August 1, 2000

EA 00-168

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

SUBJECT: NRC's R. E. GINNA INSPECTION REPORT 05000244/2000-003

Dear Dr. Mecredy:

On July 1, 2000, the NRC completed an inspection of your R. E. Ginna facility. The enclosed report presents the results of that inspection. Preliminary findings were presented to RG&E management led by Mr. J. Widay in an exit meeting on July 6, 2000.

NRC inspectors examined numerous activities related to reactor safety and compliance with the Commission's rules and regulations, and with the conditions of your operating license. The inspection consisted of a selected examination of procedures and records, observations of activities, and interviews with personnel. It involved seven weeks of resident inspection and a region-based inspection of your conduct of changes, tests, and experiments, including permanent plant modifications.

There were two green findings identified during this inspection . One green finding involved the accuracy of calculations of emergency core cooling system leakage outside containment and the other involved a radiological release from a gas decay tank. These findings and two additional issues were determined to be violations of NRC requirements. However, these violations were not cited due to their very low safety significance and because they were entered into your corrective action program. If you contest these non-cited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the United States 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; and the NRC Resident Inspector at the Ginna facility.

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Sincerely,

Michele G. Evans, Chief
Projects Branch 1
Division of Reactor Projects

Docket No. 05000244
License No. DPR-18

Enclosure: Inspection Report 05000244/2000-003

cc w/encl:
P. Wilkens, Senior Vice President, Generation
P. Eddy, Electric Division, Department of Public Service, State of New York
C. Donaldson, Esquire, State of New York, Department of Law
N. Reynolds, Esquire
F. William Valentino, President, New York State Energy Research
and Development Authority
J. Spath, Program Director, New York State Energy Research
and Development Authority
T. Judson, Central NY Citizens Awareness Network

Distribution w/encl (**VIA E-MAIL**):
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E. Adensam, NRR (ridsnrrdlpmlpdi)
D. Holody, ORA
G. Vissing, NRR
W. Scott, NRR
J. Wilcox, NRR
R. Urban, ORA
B. Summers, OE (BTS)
H. Nieh, SRI - Ginna
M. Evans, DRP
W. Cook, DRP
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R. Junod, DRP
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REGION I

Docket No:	05000244
License No:	DPR-18
Report No:	05000244/2000-003
Licensee:	Rochester Gas and Electric Corporation (RG&E)
Facility:	R. E. Ginna Nuclear Power Plant
Location:	1503 Lake Road Ontario, New York 14519
Dates:	May 14 through July 1, 2000
Inspectors:	H. K. Nieh, Senior Resident Inspector C. R. Welch, Resident Inspector G. K. Hunegs, Senior Resident Inspector, Nine Mile Point W. A. Cook, Senior Project Engineer, Division of Reactor Projects B. S. Norris, Senior Reactor Inspector, Division of Reactor Safety R. S. Bhatia, Reactor Inspector, Division of Reactor Safety F. J. Arner, Reactor Inspector, Division of Reactor Safety
Approved by:	M. G. Evans, Chief, Projects Branch 1 Division of Reactor Projects

Summary of Findings

IR 05000244-00-03; on 05/14 - 07/01/2000; Rochester Gas and Electric Corporation; R. E. Ginna Nuclear Power Plant. Barrier Integrity, Public Radiation Safety, Other Activities.

This report covers a seven-week period of resident inspection and a region based inspection of changes, tests, experiments, and permanent modifications, conducted per the NRC's Reactor Oversight Process (Attachment 1). The inspections identified two green issues, which were non-cited violations, and two additional non-cited violations. The significance of issues is indicated by their color (green, white, yellow, red) and was determined by the Significance Determination Process (SDP).

Cornerstone: Barrier Integrity

1. GREEN. The inspectors identified that RG&E did not properly translate emergency core cooling system (ECCS) design information into a procedure used for determining ECCS leakage outside containment. The procedure did not contain instructions for increasing leakage rates measured at low system pressure to those rates expected at the higher system pressures during the recirculation phase following a loss of coolant accident. Actual measured leakage rates were determined to be far below technical specification limits. This finding is a non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control." (Section 1R22)

Cornerstone: Public Radiation Safety

2. GREEN. During a gas decay tank release evolution, operators mistakenly released the wrong tank. The resultant radioactive release was well below regulatory limits. Nevertheless, the inspectors determined this issue to be a non-cited violation of technical specification 5.4.1, "Procedures," because RG&E did not sample the noted tank within twelve hours of releasing it, as required by station radioactive effluent control program procedures. (Section 2PS1)

