEAR POWER PLANT - NRC INTEGRATED INSPECTION REPORT 05000244/2003007

Dear Dr. Mecredy:

On December 31, 2003, the US Nuclear Regulatory Commission (NRC) completed an inspection at your R. E. Ginna facility. The enclosed integrated inspection report documents the inspection findings, which were discussed on January 8, 2004 with you and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

The report documents four findings of very low safety significance (Green). Three of the four findings were determined to involve a violation of NRC requirements. However, because of their very low safety significance, and because they have been entered into your corrective action program, the NRC is treating these issues as non-cited violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny the non-cited violations noted in this report, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Ginna facility.

Since the terrorist attacks on September 11, 2001, the NRC has issued five Orders and several threat advisories to licensees of commercial power reactors to strengthen licensee capabilities, improve security force readiness, and enhance controls over access authorization. In addition to applicable baseline inspections, the NRC issued Temporary Instruction 2515/148, "Inspection of Nuclear Reactor Safeguards Interim Compensatory Measures, " and its subsequent revision, to audit and inspect licensee implementation of the interim compensatory measures required by the order. Phase 1 of TI 2515/148 was completed at all commercial nuclear power plants during calendar year 2002, and the remaining inspection activities for Ginna were completed in August 2003. The NRC will continue to monitor overall safeguards and security controls at Ginna.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of the NRC's document management system (ADAMS). ADAMS is accessible from the NRC web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

James M. Trapp, Chief
Projects Branch 1
Division of Reactor Projects

Docket No. 50-244
License No. DPR-18

Enclosure: Inspection Report 05000244/2003007
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	50-244
License No:	DPR-18
Report No:	05000244/2003007
Licensee:	Rochester Gas and Electric Corporation (RG&E)
Facility:	R. E. Ginna Nuclear Power Plant
Location:	1503 Lake Road Ontario, New York 14519
Dates:	September 28, 2003 - December 31, 2003
Inspectors:	K. Kolaczyk, Senior Resident Inspector M. Marshfield, Resident Inspector S. Barr, Operations Engineer L. Cheung, Senior Reactor Inspector T. Fish, Senior Operations Engineer T. Moslak, Health Physicist K. Naidu, Senior Reactor Engineer J. Richmond, Resident Inspector, Susquehanna J. Schoppy, Senior Reactor Inspector
Approved by:	James M. Trapp, Chief Projects Branch 1 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000244/2003007; 9/28/2003 - 12/31/2003; R. E. Ginna Nuclear Power Plant; Adverse Weather, Post Maintenance Testing, Risk Assessment and Emergent Work, Event Follow-up.

The report covered a 3-month period of inspection by resident inspectors and announced inspections by regional specialists. This inspection identified three Green non-cited violations (NCVs) and one Green finding. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. <u>NRC-Identified and Self-Revealing Findings</u>

Cornerstone: Initiating Events

<u>Green</u>. The inspectors identified that although the Ginna site was experiencing high winds, control room operators did not implement the site adverse weather plan contained in procedures EPIP 1-17, "Planning for Adverse Weather," and ER-SC-1, "Adverse Weather Plan," until prompted by the inspector. Following implementation of ER-SC-1, operators manually tripped the reactor as required by Abnormal Operating Procedure AP-RCS.2, "Loss of Reactor Coolant Pump Flow," when an offsite power supply was lost because of storm-related damage. The failure to implement EPIP 1-17 and ER-SC-1 is a violation of Technical Specification 5.4.1.

This finding is associated with the "Protection Against External Factors" attribute of the initiating events cornerstone. This finding is greater than minor because it affected the objective of limiting the likelihood of those events that upset plant stability during power operation in that the severe weather caused a reactor trip. Since operators responded appropriately to the loss of the 751 line, the finding is of very low safety significance because it did not contribute to the likelihood of a primary or secondary system LOCA initiator, or to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available. Additionally, the finding did not increase the likelihood of a fire or internal/external flood. (Section 1R01)

Cornerstone: Mitigating Systems

- <u>Green</u>. The inspectors identified a non-cited violation of 10 CFR 50.65 (a)(4) when RG&E personnel installed an incorrect version of the risk analysis program on the plant intranet server. The program incorrectly modeled the impact of removing emergency diesel generators from service. RG&E personnel installed the correct version of the software when the error was identified by the inspectors.

After management review, this finding was determined to be greater than minor, because the plant risk analysis assessments, which RG&E schedulers and operations personnel had performed on several occasions, were incorrect, and in one case on November 18, 2003, unbeknownst to Ginna personnel, the plant was in an elevated risk condition. If left uncorrected, this finding could become a more significant safety concern since with the incorrect software installed, operators could not correctly assess the impact on plant risk of maintenance on mitigating systems. The safety significance of the finding was very low, because the plant was not in a high risk condition at any time during the period that the wrong program was installed. (Section 1R13)

- Green. The inspectors identified that RG&E had performed maintenance on four main steam safety valves prior to performing required surveillance testing. This practice may mask the as-found condition of the valves, and affect the results of the surveillance tests.

  This finding is greater than minor, because it is associated with the "Equipment Performance" (reliability) attribute of the mitigating systems cornerstone, and it would adversely affect the cornerstone objective because failure to conduct as-found testing may mask any valve degradation. This could adversely impact the reliability of the steam generator overpressure protection system to prevent undesirable consequences. The finding is of very low safety significance because it was not a design or qualification deficiency, it did not represent a loss of safety function, and was not potentially risk significant due to seismic, flood, fire, or weather related initiating event. Further, the finding is of very low safety significance since the issue involved inadequate testing, and did not degrade the ability of the main steam safety valves to perform their intended function for the next operating cycle. (Section 1R19)

- Green. A self-revealing non-cited violation of Ginna Station Technical Specification (TS) Limiting Condition for Operation (LCO) 3.0.4 was identified when plant operators conducted PT-16Q-T after transition to Mode 2 from Mode 4 and found that flow could not be achieved from the turbine driven auxiliary feedwater (TDAFW) pump to the "B" steam generator. The line had been isolated through a sequence of lineups and testing, which was conducted prior to the Mode change that did not properly restore the system to the required line-up for the Mode change. The flow path was immediately restored by RG&E personnel when the deficiency was discovered. Procedures will be revised to minimize the possibility of event recurrence.

  This finding, associated with the "Configuration Control" attribute of the mitigating systems cornerstone, is greater than minor because it affected the objective of ensuring the reliability and capability of systems to prevent undesirable consequences in that the TDAFW system was inoperable for three days. The finding is of very low safety significance because it was not a design or qualification deficiency, it did not represent a loss of safety function (the remaining diverse trains of AFW remained operable), and was not potentially risk significant due to seismic, flood, fire, or weather related initiating events. Further, the exposure time was less than the LCO action time of seven days. (Section 4OA3)

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status

At the start of the inspection report period, Ginna was in a refueling outage in Mode 6. On October 3, 2003, refueling was completed, the plant entered Mode 5, and began a gradual heatup. On October 11, the plant entered Mode 4, and transitioned to Mode 3 on October 12, 2003. On October 14, the plant entered Mode 2 and remained in this condition until October 15, when operators manually tripped the reactor when a reactor coolant pump tripped due to a loss of an offsite power supply line caused by severe weather. Following a post-trip review, the reactor was restarted later that day and the plant entered Mode 1 on October 16. The reactor reached full power on October 20, 2003, and remained at that level until December 12, when the plant experienced an automatic turbine runback to 94% power as a result of an electrical transient caused by maintenance activities. After several hours, reactor power was returned to 100%, and it remained at that power level for the remainder of the report period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

1. High Wind Events

a. Inspection Scope

(71111.01 - 2 samples) On October 15 and November 13, 2003, high wind conditions occurred at the Ginna site. The inspectors were in the control room for several hours when the high wind conditions occurred, and reviewed the actions RG&E personnel took in response to the inclement weather. In addition to observing actions in the control room, the inspectors reviewed the following procedures to verify operators were adhering to the requirements contained in the documents:

- ER-SC-1, “Adverse Weather Plan”
- EPIP 1-17, “Planning for Adverse Weather”

b. Findings

Introduction. A green non-cited violation (NCV) was identified by the inspectors when they identified that although the Ginna site was experiencing high winds, control room operators did not implement the site adverse weather plan contained in procedures ER-SC-1, “Adverse Weather Plan,” and EPIP 1-17, “Planning for Adverse Weather,” until prompted by the inspector. Following implementation of the plan, operators were required to manually trip the reactor when an offsite power supply was lost because of storm-related damage.

Description. On October 15, 2003, the Ginna site experienced high winds because of a severe fall storm. The inspector noted that the 33 foot meteorological tower wind speed indicator, which can read wind speeds up to 50 mile per hour, was off-scale high. The inspector requested that the control room operators review the wind speed at the 150 and 250 foot levels on the primary plant computer system (PPCS). The inspector also requested that the operators review the wind speed entry conditions for ER-SC-1. When the operator checked the wind speed on the PPCS, he noted that the sustained wind speed was greater than 60 mph at the 150 and 250 foot levels on the meteorological tower. The inspector noted that this condition had caused an alarm to annunciate on the PPCS panel, which had been acknowledged by operators, but had not been acted upon.