Cross-cutting Issues: Human Performance

3. NO COLOR. RG&E did not promptly enter problems into their corrective action program for two equipment issues. First, station personnel had noted containment tendon grease leakage since September 1999; however, the impact on plant equipment and the development of long-term corrective actions were overlooked until an Action Report was written in June 2000. This finding is a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions." In the second example, RG&E personnel did not promptly initiate an Action Report for emergency siren test failures. As a result, RG&E failed to complete an NRC notification in a timely manner. The inspectors observed three examples of inattention to detail in the conduct of day-to-day activities. In one instance, during the resolution of an inoperable containment pressure instrument, inattention to detail resulted in the preparation and acceptance of a weakly supported technical evaluation and a poorly implemented procedure

change. The other examples involved the inadvertent release of a gas decay tank and the incorrect de-energization of a running service water pump. (Section 4OA4.1)

4. NO COLOR. RG&E did not perform safety evaluations as required by 10 CFR 50.59 and Ginna's associated implementing procedure (IP-SEV-1). Four examples were identified and determined not to be unreviewed safety questions. The examples indicated improper procedure implementation during the safety review process. This issue was determined to be a non-cited violation of 10 CFR 50.59. (Section 4OA4.2)

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Report Details

SUMMARY OF PLANT STATUS

Ginna began the period at full power. RG&E took the station off line on May 20 for planned offsite electrical distribution maintenance. Full power operation resumed on May 23 until June 29, when operators reduced power to approximately 75% to repair a steam leak. The plant was returned to full power operation on June 30, where it remained through the end of the inspection period.

1. REACTOR SAFETY
Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R02 Evaluation of Changes, Tests, or Experiments

a. Inspection Scope

The inspectors reviewed a sample of safety evaluations (SEs) performed by RG&E to verify that changes related to systems, structures, or components (SSCs) and procedures, as described in the updated final safety analysis report (UFSAR), were reviewed and documented in accordance with 10 CFR 50.59. The SEs were selected from those performed during the last eighteen months, taking into consideration the risk significance of the change, and the impact on the three reactor safety cornerstones. The inspectors also reviewed a sample of safety reviews (SRs) associated with changes to SSCs and procedures for which RG&E determined that an SE was not required (see Section 4OA4.2). This review was to verify that RG&E's threshold for performing SEs was consistent with 10 CFR 50.59. The inspectors reviewed 16 SEs and 40 SRs. The specific documents reviewed are listed in Attachment 2 to this inspection report.

b. Issues and Findings

There were no findings identified.

1R04 Equipment Alignment

a. Inspection Scope

The inspectors performed partial system walkdowns of the following system trains while their redundant trains were out-of-service for maintenance.

a. A - spent fuel pool cooling
b. A - containment hydrogen monitor

Additionally, a complete walkdown was performed on accessible portions of the containment spray (CS) system. These inspections verified that key valves and electrical circuit breakers were properly aligned in accordance with plant procedures and drawings. During the complete walkdown, the inspectors discussed overall CS system condition, which was good, with the associated system engineer. The inspectors also verified the adequacy of planned corrective actions for two CS system instrument lines that RG&E had identified as inappropriately supported.

b. Issues and Findings

There were no findings identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors reviewed Action Report No. 98-0289 for the 1B 125 VDC battery charger, which documented the number of functional failures being greater than established performance criteria. This review verified that RG&E properly implemented maintenance rule requirements such as component scoping, functional failure determination, and monitoring. The inspectors also examined the first quarter 2000 systems engineering performance report, dated May 9, 2000, and quality assurance audit, AINT-2000-0005-RTD, “Maintenance Rule Audit Report,” dated May 24, 2000.

b. Issues and Findings

There were no findings identified.

1R13 Maintenance Risk Assessments

a. Inspection Scope

This inspection verified that RG&E effectively assessed plant risk before performing the following maintenance activities.

a. Seismic upgrade of B emergency diesel generator output breakers, 6/12/2000
b. Residual heat removal (RHR) system valve surveillance, 6/13/2000
c. B RHR system outage, 6/19/2000

The inspectors discussed the use of RG&E’s online risk monitoring software with scheduling department personnel. Additionally, the inspectors verified that RG&E’s risk management actions were consistent with those described in procedure IP-PSH-2, “Integrated Work Schedule Risk Management.”

b. Issues and Findings

There were no findings identified.

1R15 Operability Evaluations

a. Inspection Scope

During an A emergency diesel generator (EDG) surveillance test in January 2000, RG&E noted that fuel oil pressure (45.8 psig) was slightly outside the expected test range at full load (35 to 45 psig). RG&E determined that the A EDG was operable and that no further corrective actions were required since the observed pressure was within the acceptable range as determined by the American Society of Mechanical Engineers code (41.85 to 49.5 psig). The inspector reviewed the operability evaluation for the associated Action Report (No. 2000-0072) and verified that the operability decision was acceptable and that the documentation to support the decision was appropriate.

The inspectors also reviewed an initial operability evaluation associated with Action Report No. 2000-0678, dated 6/22/2000, which documented the identification of grease leaking from the grease fill pipes of three containment tendons. The inspectors referenced applicable portions of the UFSAR, visually inspected the condition, and discussed the issue with the associated system engineer to verify that the containment structure remained operable. The inspectors also questioned the timeliness of this operability evaluation and assessed corrective action program implementation for this issue (see Section 4OA2).

b. Issues and Findings

There were no findings identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed selected portions of the permanent plant modifications listed in Attachment 2. The modifications were selected from approved changes that had been completed within the last eighteen months. The selection was based on risk significance, impact on the reactor safety cornerstones, and a representative sample of engineering disciplines and plant activities. The modifications encompassed equivalency evaluations, setpoint changes, design calculations, and changes to normal, abnormal, and emergency procedures. The review included the design, the as-installed implementation, the post-modification testing, and the completeness of associated documentation. As needed, discussions were held with the responsible design and system engineers, and other personnel familiar with the changes. The inspectors' review included: 19 plant change records (physical modifications), 17 technical evaluations, 3 commercial grade dedication evaluations, 15 design analyses, and 24 procedure change notices. The specific documents reviewed are listed in Attachment 2 to the inspection report.

b. Issues and Findings

There were no findings identified.

1R19 Post Maintenance Testing

a. Inspection Scope

The inspectors reviewed the post maintenance tests for the following activities to verify that RG&E appropriately demonstrated the components' ability to perform its intended safety function:

a.	WO 19904380	Seismic upgrade for B emergency diesel generator output breaker to bus 16
b.	WO 19904382	Seismic upgrade for B emergency diesel generator output breaker to bus 17
c.	WO 20000073	B residual heat removal pump breaker inspection

b. Issues and Findings

There were no findings identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors witnessed the performance of portions of the following tests, and verified that selected test acceptance criteria were technically appropriate.

a.	PT-2.5	Air operated valve quarterly surveillance test - auxiliary building
b.	PTT-23.17C	Containment isolation valve leak rate testing
c.	PT-39	Leakage evaluation of primary coolant sources outside containment

b. Issues and Findings

The inspectors identified that procedure PT-39 did not provide adequate instructions to determine the expected leak rates from portions of the containment spray and safety injection systems during the recirculation phase following a postulated loss of coolant accident. Leak rates identified when portions of the systems are at low pressures are not normalized (increased) to those leak rates that would exist at the higher system pressures experienced during recirculation phase. Although this method is not conservative, RG&E has not identified any leakage that should have been normalized. Additionally, the current total leakage from these systems is very minor (approximately 0.87 gallons per hour) and there is sufficient margin to the technical specification limit of 2.75 gallons per hour.

This issue was screened using the Significance Determination Process and was considered to be of very low safety significance (Green). Nonetheless, this condition is a violation of 10 CFR 50, Appendix B, Criterion III, “Design Control,” which requires, in part, that design information be translated into procedures. This issue is being treated as a Non-Cited Violation (NCV), consistent with Section VI.A of the Enforcement Policy, issued on May 1, 2000 (65 FR 25368). This finding has been entered into RG&E’s corrective action program (Action Report No. 2000-0720). **(NCV 05000244/2000-003-01)**

Cornerstone: Emergency Preparedness [EP]

1EP1 Drill, Exercise, and Actual Events

a. Inspection Scope

On May 17, 2000, RG&E conducted an EP exercise. The inspectors reviewed the exercise scenario, applicable emergency plan implementing procedures and emergency action levels. During the exercise, the inspectors monitored event classification; offsite authority notification; dose assessment activities; and worker accountability and evacuation. Mitigation strategies and communications were also observed. The inspectors verified that EP equipment and facilities were satisfactorily maintained in the technical support center and satellite operations support center. The inspectors also observed the post-exercise critique to verify that RG&E appropriately identified EP performance issues.

b. Issues and Findings

There were no findings identified.