The operator noted an entry condition for ER-SC-1 was sustained wind speed greater than 55 mph. Satisfied that an entry condition for ER-SC-1 had been met, the operator informed the control room foreman of this fact. The foreman then rechecked wind speeds on the PPCS, and entered ER-SC-1.

While in the procedure, the shift supervisor briefed the crew on several possible scenarios that could occur as a result of the windy conditions; including what actions they should take if an offsite power line was lost. Shortly after completing the briefing, offsite power line 751 failed, which necessitated a manual reactor trip.

While conducting followup to this event, the inspector noted that in addition to not implementing procedure ER-SC-1 when the site was experiencing high winds, prior to the onset of the adverse weather, the operators did not implement Emergency Plan Implementing Procedure (EPIP) 1-17, "Planning for Adverse Weather." Step 5.1 of EPIP 1-17 indicates the procedure should be entered whenever severe weather is forecast for the Rochester area.

Analysis. The performance deficiency in this event was a failure by operators to implement procedures ER-SC-1, and EPIP 1-17 when entry conditions had been met. This finding is associated with the "Protection Against External Factors" attribute of the initiating events cornerstone. This finding is greater than minor because it affected the objective of limiting the likelihood of those events that upset plant stability during power operation in that the severe weather caused a reactor trip.

In accordance with Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspectors conducted a SDP Phase 1 screening and determined that the finding was of very low safety significance (Green). Since operators responded appropriately to the loss of the 751 line, the finding is of very low safety significance because it did not contribute to the likelihood of a primary or secondary system LOCA initiator, or to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available. Additionally, the finding did not increase the likelihood of a fire or internal/external flood.

Enforcement. Plant Technical Specification (TS) 5.4.1 "Plant Procedures" states, in part, that "Written procedures shall be established, implemented, and maintained covering . . . the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978." Regulatory Guide 1.33, "Quality Assurance Program Requirements," identifies that procedures regarding "Acts of Nature, (e.g., tornado, floods, dam failures, earthquakes)," should be implemented. Contrary to the above, procedures ER-SC-1 and EPIP 1-17 were not implemented when entry conditions to the procedures were satisfied. Because the procedure noncompliances were of very low safety significance, and were entered into the Ginna corrective action program under AR 2003-2709, "Entered ER-SC.1 Due to High Wind Speed," this violation is being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000244/2003007-01, Failure to Implement Severe Weather Procedures.

2. Cold Weather Preparations

a. Inspection Scope

(71111.01 - 1 sample) The inspectors reviewed RG&E's preparations for cold weather by walking down the plant using Ginna Procedure A-54.4.1, "Cold Weather Walkdown Procedures," as a reference. Procedure M-1306.1, "Ginna Station, Maintenance Department Winterizing Inspection Program," was also used to determine if the plant was ready for adverse weather conditions. This inspection, which was conducted in late October, identified that preparations had started, but were not scheduled for completion until November 1, 2003. Followup walkdowns of areas where preparations were not completed during the October inspection, were conducted by the inspector during the first week of November.

b. Findings

No findings of significance were identified.

1R02 Evaluation of Changes, Tests, or Experiments (71111.02 - 5 samples)

a. Inspection Scope

The inspectors reviewed two safety evaluations associated with the mitigating systems cornerstone (no safety evaluations associated with the initiating event and barrier integrity cornerstones were available for review at the time of this inspection) to verify that changes to the facility or procedures as described in the Updated Final Safety Analysis Report (UFSAR) were reviewed and documented in accordance with 10 CFR 50.59. The inspectors also verified that the safety issues pertinent to the changes were properly resolved or adequately addressed. The safety evaluations were completed during the past three years, and were selected based on the safety significance of the changes and the risk to structures, systems, and components.

The inspectors also reviewed three screen-out and equivalency evaluations for changes, tests, and experiments for which RG&E determined that safety evaluations were not required. This review was performed to verify that RG&E's threshold for performing safety evaluations was consistent with 10 CFR 50.59.

The listing of the safety evaluations and screen-out evaluations reviewed is provided in the attachment to this report.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

Partial System Walkdown. (71111.04Q - 1 Sample) The inspectors completed a walkdown of the “B” reactor coolant pump lube oil collection system. The inspection reviewed the alignment of system valves to ensure proper in-service or standby configurations as described in plant procedures and drawings. During the walkdown, the inspectors also evaluated material conditions and general housekeeping of the system and adjacent spaces. The condition of the “B” reactor coolant pump lube oil collection system was examined because of its importance in preventing containment fires. The Ginna probabilistic risk analysis indicates plant fires are the most likely initiating events that could lead to a core damage event.

Complete System Walkdown. (71111.04S - 1 Sample) The inspectors conducted a detailed walkdown of the alignment and condition of the 125-V dc system. The 125-V dc system was chosen because of its high risk significance. In addition to verifying proper system alignment as required by the plant TS, UFSAR, and RG&E procedures and drawings, the inspector reviewed system maintenance and action reports (AR)s documenting grounds on the “A” and “B” batteries.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q - 2 samples)

a. Inspection Scope

The inspectors performed walkdowns of the following fire areas to determine if there was adequate control of transient combustibles and ignition sources. The material condition of fire protection systems, equipment and features, and the material condition of fire barriers were also inspected against industry standards and the Ginna fire protection program. In addition, the passive fire protection features were inspected, including the ventilation system fire dampers, structural steel fire proofing, and electrical penetration seals. The following plant areas were inspected:

- Air Handling Room
- Relay Room

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11)

1. Resident Inspector Simulator Scenario Observations

a. Inspection Scope

(71111.11 - 2 samples) On November 4, 2003, the inspectors observed two licensed operator annual simulator scenario tests. The tests observed were scenarios ES3123-07, “Steam Generator Tube Rupture,” and ECA 1112-081213-02, “LOCA Outside Containment.” The inspectors reviewed the critical tasks associated with the scenarios, observed the operators’ performance, and observed the post-evaluation critique. The inspectors also reviewed and verified compliance with Ginna procedure OTG-2.2, “Simulator Examination Instructions.”

b. Findings

No findings of significance were identified.

2. Licenced Operator Training Performance Evaluation

a. Inspection Scope

(71111.11 - 2 samples) The following inspection activities were performed using NUREG-1021, Rev. 8, Supplement 1, "Operator Licensing Examination Standards for Power Reactors," Inspection Procedure Attachment 71111.11, "Licensed Operator Requalification Program," NRC Manual Chapter 0609, Appendix I, "Operator Requalification Human Performance Significance Determination Process (SDP)," and 10 CFR 55.46, “Simulator Rule,” as acceptance criteria.

The inspectors reviewed documentation of operating history since the last requalification program inspection. The inspectors also discussed facility operating events with the resident staff. Documents reviewed included NRC inspection reports and ARs 2003-0369, 2003-0476, 2003-0819, and 2003-1804 that involved human performance issues.

The inspectors reviewed three examples of the 2003 comprehensive written exams, and one example of the 2003 annual operating test (consisting of three scenarios and five job performance measures). The inspectors observed the administration of the comprehensive written exam and annual operating test to one operating crew. The review was to ensure that the quality of the written exams, the annual operating tests, and the administration and evaluation of the operating tests met or exceeded the criteria of the examination standards and 10 CFR 55.59.

The inspectors interviewed two instructors, one evaluator, two training supervisors, three ROs, and two SROs for feedback regarding the implementation of the licensed operator requalification program, and to ensure training staff modified the program, when appropriate, to recommended changes. In addition, four plant and industry events or changes were reviewed to ensure that these items were adequately addressed in the requalification training program.

Inspectors assessed the adequacy of remedial training packages for two individuals that failed an evaluation during the current two-year cycle.

Inspectors reviewed the following documents and records to verify operators were complying with license conditions:

- A sample (seven) of attendance records for the current training cycle
- A sample (six) of medical records. Records were complete, restrictions noted by the doctor were reflected on the individual's license, and the physical exams were given within the last 24 months
- A sample of license renewals (three records) and proficiency watch-standing documents (five records)

The inspectors observed simulator performance during the conduct of the examinations, and reviewed simulator performance tests and discrepancy reports to verify compliance with the requirements of 10 CFR 55.46. Ginna is committed to the ANSI 3.5-1985 standard. The inspectors reviewed simulator configuration control and performance testing through interviews and the review of: facility simulator procedures; open and closed simulator ARs and discrepancy reports; and the review of test results. Specifically, the following tests were reviewed:

Normal operations tests:

- 100% Steady State Accuracy Check
- Residual Heat Removal System Test
- Reactor Coolant System Test

Transient tests:

- Manual Reactor Trip
- Simultaneous Closure of Both Main Steam Isolation Valves
- Steam Generator Tube Rupture

Malfunction tests:

- Rod-15 - Uncoupled Rod
- MIS-1 - Loss of Instrument Air
- EDS-4 - Loss of Emergency Bus
- STM-1 - Steam Flow Channel Failure

Core Performance tests:

- Startup Test - Initial Criticality and Low Power Physics
- NSSS-BOP Energy Balance

On December 16, 2003, inspectors conducted an in-office review of RG&E test results for 2003. The inspection assessed whether pass rates were consistent with the guidance of NRC Manual Chapter 0609, Appendix I, "Operator Requalification Human Performance Significance Determination Process (SDP)." The inspectors verified that:

- Crew failure rate was less than 20%. (Crew failure rate was 12.5%)
- Individual failure rate on the dynamic simulator test was less than or equal to 20%. (Individual failure rate was 5.7%)
- Individual failure rate on the walk-through test was less than or equal to 20%. (Individual failure rate was 0%)
- Individual failure rate on the comprehensive written exam was less than or equal to 20%. (Individual failure rate was 2.9%)
- Overall pass rate among individuals for all portions of the exam was greater than or equal to 75%. (Overall pass rate was 91.4%)

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12 - 2 samples)

a. Inspection Scope

The inspectors evaluated RG&E’s work practices and follow-up corrective actions for selected system, structure, or component (SSC) issues to assess the effectiveness of RG&E's maintenance activities. The inspectors reviewed the performance history of those SSCs and assessed RG&E’s extent of condition determinations for those issues with potential common cause or generic implications to evaluate the adequacy of RG&E’s corrective actions. The inspectors reviewed RG&E's problem identification and resolution actions for these issues to evaluate whether RG&E had appropriately monitored, evaluated, and dispositioned the issues in accordance with RG&E procedures and the requirements of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance." In addition, the inspectors reviewed selected SSC classification, performance criteria and goals, and RG&E's corrective actions that were taken or planned, to verify whether the actions were reasonable and appropriate. The following issues were reviewed:

Equipment Issues

- Primary containment isolation valve (PCIV) AOV-966C exceeded allowable leakage limit (AR 2003-0283); compensatory measure implemented to utilize manual valve V-956D in lieu of the automatic PCIV (AOV-966C), with subsequent failure of the manual operator for V-956D.

- Air Ejector Radiation Monitor R-15A, has experienced several failures over the last eighteen months because of water intrusion, software configuration control, and flow balancing issues.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation (71111.13 - 2 samples)

a. Inspection Scope

The inspectors evaluated the effectiveness of RG&E's maintenance risk assessments required by paragraph a(4) of 10 CFR 50.65. This inspection included discussions with control room operators and scheduling department personnel regarding the use of RG&E's online risk monitoring software. The inspectors reviewed equipment tracking documentation, daily work schedules, and performed plant tours to gain reasonable assurance that actual plant configuration matched the assessed configuration. Additionally, the inspectors verified that RG&E's risk management actions, for both planned and/or emergent work, were consistent with those described in procedure IP-PSH-2, "Integrated Work Schedule Risk Management." Risk assessments for the following out of service systems, structures, and/or components were reviewed:

- Planned maintenance on the "A" charging pump conducted on October 29, 2003, which resulted in an unexpected orange risk characterization when sprinkler system S-01 was removed from service.

- Planned maintenance on the "A" EDG conducted on December 5, 2003.

b. Findings

Introduction. A green NCV was identified when the inspectors noted that removing the "A" emergency diesel generator (EDG) from service did not significantly increase the calculated plant risk. The calculation model had been replaced by plant Probabilistic Risk Assessment (PRA) personnel at the end of the recent outage with an incorrect version.

Description. During the 2003 refueling outage, RG&E PRA personnel tested a revised model for conducting risk analysis during plant operations. The new model improved the calculation of plant risk by recalculating cutsets for the PRA when a piece of equipment was removed from service. The old model used predetermined cutsets, which may not account for all equipment that was out-of-service. At completion of the testing, and prior to coming out of the outage, the new program was installed. It was at this point that the wrong file was moved from the testing file onto the station intranet for RG&E personnel to use. The operators and schedulers began to use the new program at startup, and continued to use it until December 5, 2003, when the inspectors noted that removing the EDG from service had no significant impact on plant risk. Under the previous model, the plant risk would increase to almost yellow from green when an EDG was removed from service. This change in plant risk was consistent with risk associated with the EDG system in Ginna's PRA analysis.

Analysis. After management review, this finding was determined to be greater than minor, because the plant risk analysis assessments, which RG&E schedulers and operations personnel had performed on several occasions, were incorrect, and in one case on November 18, 2003, unbeknownst to plant personnel, the plant was in an elevated risk condition. If left uncorrected, this finding could become a more significant safety concern since with the incorrect software installed, operators could not correctly assess the impact on plant risk of maintenance on mitigating systems. The safety significance of the finding was not greater than green, because the plant was not in a high risk condition at any time during the period that the wrong program was installed. This finding is similar to example 3.a of NRC Manual Chapter 0612, "Power Reactor Inspection Reports" where the calculational impact could have been significant and the program required replacement.

Enforcement. 10 CFR 50.65 (a)(4) states, in part, that "Before performing maintenance activities, the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities." Contrary to the above, from October 16, 2003 until December 8, 2003, RG&E did not correctly assess and manage the risk from maintenance activities because the software program that was being used to assess the risk was not functioning properly due to use of incorrect input assumptions. Because this procedure noncompliance was of very low safety significance, and was entered into the Ginna corrective action program under AR 2003-3293, "Incorrect EOOS Models Being Used by Control Room and Scheduling," this violation is being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000244/2003007-02, Failure to Correctly Assess Risk of Maintenance Activities.

1R14 Personnel Performance During Non-Routine Plant Evolutions (71111.14 - 2 samples)

a. <u>Inspection Scope</u>

For the following two non-routine evolutions, the inspectors observed operator performance, reviewed operator logs, plant computer data, and strip charts to verify that operators responded in accordance with plant procedures:

- The inspectors reviewed operator response to a reactor trip that was manually initiated on October 15, 2003, at 10:26, when the 751 offsite power circuit was lost. Before operators manually initiated the trip, reactor power was being held at three percent while plant personnel were adjusting steam generator chemistry. Prior to the trip, operators had entered ER-SC-1 "Adverse Weather Plan," due to sustained winds greater than 55 mph. Inspector observations regarding operator use of ER-SC-1 are provided in Section 1R01 of this report.

  At 10:25 offsite power circuit 751 was lost when a tree fell on the line. This event led to a loss of power to the "B" reactor coolant pump, which tripped on undervoltage, and a loss of power to safeguards busses 16 and 17 which caused the "B" diesel generator to start. When the "B" reactor coolant pump was lost, operators entered AP-RCS.2, "Loss of Reactor Coolant Pump Flow," which instructed operators to trip the reactor.

  The inspectors verified that operators took appropriate actions to restore offsite electrical power to busses 16 and 17. Operator response to the trip was complicated by a valid fire alarm. A ventilation fan in the guardhouse failed and generated sufficient fumes to cause smoke detectors to alarm. The inspectors accompanied fire brigade personnel when they responded to the guardhouse, and verified the brigade utilized appropriate methods to diagnose the source of the smoke. Once plant conditions were stable, the inspectors walked down control room panels to ensure plant temperatures and pressures were within expected parameters.

  During the period of plant recovery, the inspectors monitored the RG&E post-trip review process. Plant response and actions were reviewed by the inspectors for compliance with procedures and proper system response. The inspectors reviewed the plant computer sequence of events printout and compared it to plant data collected from plant logs.

  On the night of October 15, 2003, the reactor was restarted. Once steam generator chemistry conditions were met, reactor power was increased and the generator was synchronized to the electrical grid. The inspectors observed the operators as they transitioned the plant from Mode 2 to Mode 1, and placed the turbine onto the electrical grid.

- On October 14, 2003, with the plant in Mode 2 at 2% reactor power, surveillance testing revealed that the turbine driven auxiliary feedwater (TDAFW) pump flow path to the "B" steam generator was isolated. Control room personnel recognized that they were not in compliance with Technical Specification (TS)

3.7.5 for TDAFW flow path operability requirements and that TS 3.0.4, which prohibits mode change while not meeting all requirements for the higher mode, had not been met during the startup. Prompt action was taken to restore the flow path by opening the closed isolation valve. The underlying cause was determined to be personnel error, complicated by conducting lineup procedures at the same time that a procedure was being utilized to fill steam generators in preparation for plant startup, which did not fully restore the TDAFW system to the appropriate status.

b. Findings

A finding regarding isolation of the TDAFW pump is discussed in Section 4OA3 of this report. No other findings of significance were identified.

1R15 Operability Evaluations (71111.15 - 4 samples)

a. Inspection Scope

The inspectors reviewed operability determinations to verify that the operability of systems important to safety was properly established, that the affected components or systems remained capable of performing their intended safety function, and that no unrecognized increase in plant or public risk occurred. In addition, the inspectors reviewed the following operability evaluations to determine if system operability was properly justified in accordance with IP-CAP-1.1, "Technical Evaluation for Current Operability and Past Operability Determination Worksheet":

- Action Report (AR) 2003-2765, "S/G "A" Feed Regulating Valve is Open more than in Past Cycles"
- AR 2003-3112, "Control Building Roof Leaks"
- AR 2003-3129, "Block Heater Not Working on Diesel Fire Pump"
- AR 2003-1288, "PASS Boron Analyzer Failed"

b. Findings

No findings of significance were identified.