2. RADIATION SAFETY

Cornerstone: Public Radiation Safety

2PS1 Radioactive Liquid Release

a. Inspection Scope

The inspectors reviewed circumstances and licensee response associated with a gas decay tank release which occurred on May 7, 2000.

b. Issues and Findings

On May 7, 2000, operators had sampled and briefed the release of the C gas decay tank (GDT). However, due to ineffective communications and lack of questioning attitude, operators inadvertently released the D GDT, which contained slightly more radioactivity than the C GDT. This issue was evaluated in the Significance Determination Process (public radiation safety cornerstone) and determined to be of very low safety significance (Green) because the radioactivity release was well below regulatory limits. The inspectors determined that RG&E personnel did not meet procedure CH-RETS-GDT-REL, “Gas Decay Tank Release,” which requires that a GDT be sampled within twelve hours of its release. This issue is a violation of TS 5.4.1, “Procedures,” and is being treated as a Non-Cited Violation, consistent with Section VI.A of the Enforcement Policy, issued on May 1, 2000 (65 FR 25368). **(NCV 05000244/2000-003-02)** This issue is in the licensee’s corrective action program (Action Report No. 2000-0548).

4. OTHER ACTIVITIES [OA]

4OA4 Cross-cutting Issues

.1 Human Performance Problems

a. Inspection Scope

The inspectors reviewed selected plant issues to examine RG&E staff response and problem resolution. Procedure IP-CAP-1, “Abnormal Condition Tracking Initiation or Notification (Action) Report,” was used as a reference.

b. Issues and Findings

Two examples were noted where RG&E did not enter equipment problems into their corrective action program. As a result, RG&E did not evaluate equipment operability, corrective actions, and reporting requirements in a timely manner. Three examples of inattention to detail were also noted this inspection period.

Containment Tendon Grease Leakage

During an inspection of the reactor containment structure on June 22, 2000, RG&E personnel noted several tendon grease fill pipes that were leaking due to through-wall corrosion (see Section 1R15). Through a review of various records, such as work orders, maintenance identification tags, and work request/trouble record forms, the inspectors determined that this leakage was previously noted by plant personnel on several occasions since September 1999. However, the inspectors did not find any associated action reports (AR). This degraded condition was not entered into the corrective action program until June 23 (Action Report No. 2000-0678). As a result, an engineering technical evaluation for equipment operability and the development of long-term corrective actions were neglected for approximately ten months. At the end of the inspection period, RG&E engineering personnel were in the process of developing corrective actions. The Significance Determination Process was not used since the structural integrity of the containment building was unaffected. Nonetheless, this failure to promptly identify and correct a condition adverse to quality is a violation of 10 CFR 50, Appendix B, Criterion XVI, “Corrective Actions.” This issue is being treated as a Non-Cited Violation, consistent with Section VI.A of the Enforcement Policy, issued on May 1, 2000 (65 FR 25368) **(NCV 05000244/2000-003-03)**

Offsite Alert and Notification System Silent Test Failure

On June 26, 2000, the Wayne County emergency management staff informed RG&E that they were unable to satisfactorily complete silent siren tests from either of the county activation centers. Later that same day, RG&E repair personnel responded to troubleshoot the system. No problems were identified and follow-up testing from both activation sites was satisfactorily completed.

Although RG&E took appropriate actions to troubleshoot and re-perform the tests, they did not recognize the need to write an Action Report (AR) for the initial test failures. As a result, RG&E failed to identify a one-hour notification required by 10 CFR 50.72 (b)(1)(v). This failure to comply with 10 CFR 50.72 is considered a violation of minor significance and is not subject to formal enforcement action. On June 28, RG&E initiated AR No. 2000-0700 for the initial test failures and made a one-hour report to the NRC. RG&E also initiated AR No. 2000-0703 to evaluate the untimely communications for this issue. The inspectors did not evaluate this finding in the Significance Determination Process because it will be included in an emergency preparedness performance indicator for alert and notification system reliability.

Narrow Range Containment Pressure Indicator (PI-944) Inoperability

On May 8, 2000, control room operators declared PI-944 inoperable due to erratic indications. PI-944 is the indicator used to perform technical specification surveillance requirement 3.6.4.1, which verifies that containment internal pressure is between -2.0 and +1.0 psig. RG&E generated action report (AR) 2000-0553 to document this issue.

Control room operators began using a wide range indication on the plant process computer system (PPCS) as an alternative to PI-944. RG&E prepared an interoffice memo on May 12 that documented the basis for using the PPCS indication. The inspectors questioned the technical adequacy of this memo because it did not fully determine the acceptability of using the PPCS indication. Specifically, RG&E did not verify the accuracy and calibration of the PPCS indication and did not consider the fact that the PPCS indication could not indicate pressures less than zero psig. After further review of the PPCS indication, the inspectors concluded that its use was acceptable because the indication had been calibrated and operators had maintained containment pressure greater than zero psig.