1R16 Operator Work-Arounds (71111.16 - 1 sample)

a. Inspection Scope

The inspectors conducted a control room walkdown and interviewed operators to identify operator work-arounds. The inspectors reviewed control room deficiencies, maintenance identification tags on main control boards, degraded conditions on equipment important to safety, temporary alterations, RG&E-identified operator work-arounds and operator challenges, and selected corrective action reports. The inspectors compared their observations to the requirements in A-52.16, "Operator Work-around & Challenge Control," and the A-52.16 Attachment 3, 4, and 5 Lists, dated September 3, 2003 (pre-refuel outage) and October 20, 2003 (post-refuel outage).

The inspectors evaluated the operators' ability to implement normal, off-normal, and emergency operating procedures, during postulated plant transients with the existing equipment deficiencies, to determine whether the functional capability of a system or an operator's response to an initiating event would be affected. In addition, the inspectors evaluated the cumulative and synergistic effects of the identified operator work-arounds to determine whether there was an effect on multiple mitigating systems.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19 - 3 samples)

a. Inspection Scope

The inspectors observed and/or reviewed post-maintenance tests associated with work activities to verify that equipment was properly returned to service, and that proper testing was specified and conducted to ensure that the equipment could perform its intended safety function following maintenance. Post-maintenance testing associated with the following maintenance activities was reviewed:

- WO 2003-2487, "'B' AFW LO Pump Modification"
- WO 2003-2742, "Repair 'A' CCP Varidrive"
- WO 2003-0189, "Test Main Steam Safety Valves"

b. Findings

Introduction. A Green finding was identified by the inspectors when it was noted that RG&E had performed maintenance on four main steam safety valves prior to performing required surveillance testing. This practice may mask the as-found condition of the valves, and affects the results of the surveillance tests.

Description. While observing RG&E personnel test the lift setpoints of the eight main steam safety valves, the inspector determined that RG&E had performed preventative maintenance on four safety valves before conducting the surveillance test. The inspectors determined that this practice was not consistent with the sequence discussed in NRC Information Notice 97-16, “Preconditioning of Plant Structures, Systems and Components Before ASME Code In-Service Testing or Technical Specification Testing,” and NRC Inspection Manual (IM) 9900 guidance regarding preconditioning. The inspectors determined that the preventative maintenance performed on the valves, which included jacking and lapping the seat of the safety valves, could precondition the valves and affect their as-found condition. RG&E had not evaluated whether it was acceptable to perform the maintenance prior to performing the required surveillance test. RG&E documented the inspector’s observation in Action Report (AR) 2003-2681, “Unable to Obtain As-Found Data for Main Steam Safety Valves Due to Preventative Maintenance Performed on the Valves During the 2003 RFO.”

Analysis. The performance deficiency in this event is a failure to properly sequence the performance of work and surveillance activities. This finding is greater than minor, because it is associated with the “Equipment Performance” (reliability) attribute of the mitigating systems cornerstone, and it could adversely affect the cornerstone objective because failure to conduct as-found testing may mask any valve degradation. This could adversely impact the reliability of the steam generator overpressure protection system to prevent undesirable consequences. In accordance with Manual Chapter 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” the inspectors conducted an SDP Phase 1 screening and determined that the finding is of very low safety significance (Green). The finding is of very low safety significance because it was not a design or qualification deficiency, it did not represent a loss of safety function, and was not potentially risk significant due to seismic, flood, fire, or weather related initiating event. Further, the SDP process screens to green since the issue involved inadequate testing, and did not degrade the ability of the main steam safety valves to perform their intended function for the next operating cycle.

Enforcement. No violation of regulatory requirements occurred. The inspectors determined that the finding did not represent a noncompliance because IN 97-16 and IM Part 9900 are not regulatory requirements. (FIN 05000244/2003007-03, Failure to Properly Sequence Work Activities During Surveillance Testing)

1R20 Refueling and Outage Activities (71111.20 - 1 sample)

a. Inspection Scope

Licensee Control of Outage Activities

Periodically during the refuel outage, the inspectors verified that the status of electrical systems and emergency core cooling systems met technical specification requirements. Several plant areas that are not routinely accessible when the reactor is at power, including the volume control tank and chemical and volume control system filter rooms, were toured by the inspector to verify systems in the rooms were intact and plant personnel had entered discrepant conditions into the corrective action program.

Reduced Inventory and Mid-Loop Operations

Repairs to two primary loop temperature detectors required the plant to be placed in mid-loop operations. For a period of about two days, the plant entered into a reduced inventory/mid-loop status. The preparations and operation in this condition was monitored by the inspectors. The plant responded as expected, and the operators demonstrated good control over level in the loops throughout the drain down and subsequent flood up process. Level indication systems were well controlled throughout the reduced inventory/mid-loop conditions.

Monitoring of Heatup and Startup Activities

Prior to entering Mode 4, the inspectors toured the containment to verify that, to the extent practicable, RG&E had removed material from containment that could potentially clog the containment recirculation sump during certain accident scenarios. During the tour, the inspectors entered both the “A” and “B” containment sumps and verified they had been cleaned of loose material. The inspector also visually assessed the condition of systems in containment, and verified that RG&E had identified and entered discrepant conditions into the corrective action program.

The inspector observed portions of the plant heatup and physics testing program and operations associated with synchronizing the turbine with the electrical grid.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22 - 5 samples)

a. Inspection Scope

The inspectors observed and/or reviewed the five surveillance tests listed below that are associated with selected risk-significant systems, structures, and components (SSCs) to verify that technical specifications were followed, and that test acceptance criteria were properly specified. The inspectors also verified that proper test conditions were established as specified in the procedures, that no equipment preconditioning activities occurred, and that acceptance criteria had been met.

- M-1306, “Ginna Station Material Condition Inspection Program,” performed on November 7, 2003
- PT-16Q-A, “Auxiliary Feedwater Pump A - Quarterly,” performed on October 30, 2003
- RSSP-2.2, “Diesel Generator Load and Safeguard Sequence Test,” performed on October 8, 2003
- PT-16.3B, “AFW Pump ‘B’ Discharge MOV and Check Valve Test,” performed on October 10, 2003

- PT-2.3.1Q, “Post Accident Charcoal Filter Dampers - Quarterly,” performed on December 10, 2003

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23 - 1 sample)

a. Inspection Scope

The following temporary modification (TM) was reviewed and visually inspected by the inspectors during the triennial fire protection program inspection documented in Inspection Report 50-244/03-013. This TM was reviewed to verify that the it was installed in conformance with the instructions contained in procedure IP-DES-3, “Temporary Modifications”:

- 2003-0027, “Coffer Dam Cover for Transformer Yard Hatch”

b. Findings

No findings of significance were identified.

Cornerstone. Emergency Preparedness

1EP6 Drill Evaluation (71114.06 - 2 samples)

a. Inspection Scope

On November 4, 2003, the inspectors observed two licensed operator simulator scenarios that included a limited test of the Ginna emergency response plan. Scenarios ECA1112-08, “LOCA Outside of Containment,” and ES3123-08, “Steam Generator Tube Rupture,” were observed. The inspectors verified that the appropriate emergency classifications were identified, and external notifications to responsible parties were completed in a timely manner as required by the Ginna emergency response plan.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151 - 4 samples)

1. Safety System Unavailability

a. Inspection Scope

Using the criteria specified in Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 2, the inspectors verified the completeness and accuracy of the following safety system unavailability performance indicators (PI) associated with the mitigating system cornerstone:

- High Pressure Injection System
- Heat Removal System (AFW)

The inspection was accomplished by discussion with plant personnel and review of unavailability tracking documentation, operator logs, action reports, work orders, and selected surveillance procedures. The period reviewed included the first and fourth quarters of 2002, and the first, second, and third quarters of 2003.

b. Findings

No findings of significance were identified.

2. Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspector reviewed implementation of RG&E's occupational exposure control effectiveness performance indicator (PI) Program. Specifically, the inspector reviewed recent ARs, and associated documents, for occurrences involving locked high radiation areas, very high radiation areas, and unplanned exposures against the criteria specified in Nuclear Energy Institute (NEI) 99-02, to verify that all occurrences that met the NEI criteria were identified and reported as performance indicators. This inspection activity represents the completion of one (1) sample relative to this inspection area; completing the annual inspection requirement.

b. Findings

No findings of significance were identified.

3. RETS/ODCM Radiological Effluent Occurrences

a. Inspection Scope

The inspector reviewed relevant effluent release reports for the period October 1, 2002 through December 1, 2003, for issues related to the public radiation safety performance indicator, which measures radiological effluent release occurrences that exceed 1.5 mrem/qtr whole body or 5.0 mrem/qtr organ dose for liquid effluents; 5 mrads/qtr gamma air dose, 10 mrads/qtr beta air dose, and 7.5 mrads/qtr for organ dose for gaseous effluents. This inspection activity represents the completion of one (1) sample relative to this inspection area; completing the annual inspection requirement.

The inspector reviewed the following documents to ensure RG&E met all requirements of the performance indicator from the fourth quarter 2002 to the fourth quarter 2003 (four quarters):

- Monthly projected dose assessment results due to radioactive liquid and gaseous effluent releases
- Quarterly projected dose assessment results due to radioactive liquid and gaseous effluent releases
- Dose assessment procedures

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152 - 2 samples)

1. Selected Issue Follow-up Inspection - Failures of Instrument Bus Voltage Regulators, Action Reports (AR) 2001-1943 and 2001-2232

a. Inspection Scope

There are seven instrument bus voltage regulators (Twinco units) at Ginna, six installed units (MQ-400A through F), and one spare. Each installed unit supplies regulated voltage (118 V ±2%) ac power to a distribution panel, which provides ac power to safety-related instrumentation for the reactor protection system (RPS) and the engineered safety feature system (ESFS). Instrument Buses A and B each had two units and Instrument Buses C and D each had one. The Twinco units were original equipment units supplied by Westinghouse, and are no longer manufactured. The inspector selected this topic for review because proper operation of the Twinco units substantially contributes to plant safety.