Also, the operations department made a temporary change to the procedure used for adjusting containment pressure and incorrectly specified the use of a main control board wide range indicator rather than the intended PPCS indication. Although both indications are derived from the same transmitter, the procedure could not be performed as modified since the main control board indicator’s scale was inadequate for measuring the small pressure increments described in the procedure. Based on discussions with control room operators, the inspectors determined that the operators were using the appropriate PPCS indication, but did not pursue correction of the procedure discrepancy. This failure to properly follow procedures is a violation of minor significance and not subject to formal enforcement action. Station management acknowledged the inspectors’ observations and issued a memo to the operating crews emphasizing the need to focus on attention to detail, questioning attitudes, and procedure compliance. Operators returned PI-944 to service following repairs on May 31.

Inadvertent Gas Decay Tank Release and Service Water Pump Trip

Two additional lapses in operator attention to detail were noted during routine evolutions this inspection period. One issue occurred due to a lack of self-checking by an operator implementing a service water (SW) pump tagout. On May 19, 2000, an operator inadvertently tripped a running SW pump. This resulted in several main control board alarms and required control room operators to start another SW pump. This issue was not evaluated in the Significance Determination Process because the inadvertent pump trip did not adversely affect the SW system's ability to perform its intended safety functions. The second operator inattention to detail issue involved the inadvertent release of the D gas decay tank (reference Section 2PS1). RG&E entered both of these issues into their corrective action program (Action Report Nos. 2000-0579 and 2000-0548, respectively).

.2 Conduct of 10 CFR 50.59 Safety Reviews

a. Inspection Scope

During the inspection of plant changes, tests, and experiments (reference Section 1R02), the inspectors reviewed a sample of the safety reviews (SRs) associated with changes to systems and procedures for which RG&E determined that a safety evaluation (SE) was not required. This review was to verify that RG&E's threshold for performing SEs was consistent with 10 CFR 50.59.

b. Issues and Findings

The inspectors identified four examples where RG&E failed to perform the required SE in accordance with procedure IP-SEV-1, "Preparation, Review and Approval of Safety Reviews". Specifically, procedure IP-SEV-1 provides a screening process used to determine if an SE is required per 10 CFR 50.59. Question 4.A on the SR form asks if the change or condition would alter the design, function, or method of performing a function of a structure, system, component or procedure as described in the UFSAR. If the answer is yes, the procedure requires a written 10 CFR 50.59 SE. In each of the following instances, RG&E inappropriately answered "NO" to question 4.A.

- In December 1998 and March 1999, plant change records 98-101 and 99-020, respectively, replaced 14 inch globe valves (SW-4619 and SW-4620) with 10-inch butterfly valves. The valves are the service water (SW) outlet isolation valves for the component cooling water heat exchangers. The valves were described in the UFSAR as globe valves (Section 9.2.1.2.4), and shown as 14-inch globe valves on the UFSAR drawing (Figure 9.2-1).

- Between June 1998 and August 1999, in accordance with technical evaluation No. 94-0586, RG&E replaced the bronze impellers on the SW pumps with a stainless steel upgrade. An unexpected consequence was an increase in pump capacity, with an associated increase in electrical load on the vital bus (243 KW to 257 KW per pump). After the last impeller was changed, RG&E updated the UFSAR (Table 8.3-2) to reflect the new loading of the emergency diesel generators.

- In May 2000, containment pressure indicator PI-944 was declared inoperable due to spiking. The licensee processed a temporary procedure change notice to substitute PI-945 for PI-944. PI-944 is specifically identified in UFSAR, Section 6.2.1.5.1, as the instrument used to ensure compliance with technical specification surveillance requirement 3.6.4.1, for verifying containment pressure every 12 hours.

The above examples were determined to be of very low significance since there was no impact on the affected systems' abilities to perform their intended safety functions. Although the issues were technically evaluated by Ginna, and the UFSAR was changed as necessary, 10 CFR 50.59 and RG&E implementing procedure IP-SEV-1 required a written SE. Although the risk significance was very low, this failure to perform an SE when the design, function or method of performing a function of a system or procedure, as described in the UFSAR was affected to verify the change did not involve an unreviewed safety question, is a violation of 10 CFR 50.59. This issue is being treated as a Non-Cited Violation, consistent with Section VI.A of the Enforcement Policy, issued on May 1, 2000 (65 FR 25368). RG&E initiated Action Report No.2000-0652 to evaluate and correct this issue. **(NCV 50000244/2000-003-04)**

4OA5 Management Meetings

a. Exit Meeting Summary

On July 6, 2000, the inspectors presented their inspection results to members of RG&E management led by Mr. J. Widay . RG&E management acknowledged the findings presented. No proprietary information was identified.

b. NRC/RG&E Management Meeting

On June 8, 2000, members of NRC and RG&E management met to discuss the results of Ginna's most recent Plant Performance Review (PPR), documented via letter, dated March 31, 2000. This meeting was open to public observation.