On November 3, 2001, Twinco unit MQ-400E failed (AR 2001-1943) due to a shorted wire. The unit was replaced with a spare, and the shorted wire was repaired. Subsequently on December 20, 2001, RG&E replaced the spare Twinco unit with a regulator manufactured by Nuclear Logistic Inc. (NLI), using the one-for-one replacement process through equivalency evaluation TE-99-0072. However, the output voltage from the NLI unit was later found unsatisfactory (AR 2001-2232) due to its excessive sensitivity to harmonic distortion feedback. The replaced Twinco unit was then reinstalled.

RG&E worked with NLI to resolve the harmonic distortion problem on the NLI units. During the 2002 refueling outage, with assistance of an NLI representative, RG&E replaced Twinco Unit MQ-400C. However, the output voltage was again found unacceptable due to excessive harmonic distortion (heaviest on the 35^{th} harmonic.) As a result, RG&E decided not to use the NLI units and reinstalled the Twinco unit.

Subsequently, RG&E generated Plant Change Record (PCR) 2002-0050 and design analysis (DA-EE-2002-059) that authorized removing the Twinco units from the instrument power supplies. This decision was based on a conclusion that recently installed constant voltage transformers, which supplied ac power to the instrument buses, had output voltages that could meet the output voltage requirement of the Twinco units except for several circuits that had higher voltage drops. For those circuits, RG&E replaced the Foxboro power supply units (Models 610 and 610A) that required the higher input voltage with NUS power supply units (Model SPS500) that could tolerate lower input voltage. At the time of the inspection, one Twinco unit (MQ-400D) had been removed.

The inspector reviewed the action reports associated with failures of the Twinco units to determine whether RG&E's corrective actions were appropriate. The inspector also reviewed the design change packages and the associated safety evaluations (screening and equivalency evaluations) to determine the design adequacy. In addition, the inspector walked down the installed units to determine whether there were any abnormal conditions.

b. Findings

No findings of significance were identified.

2. Selected Issue Follow-up Inspection - Inverter MQ483 Failures (ARs 2000-1695 and 2000-0412)

a. Inspection Scope

Inverter MQ483 was installed about 15 years ago at Ginna to provide a more reliable ac power source to three safety-related instruments in instrument rack Y2. The installed unit and two spares were supplied by Solidstate Controls Inc. (SCI). The inspector selected this topic for review because proper operation of Inverter MQ483 substantially contributes to plant safety.

On March 27, 2000, one of the spares in the warehouse was found inoperative (no output voltage) during a routine testing (documented in AR 2000-0412). The unit was sent to the vendor for evaluation and repair on June 22, 2000, and was not returned until January 2, 2001. The failure mechanism was determined to be age-related degradation. On December 13, 2000, an installed inverter failed (documented in AR 2000-1965) due to overheating of internal transformers (also caused by overaged capacitors). The second failure caused a fire in the relay room and the release of the halon system, in addition to the loss of the safety-related instruments. The failed unit was replaced with a spare, and was sent to the vendor for evaluation and repair.

The inspector reviewed the action reports to determine the adequacy of RG&E's corrective actions. The inspector also verified that the required replacement of the capacitors in the inverters every eight years had been entered into RG&E's maintenance program.

b. Findings

No findings of significance were identified.

3. Cross Reference to Cross Cutting Issues Documented Elsewhere

Section 4OA3 of this report describes a finding that is related to the human performance cross-cutting area. Specifically, operators did not properly sequence the performance of two procedures that involved realigning valves in the TDAFW system prior to the commencing a plant heatup. As a result, the TDAFW system was inoperable during two Mode changes.

4. Review of Corrective Action Documents Associated With The Reactor Vessel Closure Head (RVCH) Replacement

a. Inspection Scope

The inspectors reviewed corrective action documents associated with the reactor vessel closure head (RVCH) replacement to ensure that the Rochester Gas and Electric Company (RG&E) appropriately identified, evaluated, and initiated actions to correct problems associated with these activities. The inspectors also discussed RVCH related action reports (ARs) with RG&E engineers and managers.

b. Findings

No findings of significance were identified.

5. Problem Identification and Resolution Document Screening

a. Inspection Scope

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's corrective action program. This review was accomplished by reviewing hard copies of each condition report, attending daily screening meetings and accessing the RG&E computerized database.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up

1. (Closed) LER 05000244/2003002, Major Power Grid Disturbance Causes Loss of Electrical Load and Reactor Trip

On August 14, 2003, an automatic reactor trip occurred as a result of a major electrical disturbance that affected the Northeastern United States and Canada. Following the reactor trip, the "B" motor driven auxiliary feedwater (MDAFW) pump was damaged when plant operators failed to align the auxiliary feedwater system as required by procedure. Details regarding the "B" MDAFW pump failure are discussed in NRC inspection report 50-244/03-06. This LER was reviewed by the inspectors, and with the exception of the procedure violation documented in NRC inspection report 50-244/03-06, no findings of significance were identified. RG&E documented the failure of the "B" MDAFW pump in Action Report (AR) 2003-1821. This LER is closed.

2. (Closed) LER 05000244/2003003, Containment Sump As-Found Condition not in Accordance With Design Requirements

On September 18, 2003, during an inspection of the "B" containment recirculation sump, RG&E personnel discovered three holes in steel deck plating that would allow flow to bypass the recirculation sump inner screen. Additional RG&E and NRC inspection of the sump area revealed other deficiencies including solid and foreign material on the sump floor, and a screen mesh size that did not match the criteria contained in the plant Updated Final Safety Analysis Report (UFSAR). An NRC special inspection team assessed the significance of the sump deficiencies, and documented their findings in NRC Inspection Report 50-244/03-012. This LER was reviewed by the inspectors, and with the exception of the issues already discussed in NRC Inspection Report 50-244/03-012, no findings of significance were identified. RG&E documented the sump design issues in several ARs including AR 2003-2170. This LER is closed.

3. (Closed) LER 05000244/2003004, Auxiliary Feedwater Flowpath Inoperable During Mode Changes Due to Personnel Error, Resulted in Condition Prohibited by Technical Specifications

a. Inspection Scope

The inspectors reviewed the LER and Action Report 2003-2703 which documented the unintended isolation of the turbine driven auxiliary feedwater (TDAFW) pump flowpath discussed in Section 1R14 of this report. RG&E's assessment of cause and review of safety consequences was adequate. RG&E's intended corrective actions may include a revision to the procedure for adding water to the steam generators, T-41G, which will address realignment of the system when the startup lineup procedures of T-41A are in effect.

b. Findings

Introduction. A Green self-revealing violation of very low safety significance was identified when plant operators discovered during a surveillance test that the TDAFW was not operable prior to making a mode change. This discovery was a violation of plant Technical Specification 3.0.4.

Description. On October 14, 2003, a self-revealing finding was identified during the performance of surveillance test PT-16Q-T, "Auxiliary Feedwater Pump - Quarterly," when RG&E personnel identified that valve V-4006 (manual isolation valve in the flowpath to the"B" steam generator) was closed when it was required by the plant TS to be in the open position. An RG&E investigation revealed the event occurred on October 10, when two procedures T-41A, "Alignment of Auxiliary Feedwater System Prior to Power Operation," and T-41G, "Alignment of Auxiliary Feedwater System to Fill S/G's While in Modes 4, 5, 6, Defueled," were performed out of sequence. Specifically while aligning plant systems in preparation for plant heatup, operators performed T-41A, which ensured valve V-4006 was open. Later in the day, procedure T-41G was performed, which closed V-4006. Although operators were aware that V-4006 was closed and had annotated on a log book to open V-4006 by reperforming T-41A, this action was not completed. Subsequently the TDAFW pump was not operable as required by TS 3.0.4 when the plant entered Modes 3 and 2.

Analysis. The performance deficiency associated with this event was a personnel error, which rendered the TDAFW system inoperable during a plant heatup and resultant Mode changes. This finding, associated with the "Configuration Control" attribute of the mitigating systems cornerstone, is greater than minor because it affected the objective of ensuring the reliability and capability of systems to prevent undesirable consequences in that the TDAFW system was inoperable for three days. The finding is of very low safety significance because it was not a design or qualification deficiency, it did not represent a loss of safety function (the remaining diverse trains of AFW remained operable), and was not potentially risk significant due to seismic, flood, fire, or weather related initiating event. Further, the exposure time was less the limiting condition for operation (LCO) action time of seven days.

Enforcement. TS 3.0.4 states when plant systems or components are not operable, unless otherwise specified in the TS, the plant shall not enter a higher level Mode. Contrary to the above, on October 12 and 14, 2003, RG&E changed Modes with the TDAFW pump inoperable, and the Mode changes were not required to comply with actions. Because the failure to maintain the TDAFW pump operable during the Mode changes was of very low safety significance and has been entered into the RG&E corrective action program as Action Report 2003-2703, “TDAFW Pump Valve Found Out of Position,” this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000244/2003007-04, TDAFW System Was Not Operable During Mode Change.

4. (Closed) LER 05000244/2003005, Manual Reactor Trip Resulting From Loss of Off-Site Power Circuit 751

On October 15, 2003, with the plant at approximately 2% power during a reactor startup, high winds resulted in the loss of off-site circuit 751. Because of the electrical lineup that was in effect, the electrical bus that was powering the “B” reactor coolant pump was de-energized. As a result, operators manually tripped the reactor as required by Abnormal Operating Procedure AP-RCS.2, “Loss of Reactor Coolant Flow.” This LER was reviewed by the inspectors, and no findings of significance were identified. RG&E documented the reactor trip in AR 2003-2714. This LER is closed.

4OA5 Other Activities

1. Reactor Containment Sump Blockage (Temporary Instruction 2515/153)

a. Inspection Scope

On June 9, 2003, the NRC issued Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors," to all pressurized-water reactor (PWR) licensees requesting that they provide a response within 60 days. On August 8, 2003, RG&E responded and stated that they had implemented the following interim compensatory measures (Option 2 as outlined in Bulletin 2003-01): (1) operator and staff training on indications of and responses to sump clogging; (2) procedural modifications, if appropriate, that would delay the switchover to containment sump recirculation; (3) ensuring that alternative water sources are available to refill the refueling water storage tank (RWST) or to otherwise provide inventory to inject into the reactor core and spray the containment atmosphere; (4) more aggressive containment cleaning and increased foreign material controls; (5) ensuring containment drainage paths are unblocked; and (6) ensuring sump screens are free of adverse gaps and breaches.

During the week of October 6, concurrent with NRC Special Inspection 50-244/03-012, the inspectors reviewed RG&E’s activities in response to NRC Bulletin 2003-01 to assess whether RG&E effectively implemented reasonable compensatory measures. The inspectors independently verified that RG&E had implemented the interim compensatory measures or had planned and scheduled these activities consistent with their response. The inspectors conducted two independent internal containment sump inspections and containment walkdowns. The inspectors reviewed operator training records, RG&E Commitment and Action Tracking System (CATS) reports, procedures, documentation of containment inspections and foreign material control activities, and containment sump related corrective action reports. (See the attachment for a listing of documents and corrective action reports reviewed). The inspectors also discussed RG&E's Bulletin response with the NRR Project Manager and reviewed NRR 's draft acknowledgment letter.

During the inspection period, the resident inspectors interviewed seven operators, representing three operating shifts, to assess their awareness of reactor containment sump blockage issues and expected operator mitigating actions.

b. Findings & Observations

No findings of significance were identified with respect to RG&E's response to NRC Bulletin 2003-01. NRC inspectors documented two findings associated with RG&E's corrective actions for longstanding containment sump related deficiencies in NRC Special Inspection Report 50-244/03-012.

Operator Training

During Operator Requalification Cycle 2003-2 (2/24/03 - 4/5/03), reactor operators received classroom and simulator training on containment sump clogging issues. The classroom training covered sump clogging, debris sources, containment sump design, history of the sump blockage issues, indication of sump blockage, and operator response to sump blockage. Based on interviews, the inspectors determined that operators were aware of containment sump blockage issues and their expected actions.

During their review of Bulletin 2003-01, RG&E identified a potential modification to the existing residual heat removal (RHR) instrumentation (currently used only during non-power operations) to reduce the risk of sump screen blockage. This modification, would allow operators to use the RHR instrumentation as a sump fouling diagnostic tool following a LOCA. Engineering planned to complete the technical evaluation of this potential modification (CATS ID 11339).

Potential Procedural Modifications

For small to medium sized loss-of-coolant accidents (LOCAs), RG&E procedure ES-1.2, "Post LOCA Cooldown and Depressurization," provides guidance as to how to delay RWST depletion before switchover. This procedure requires that action be taken to cool down and depressurize the reactor coolant system (RCS) to reduce break flow and thereby reduces the injection flow from the RWST. Based on preestablished criteria that maintain core cooling, operators sequentially stop safety injection pumps to reduce injection flow. For smaller LOCAs, RG&E determined that it is possible to cool down and depressurize the RCS to cold shutdown conditions before the RWST is drained to the switchover level.

RG&E reviewed their procedures to identify potential changes that could be made to delay the switchover to containment sump recirculation following a large break LOCA. RG&E determined that no immediate compensatory procedure changes could be made without adverse risk consequences. The Ginna Station accident mitigation strategy does not allow actions on large break LOCAs that would delay RWST inventory depletion. RG&E procedure ECA-1.1, "Loss of Emergency Coolant Recirculation," provides guidance as to how to delay RWST inventory depletion. If operators identify blockage following switchover, procedure ES-1.3, "Transfer to Cold Leg Recirculation," requires that operators reduce flow to the minimum for decay heat removal and to consult with the Technical Support Center (TSC) for further actions. If recirculation capability is lost following switchover, ES-1.3 directs the operator to transition to ECA-1.1.

Interim compensatory measures (e.g., shutting down one train of emergency core cooling and/or containment spray) require changes to the generic emergency response guidelines (ERGs) and emergency procedure guidelines (EPGs). The Westinghouse Owners Group (WOG) planned to evaluate potential changes involving containment sump debris blockage by March 31, 2004. RG&E planned to participate in the associated WOG activities (CATS ID 11337). Following this WOG review, RG&E planned to evaluate the changes for potential implementation at Ginna Station. RG&E planned to complete their evaluation by October 31, 2004 (CATS ID 11338).

Alternative Water Sources

In response to a loss of sump recirculation, ECA-1.1 directs the operator to make up to the RWST as necessary using available alternate water sources. Procedure ECA-1.1 also identifies the water sources and available flow paths for injecting into the RCS. Following WOG review of potential EOP modifications associated with alternative water sources, RG&E planned to evaluate the changes for potential implementation at Ginna Station (CATS ID 11338).

More Aggressive Containment Cleaning and Increased Foreign Material Controls

In their Bulletin response, RG&E stated that they have an aggressive program to maintain containment cleanness and foreign material control using procedures IP-HSC-3, "House Keeping Control"; ND-HSC, "Housekeeping and System Cleanness and Foreign Material Exclusion"; and A-3.1, "Containment Storage Inspection." Based on a review of a post-outage (April 2002) containment video and interviews with responsible personnel (ALARA and housekeeping), RG&E determined that their containment was well maintained and free of debris that could be easily transported to the sump. In addition, RG&E had initiated efforts to eliminate paper equipment tags, tie wraps and other sources of debris. Prior to startup from their Fall 2003 refueling outage (RFO), RG&E further assessed containment cleanliness, quantified and removed remaining items.

During a containment sump inspection in the Fall 2003 RFO, RG&E discovered that the containment sump floor was covered by approximately one inch of solid material. RG&E

initiated action report (AR) 2003-2196 and subsequently determined that the solid material was predominantly boric acid residue. During the NRC Special Inspection follow-up, inspectors noted that miscellaneous foreign objects had been embedded within the solid material. RG&E also evaluated this containment sump debris under AR 2003-2196. During subsequent containment sump walkdowns, inspectors identified additional small items of debris and other deficiencies internal to the sump (see NRC Special Inspection Report 50-244/03-012, Section 4OA3.2). Based on these observations, the inspectors determined that RG&E did not have an aggressive program to maintain containment cleanliness and foreign material control. However, based upon their identified sump issues in September 2003 and their review of NEI 02-01, Revision 1, "Condition Assessment Guidelines: Debris Sources Inside PWR Containments" (CATS ID 10999), RG&E revised their containment sump inspection procedure (A-3.1). Inspectors reviewed RG&E's final containment storage and closeout inspection and independently walked down containment, and performed an internal sump inspection prior to plant restart. The inspectors observed that RG&E's attention to containment cleanliness and foreign material control had improved.

During a containment sump inspection, the inspectors directly observed RG&E efforts to identify and replace paper tags and old tie wraps with containment-approved substitutes. During containment walkdowns, the inspectors noted an abundance of old masking tape (used primarily for labeling) attached to plant equipment throughout all levels of containment. RG&E efforts in the Fall 2003 RFO focused primarily on removing these unwanted items (unapproved tags, ties, and tape) in the basement level of containment. RG&E was not able to completely eliminate these items from containment; however, they did conduct high-definition video recordings of containment in an effort to help quantify these items.

<u>Ensuring Containment Drainage Paths Are Unblocked</u>

RG&E procedure A-3.1, "Containment Storage and Closeout Inspection," contains inspection guidance to ensure that containment drainage paths to the sump are unblocked and that no unrestrained items are placed such that they could block a drainage path. The inspectors reviewed RG&E' s final containment storage and closeout inspection and independently walked down containment prior to plant restart. The inspectors did not identify any potential "choke points" where the drainage to the recirculation sump could become constricted due to debris.

<u>Ensuring Sump Screens Are Free of Adverse Gaps and Breaches</u>

In their Bulletin response, RG&E stated that they performed an initial evaluation of the sump screens using photographs taken during the April 2002 RFO and identified no obvious gaps or breaches. Because the photographs did not provide conclusive evidence that no gaps or breaches existed, RG&E planned to perform additional sump screen inspections using procedure A-3.1 during their Fall 2003 RFO.