ITEMS OPENED AND CLOSED

Opened/Closed

NCV	05000244/2000-003-01	Failure to properly translate system design information into surveillance testing procedures.
NCV	05000244/2000-003-02	Failure to properly sample a gas decay tank prior to release.
NCV	05000244/2000-003-03	Failure to promptly identify and correct leaking containment tendons grease fill piping.
NCV	05000244/2000-003-04	Failure to perform 10 CFR 50.59 safety evaluations.

LIST OF ACRONYMS USED

AR	Action Report
CFR	Code of Federal Regulations
CS	Containment Spray
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EP	Emergency Preparedness
GDT	Gas Decay Tank
KW	Kilowatt
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
PERR	Public Electronic Reading Room
PPCS	Plant Process Computer System
PPR	Plant Process Review
PSIG	Pounds per Square Inch Gauge
RG&E	Rochester Gas and Electric Corporation
RHR	Residual Heat Removal
SE	Safety Evaluation
SR	Safety Review
SSC	Systems, Structures, Components
SW	Service Water
UFSAR	Updated Final Safety Analysis Report
VDC	Volts Direct Current

ATTACHMENT 1

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.

ATTACHMENT 2

PARTIAL LIST OF DOCUMENTS REVIEWED

Plant Change Records:
PCR 94-004, Revision 0, RCP Seal Leakoff Flowmeter and Transmitter Replacement
PCR 95-015, Revision 0, Instrument Air System Upgrade
PCR 96-015, Revision 0, Vacuum Breakers at SW Pump Discharge
PCR 97-010, Revision 1, Installation of Thermal Pressure Relief Devices on Piping Lines Inside the Containment
PCR 97-026, Revision 2, Reactor Protection RTD Input Module Replacement
PCR 97-028, Revision 1, Replace Inverter A (INVTCVTA) and Inverter B (INVTCVTB)
PCR 97-065, Revision 1, Control Room Upgrade
PCR 97-067, Revision 0, Installation of Vacuum Breakers on SW Supply to SAFW
PCR 97-069, Revision 0, Setpoint Change for FIA-2033, 2034, 2035, 2036 (Service Water)
PCR 98-004, Revision 0, Replacement of Pressurizer Mirror Insulation on the Safety Valve Loop Seals
PCR 98-015, Revision 0, Diesel Generator Supply Breaker Time Delay Relays
PCR 98-040, Revision 0, Provide Isolation Devices for EDG Vault Sump Pump Level Switches
PCR 98-049, Revision 0, MOV'S 857A-C and 860A-D Motor Replacement
PCR 98-062, Revision 0, Protective Relay Setting Changes as Identified in Design Analysis DA-EE-93-107-07
PCR 98-071, Revision 0, Change Timer Circuit and Provide over Current Protection for Circuit
PCR 98-085, Revision 0, Control Room Radiation Monitor Setpoint Changes
PCR 98-101, Revision 0, Replacement of 14" Globe Valve (4619) with 10" Butterfly Valve
PCR 99-020, Revision 0, Replacement of 14" Globe Valve (4620) with 10" Butterfly Valve
PCR 99-057, Revision 0, MCCD Overload Heater Replacement per DA-EE-96-005-07

Technical Evaluations:
TE 93-0555, Revision 3, Foxboro H-Line Controllers LC-428F & PC-431K Replaced with Westinghouse PID 9000 or Scientec/NUS PIDA 700 Modules
TE 94-0586, Revision 3, Material Change for Service Water Pump Impellers and Wear Rings
TE 98-0200, Revision 1, CCW HX Re-Tubing
TE 99-0006, Revision 0, GE Series CR120B Replacement for Containment Isolation Gould J13 Series
TE 99-0007, Revision 0, Agastat Relay Timer Model E7012PA002 Versus New Model E7012PA004
TE 99-0011, Revision 0, Evaluate Spare RHR Pump Rotating Assembly for Use at Ginna
TE 99-0015, Revision 1, Replacement of Potter and Brumfield KH 4911-4 Relays with Potter and Brumfield KH 5678 for Foxboro 62 and 67 Series Controllers
TE 99-0016, Revision 0, Equivalency Evaluation-EDG Lube Oil Check Valve Change from Parker Hanningfin (5 psig cracking pressure) to NUPRO (15 psig cracking pressure)
TE 99-0017, Revision 0, Installation of Bushings in B CCW Pump Bearing Housings
TE 99-0020, Revision 0, EDG Starting Air Pressure Control Valves Material ID Change
TE 99-0021, Revision 0, D CNMT Recirculation Fan Discharge Valve Pin
TE 99-0023, Revision 0, Replace Crosby JMB-C Relief Valves with SS Crosby OMNI Series 900 Relief Valves
TE 99-0043, Revision 0, Moeller Temperature Switch Replacement with Ashcroft Duratemp