During a containment sump screen inspection in the Fall 2003 RFO, RG&E identified several containment sump bypass flowpaths. The bypass paths consisted of three holes through the steel deck plate covering the RHR suction side of the sump. The total surface area of the bypass holes was approximately 26 square inches. RG&E initiated action report (AR) 2003-2170 and subsequently identified several other smaller screen bypass paths with a total area of less than 30 square inches (AR 2003-2213). RG&E repaired these bypass flowpaths during the Fall 2003 RFO.

During the NRC Special Inspection follow-up, inspectors identified two additional screen bypass flowpaths with a total area of less than 10 square inches. RG&E initiated corrective actions (ARs 2003-2603 and 2003-2608) and repaired these bypass flowpaths prior to plant restart (see NRC Special Inspection Report 50-244/03-012, Section 4OA3.1). Based upon their identified sump issues in September 2003 and their review of NEI 02-01, Revision 1, "Condition Assessment Guidelines: Debris Sources Inside PWR Containments" (CATS ID 10999), RG&E revised their containment sump inspection procedure (A-3.1). Inspectors reviewed RG&E's final containment storage and closeout inspection and independently performed an internal sump inspection prior to plant restart. The inspectors observed that RG&E, by outage end, demonstrated an increased awareness to sump screen bypass concerns.

2. Reactor Vessel Closure Head Walkdown (71007 - 1 Sample)

a. Inspection Scope

Recent industry events involving Primary Water Stress Corrosion Cracking (PWSCC) of Alloy 600 at other plants throughout the industry prompted RG&E to take the preemptive measure of replacing the RVCH during their Fall 2003 refueling outage (see NRC Inspection Report 50-244/03-006). By October 6, 2003, RG&E had completed reactor vessel re-assembly, including installation of the new RVCH. The inspectors performed a walkdown of accessible areas of the replacement RVCH, new control rod drive mechanism (CRDM) guide tubes, head assembly upgrade package (HAUP), relocated head vents, insulation and support package, core exit thermocouple nozzle assemblies (CETNAs), and the CRDM cooling coil shroud. The inspectors visually examined these components for primary pressure boundary leakage and to verify that RG&E implemented appropriate foreign material exclusion controls in these areas.

b. Findings

No findings of significance were identified.

3. Reactor Head Vent Valve Orientation

a. Inspection Scope

The inspectors reviewed Modification Design Change Notice 2446, "Orientation of Reactor Head Vent Solenoid Valve," and Plant Change Record (PCR) No. 2001-0042 work order (WO) 2030- 1196, to ensure they were properly implemented. Modification Design Change Notice 2446 was issued to change the orientation of the reactor head vent solenoid-operated valves (SOVs) 590, 591, 592, and 593 to eliminate interference with the reactor head stud tensioning tool, and PCR 2001-0042 was issued to complete the SOV electrical work. The inspectors also reviewed the adequacy of RG&E's corrective actions for the two deficiencies associated with the SOV electrical drawings and SOV switch labeling in the control room identified during performance of WO 2030-1196. This included a review of engineering's determination that the deficiencies were minor and had not adversely impacted head vent valve operation, and review of AR 2003-253 which was initiated to revise Drawing 11300-960 and to correct the control room labels. In addition, the inspectors reviewed procedure M-51-13-EQ, "Maintenance and Replacement of EQ Valcor Solenoid Operated Valves," Revision 9, for adequacy, and walked down the installed head vent valves and SOVs to assess the configuration changes.

b. Findings

No findings of significance were identified.

4. Rigging Issue Follow-up

a. Inspection Scope

The inspectors reviewed the adequacy of RG&E's corrective actions regarding a rigging sling which fell and landed around microprocessor rod position indication (MRPI) coil D10 (between CRDMs D-10 and C-09) on September 30, 2003, during RVCH installation activities. This included a review of AR 2003-2412 initiated to evaluate the condition and work order 2032524 initiated to test the MRPI coils on CRDM D-10 using procedure CME-55-02-MRPI Coil, "Westinghouse MRPI Coil Stacks Testing and Maintenance," as well as verification that RG&E retrieved the dropped sling and performed a visual inspection of the entire length of the D-10 and C-09 MRPI coil stacks for damage.

In addition, the inspectors reviewed the completed CME-55-02-MPRI Coil procedure to ensure the results were acceptable. This included a review of the resistance and megger test results, the MRPI detector interface card voltages, and the measuring and test equipment data (equipment used and calibration dates.)

b. Findings

No findings of significance were identified.

5. Electronic Characterization and Diagnostics Testing

a. Inspection Scope

The inspectors observed electronic characterization and diagnostics (ECAD) testing of electrical cables in the control rod drive (CRD) system. The cables connect the stationary, lift, and movable coils on the CRDMs to the CRD system panel. The testing measures the DC loop resistance of the cable, and the circuit impedance at various frequencies. The inspectors also reviewed the adequacy of RG&E's corrective actions when the stationary coil on CRDM F-12 showed an unacceptable value during the test. This included a review of AR 2003-2578 initiated to evaluate the condition, and a review of work order 200302640 issued to replace the set of three coils (stationary, lift, and movable) on CRDM F-12. The inspectors also verified all three coils were tested before they were installed.

Following the replacement of the three coils, the inspectors reviewed the results of the post maintenance tests (PMTs) on the CRDM and MRPI cables that were disconnected to access CRDM F-12, as well as the ECAD testing performed on the stationary coil of CRDM F-12 to ensure the results were acceptable.

b. Findings

No findings of significance were identified.

6. Control Rod Drive Mechanism Airflow Testing

a. Inspection Scope

With the installation of the new RVCH HAUP, the direction of the airflow to the CRDMs was reversed. Instead of blowing air down over the CRDMs, air is sucked from the bottom of the HAUP. Westinghouse performed a cooling fan analysis and determined that a minimum flow rate of 14000 cubic feet per minute (cfm) was required to keep the temperature at the CRDM 173^{o}F when the air at an ambient temperature of 120^{o}F enters the HAUP at the bottom.

The inspectors observed portions of the CRDM airflow testing performed using procedure SM-2001-0042-1, "CRDM Fan Flow Testing," and reviewed the test results to ensure the minimum acceptance criteria of 14000 cfm was met.

b. Findings

No findings of significance were identified.

4OA6 Meetings

a. Exit Meeting Summary

Periodically during the course of this inspection, the inspectors met with Ginna representatives to discuss certain aspects of the inspection. For example, on October 16, 2003, NRC regional inspectors summarized the preliminary findings regarding RG&E's response to Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors."

On January 8, 2004, the resident inspectors summarized the contents of this inspection report to Mr. Widay, and other members of his staff, who acknowledged the findings. The inspectors returned any proprietary items, and verified that no proprietary information is presented in this inspection report.

b. RG&E/NRC Management Meeting

On December 11 and 12th 2003, James Wiggins, Deputy Regional Administrator Region 1 and Mr. James Trapp, Chief, Reactor Projects Branch 1, conducted a tour of the Ginna site, and met with senior plant managers and station personnel.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

M. Flaherty	Nuclear Safety & Licensing Manager
B. Flynn	Primary Systems and Reactor Engineering Manager
J. Hotchkiss	Mechanical Maintenance Manager
F. Maciuska	Operations Manager
R. Marchionda	Nuclear Assessment Department Manager
R. Mecredy	Vice President Nuclear Operations
R. Ploof	Scheduling Manager
R. Popp	Production Superintendent
J. Smith	Maintenance Superintendent
T. White	Balance of Plant Systems Engineering Manager
J. Widay	VP, Plant Manager

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Opened and Closed

05000244/2003007-01	NCV	Failure to Implement Procedures for Severe Weather
05000244/2003007-02	NCV	Failure to Correctly Assess Risk of Maintenance Activities
05000244/2003007-03	FIN	Failure to Properly Sequence Work Activities During Surveillance Testing
05000244/2003007-04	NCV	Auxiliary Feedwater Flowpath Inoperable During Mode Changes, Due to Personnel Error, Resulted in Condition Prohibited by Technical Specifications

Closed

05000244/2003-002	LER	Major Power Grid Disturbance Causes Loss of Electrical Load and Reactor Trip (Section 4OA3)

05000244/2003-003	LER	Containment Sump As-Found Condition not in Accordance With Design Requirements (Section 4OA3)
05000244/2003-004	LER	Auxiliary Feedwater Flowpath Inoperable During Mode Changes, Due to Personnel Error, Resulted in Condition Prohibited by Technical Specifications (Section 4OA3)
05000244/2003-005	LER	Manual Reactor Trip Resulting From Loss of Off-site Power Circuit 751 (Section 4OA3)

Discussed

NONE

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Documents

A-54.4.1	Cold Weather Walkdown Procedures
M-1306	Maintenance Department Winterizing Inspection Program

Section 1R02: Evaluation of Changes, Tests, or Experiments

Safety, Screen Out, and Equivalency Evaluations

SE 2001-0002	Safety Assessment System/ Plant Process Computer System Replacement, Revision 0.
SE 2001-0006	Modification of SPF Cooling System to Allow Alignment of Either B or A and Skid-Mounted SFP Pumps to the B SFP Heat Exchanger, Revision 0
2002-0601	Twinco Constant Voltage Transformer Removal, Revision 0.
TE-99-0072	Voltage Regulator to Replace Twinco Units, Revision 1.
TE-2001-0048 N	US Instrument Model SPS500 Single Loop Power Supply Which Replaces Foxboro Models 610 and 610A Supplies.