TE 99-0061, Revision 0, Replace ASCO Solenoids LB8300D64RU with ASCO Recommended Solenoid 2126321RU
TE 2000-0002, Revision 0, Replace CVCS Hold-Up Tank Fisher Control Valve
TE 2000-0004, Revision 0, Repair/Replace Original Reactor Makeup Water Tank Liner (Plasite-7155 with Vendor Approved Replacement Liner (Plasite-7156)
TE 2000-0007, Revision 0, ASCO LB8300B64RU, and LB8300B64RU, Equivalency, to be Replaced by an ASCO NPEF8300381ERU

<u>Commercial Grade Dedication Evaluations:</u>
CG IEE 90-044, Revision 3, Armstrong ½" & 1" - 300 psi FXSC Strainer with 1/16" Perforated Screen: Service Water Strainers for Auxiliary Feedwater Pumps and Lube Oil Coolers
CG IEE 92-057, Revision 2, Bearing For Worthington Pump Model
CG IEE 92-012, Revision 1, AMOT Temperature Elements

<u>Design Analysis:</u>
DA CE-99-041, Revision 0, Seismic Analysis of Block Wall near PT-468 (Steam Generator 1A Pressure Transmitter)
DA EE-92-131-06, Revision 13, AC Motor Operated Valve Degraded Voltages
DA EE-92-098-01, Revision 3, Diesel Generator A Steady State Loading Analysis
DA EE-93-098-06, Revision 1, DC Motor Operated Valve Degraded Voltages
DA EE-93-104-07, Revision 2, 480V Volt Coordination and Circuit Protection Study
DA EE-93-107-07, Revision 3, 4160 Volt Overcurrent Relays Coordination & Circuit Protection Study
DA EE-94-122-16, Revision 2, Environmental Qualification of Auxiliary Building Sump Pump Motors
DA EE-99-013, Revision 2, DC Class 1E System Fault Analysis
DA EE-99-017, Revision 0, Time Delay Relay Setpoints Diesel Generator Supply Breaker
DA EE-99-031-16, Revision 0, Qualified Life Determination for Valcor Reactor Head Vent Valve External EPR O-Rings
DA EE-99-099, Revision 0, Closure of EOP Related PCAQs
DA ME-98-139, Revision 1, Emergency Diesel Generator Lube Oil and Jacket Water Heat Exchanger Service Water Differential Pressure Limits
DA ME-98-161, Revision 0, Time to Sump Switchover & Containment Pressure for LOCA Containment Integrity Analysis
DA ME-99-060, Revision 0, Revised EDG Pressure Drop Limits for Three SW Pump Operation
DA ME-2000-019, Revision 0, Evaluation of Large HVAC Damper Isolation Valves (AOV)

<u>Procedure Changes:</u>
PCN 98-3209 to O-15.4, Revision 9, Draining of Refueling Canal
PCN 98-3174 to O-2.1, Revision 90, Normal Shutdown to Hot Shutdown
PCN 98-3425 to P-4, Revision 17, Precautions, Limitations and Setpoints - Auxiliary Coolant
PCN 99-2522 to PT-10.4, Revision 14, Battery A Performance Test
PCN 99-3208 to O-6.11, Revision 86, Surveillance Requirements / Routine Operations Check Sheet
PCN 99-3209 to O-6.13, Revision 105, Daily Surveillance Log
PCN 99-3211 to O-1.2, Revision 146, Plant Startup from Hot Shutdown to Full Load
PCN 99-3225 to O-15.4, Revision 11, Draining of Refueling Canal
PCN 99-3585 to S-2.1, Revision 27, Reactor Coolant Pump Operation
PCN 99-4025 to E-1, Revision 16, Loss of Reactor or Secondary Coolant

PCN 99-4083 to FR-H.5, Revision 4, Response to Steam Generator Low Level
PCN 99-4145 to AP-Elec.1, Revision 16, Loss of 12A and/or 12B Busses
PCN 99-4287 to PT-12.1, Revision 95, Emergency Diesel Generator A
PCN 99-4401 to RSSP-2.1, Revision 55, Safety Injection Functional Test
PCN 99-4407 to PT-2.7.1, Revision 38, Service Water Pumps
PCN 99-4589 to AP-CVCS.3, Revision 0, Loss Of All Charging Flow
PCN 2000-3153 to O-6.13, Revision 108, Daily Surveillance Log
PCN 2000-3525 to S-12.2, Revision 36, Operator Action in the Event of Indication of Significant Increase in Leakage
PCN 2000-4138 to PT-12.1, Revision 97, Emergency Diesel Generator A
PCN 2000-4127 to AP-FW.1, Revision 11, Partial or Complete Loss of Main Feedwater
PCN 2000-4225 to E-3, Revision 25, Steam Generator Tube Rupture
PCN 2000-4231 to PT-2.7.1, Revision 42, Service Water Pumps
PCN 2000-4342 to PT-32B, Revision 24, Reactor Trip Breaker Testing - B Train
TPCN 2000-T-0103 to O-11, Revision 18, Control of Mini-Purge Exhaust Valves