Section 1R04: Equipment Alignment

Action Reports

2003-0216	Battery Bank Ground Battery A
2003-0383	Battery Bank Ground on A Battery
2003-1306	DC Ground on 200 Amp Battery Charger A
2003-1878	DC Ground on the “B” Battery

Tests

PT-10.2	Station Battery B Service Test
PT-10.3	Station Battery A Service Test

Section 1R12: Maintenance Rule Implementation

Action Reports

Action Reports 2001-1500, 2001-2003, 2002-1483, 2003-0283, and 2003-1494

Procedures and Documents Reviewed

Maintenance Rule Performance Criteria for primary containment isolation system (MR Train CTS02), function 21A
System Engineer Action Reports for AOV-966B/C
IST Memorandum on 966C Operability, ISTM-145, dated August 12, 2003
EP-2-P-0168, "Maintenance Rule Monitoring"
IP-IIT-3, revision 3, "Containment Leakage Rate Test Program"
Maintenance Rule (a)(1) Goal Determination for MR Train CTS02

Section 1R13: Maintenance Risk Assessments and Emergent Work Evaluation

Action Report

2003-2927 EOOS Risk Indicates Orange

Section 1R15: Operability Evaluations

Action Report

2003-2765	S/G “A” Feed Regulating Valve is Open more than in Past Cycles
2003-3112	Control Building Roof Leaks
2003-3129	Block Heater Not Working on Diesel Fire Pump

Section 1R19: Post Maintenance Testing

Work Orders

WO 2003-2487	‘B’ AFW LO Pump Modification
WO 2003-2742	Repair ‘A’ CCP Varidrive
WO 2003-0189	Test Main Steam Safety Valves

Section 1R22: Surveillance Testing

Procedures

M-1306	Ginna Station Material Condition Inspection Program
PT-16Q-A	Auxiliary Feedwater Pump A - Quarterly
RSSP-2.2	Diesel Generator Load and Safeguard Sequence Test
PT-16.3B	AFW Pump ‘B’ Discharge MOV and Check Valve Test

Section 4OA1: Performance Indicator Verification

Procedures
CHA-PERFORMANCE-IND, Primary Chemistry Performance Indicators, Rev 2
RPA-PERFORMANCE-IND, RP Performance Indicator Guideline, Revision 1
IP-LPC-8, NRC Performance Indicators, Revision 3
Offsite Dose Calculation Manual, Revision 18

Reports
Annual Radiological Effluent Release Report - 2002
Quarterly and Monthly Liquid Release and Dose Summary Reports 11/01/2002 to 12/01/2003
Quarterly and Monthly Gaseous Release and Dose Summary Reports 10/01/2002 to 12/01/2003

Section 4OA2: Identification and Resolution of Problems

Action Reports

AR 1999-0488	MQ 400D Output Voltage Exceeds Tech Spec High Limit
AR 2000-0412	Spare MQ 483 Failed in Storage
AR 2000-1695	MQ 483 Failed Causing Halon Actuation
AR 2001-2220	MQ 400E Output Voltage Below Normal
AR 2001-1943	Failure of Voltage Regulator MQ 400E
AR 2001-2232	Replacement MQ 400E Voltage Not Regulating
AR 2003-2650	Flex Conduit Pulled From RTD
AR 2003-3282	Steam Header Alarms
AR 3003-3328	Loose and Missing Facade Near 115kv Yard

AR 2003-1790, AR 2003-2081, AR 2003-2257, AR 2003-2426, AR 2003-1827, AR 2003-2106, AR 2003-2289, AR 2003-2427, AR 2003-1835, AR 2003-2146, AR 2003-2412, AR 2003-2466, AR 2003-1884, AR 2003-2160, AR 2003-2424, AR 2003-2531, AR 2003-2057, AR 2003-2181, AR 2003-2425, AR 2003-2578, AR 2003-2059

Design Analysis

DA-EE-2002-059 Design Analysis for Twinco AC Voltage Regulator Removal, Revision 1

Drawing

03201-0102 120 Vac Instrument Bus One-Line Diagram, Revision 17

Modification

PCR 2002-0050 Twinco Removal

Section 4OA3: Event Follow-up

Action Report

AR 2003-2703 TDAFW Valve Found Out of Position

Section 4OA5: Other Activities

Action Reports

2003-2170, 2003-2196, 2003-2213, 2003-2363, 2003-2541, 2003-2580, 2003-2581, 2003-2603, 2003-2605, 2003-2608, 2003-2743

Correspondence

NRC Generic Letter 98-04: "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment," dated July 14, 1998

NRC Bulletin 2003-01: Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors, dated June 9, 2003

RG&E Letter, Robert C. Mecredy to Robert L. Clark (NRR), dated August 8, 2003, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors" R. E. Ginna Nuclear Power Plant

NRC Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-Of-Coolant Accident," Revision 1

LA-UR-02-7562, "The Impact of Recovery From Debris-Induced Loss of ECCS Recirculation on PWR Core Damage Frequency," February 2003

NUREG/CR-6762, Vol. 1 LA-UR-01-4083, "GSI-191 Technical Assessment: Parametric Evaluations for Pressurized Water Reactor Recirculation Sump Performance"

NUREG/CR-6762, Vol. 2 LA-UR-01-1800, "GSI-191 Technical Assessment: Summary and Analysis of U.S. Pressurized Water Reactor Industry Survey Responses and Responses to GL 97-04"

Documents

Qualification Card CI 03-02, dated 2/27/03
Qualification Card RO 03-02, dated 3/27/03
Qualification Card SRO 03-02, dated 3/27/03
Qualification Card STA 03-02, dated 4/9/03
Lesson Attendance TAPOP0302C, "CNMT Sump Blockage, ES-1.3 Step 16, TWR 2002-1805"
Scenario Summary 03-02-03, "Small Break LOCA with High Head Recirculation"
Licensed Operator Training, Emergency Operating Procedures (LP No. REP01S, Rev. 9), "LOCA's," Cycle 03-02
Training Activity Page (TAP), "CNMT Sump Blockage: History, Recognition, and Mitigating Actions," dated 2/24/03
TWR 2002-1805 (LP No. TAPOP0302C), "CNMT Sump Blockage: History, Recognition and Mitigating Actions"

Procedures

ECA-1.1, "Loss of Emergency Coolant Recirculation," Rev. 22
ES-1.2, "Post LOCA Cooldown and Depressurization," Rev. 26
ES-1.3, "Transfer to Cold Leg Recirculation," Rev. 36
ND-TQS, "Training, Qualification, and Simulator," Rev. 14
Procedure A-3.1, "Containment Storage and Closeout Inspection," Rev. 30
Procedure A-3.1, "Containment Storage Inspection," Rev. 29
IP-HSC-3, "Housekeeping Control," Rev. 9
ND-HSC, "Housekeeping and System Cleanliness and Foreign Material Exclusion (FME)"
Procedure A-3.1, "Containment Storage and Closeout Inspection," dated 10/14/03
Procedure A-3.1, "Containment Storage Inspection," dated 11/18/97, 4/20/99, 10/17/00, 4/17/02
RF-65.1, Reactor Disassembly Operations, Rev. 14
CME-55-02 MRPI Coil, Westinghouse MRPI Coil Stacks Testing and Maintenance, Rev. 3
CME-55-02 CRD Cables, Maintenance And Testing For Control Rod Drive Cables And Assemblies, Rev. 01
GME-55-99 MISSHLDCAB, Disconnection And Reconnection Of Missile Shield Electrical Cables, Rev. 01
SM-2001-0042-1, CRDM Fan Flow Testing, Revision 0

LIST OF ACRONYMS

ADAMS	Agency-Wide Documents Access and Management System
AR	Action Report
CATS	Commitment Action Tracking System
CFR	Code of Federal Regulations
CR	Control Room
EDG	Emergency Diesel Generator
EOOS	Equipment Out of Service
EOP	Emergency Operating Procedure
EPIP	Emergency Plan Implementing Procedure
IM	Inspection Manual
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IPEEE	Individual Plant Examination of External Events
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
MDAFW	Motor Driven Auxiliary Feedwater
MR	Maintenance Rule
NCV	Non-cited Violation
NEI	Nuclear Energy Institute
NLI	Nuclear Logistics Inc
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
PCIV	Primary Containment Isolation Valve
PCR	Plant Change Record
PI	Performance Indicator
PMT	Post Maintenance Test
PPCS	Primary Plant Computer System
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
RFO	Refueling Outage
RG&E	Rochester Gas and Electric
RHR	Residual Heat Removal
RO	Reactor Operator
RWST	Refueling Water Storage Tank
SDP	Significance Determination Process
SRO	Senior Reactor Operator
SSC	Structures, Systems, Components
TDAFW	Turbine Driven Auxiliary Feedwater
TM	Temporary Modification
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
WO	Work Order
WOG	Westinghouse Owners Group