<u>10 CFR 50.59 Safety Evaluations:</u>
SEV 1090, Revision 2, Technical Specification Bases Change for Screenhouse Bay Lower Temperature Limit
SEV 1094, Revision 1, RPS RTD Input Module Replacement
SEV 1120, Revision 0, Removal of Dewpoint Measuring Instrumentation from the Seismic and Meteorological Instrumentation System (for PCR 98-088)
SEV 1121, Revision 0, PCN 98-4517: Change to EOP Attachment 2.1
SEV 1123, Revision 0, Spent Fuel Pool Leakage Release Path Assessment
SEV 1124, Revision 0, Valve Stem Packing Improvement Program Changes to Design Criteria EWR-4859
SEV 1125, Revision 1, Station Battery Replacement
SEV 1126, Revision 0, Revise Restart/Stopping Criteria for Containment Spray Pump During the Sump Recirculation Phase
SEV 1127, Revision 0, Diesel Generator Supply Breaker Time Delay Relays (for PCR 98-015 and TSR 97-066)
SEV 1128, Revision 0, Service Air System Upgrade, Phase B
SEV 1131, Revision 01, Cycle 28 Reload
SEV 1137, Revision 0, Conversion of Part of the New Fuel Storage Area to a Contaminated Work Area
SEV 1138, Revision 0, Turbine Stop and Control Valve Test Frequency Change
SEV 1139, Revision 0, Defeat of the High Flux at Shutdown Containment Audible Alarm
SEV 1140, Revision 0, Fuel Transfer System Modification
SEV 1141, Revision 0, Removal of Reactor Water Makeup Water Tank Diaflote Diaphragm (Bladder)

10 CFR 50.59 Safety Reviews (i.e., screenings):

PCR 94-004	PCR 59-015	PCR 96-0015	PCR 97-028
PCR 97-065	PCR 97-067	PCR 97-069	PCR 98-004
PCR 98-040	PCR 98-049	PCR 98-062	PCR 98-071
PCR 98-101	PCR 99-020	PCR 99-057	PCN 98-3425
PCN 99-4025	PCN 99-4083	PCN 99-4145	PCN 99-4407
PCN 2000-3153	PCN 2000-4127	PCN 2000-4138	PCN 2000-4225
PCN 2000-4231	TPCN 2000-T-0103	CGIEE 90-0044	TE 93-0555
TE 94-0586	TE 98-0200	TE 99-0006	TE 99-0017
TE 99-0020	TE 99-0021	TE 99-0023	TE 99-0061
TE 2000-0002	TE 2000-0004	TE 2000-0007	AR 99-0770

Action Requests:

AR 99-0747, Rebar Was Cut to Install Support under WO19900814 Without the Required MDCN

AR 99-0770, Elimination of Valve 4737B Without PCR

AR 99-0880, Plant Procedures and Drawings Not Updated after Plant Modification

AR 99-0895, Modification Work Orders Have a Higher-Than-Average Identification of Deficiencies/Concerns During QC Review

AR 99-1000, Potentially Inadequate 10 CFR 50.59 for Changes to PT-2.10.15

AR 99-1560, Correct Inconsistency Between RCT-1 and PPCS Setpoint Values

AR 2000-0220, Design Pressures and Temperatures Used in DA-ME-99-007 Is Different Than Pressure and Temperature Specified in PCRs 98-039, 98-039, & 97-085

Procedures

EP-3-P-0121, Revision 3, Design Criteria

EP-3-P-0122, Revision 5, Design Analysis

EP-3-P-0126, Revision 8, Equivalency Evaluation

EP-3-P-0140, Revision 2, Modification Design Changes

EP-3-P-0154, Revision 3, Review and Approval of Vendor Design Analyses

EP-3-S-0125, Revision 2, Design Verification and Technical Review

IP-DES-2, Revision 9, Plant Change Process

IP-SEV-1, Revision 4, Preparation, Review and Approval of Safety Reviews

IP-SEV-2, Revision 5, Preparation, Review and Approval of 10 CFR 50.59 Safety Evaluations