May 24, 1993

Docket No. 50-244

DISTRIBUTION:

Docket File	AJohnson
NRC & Local PDRs	OGC
PDI-3 Reading	ACRS(10)
SVarga	JLinville,RI
JCalvo	CMcCracken
WButler	JTatum
TClark	WLazarus,RI

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

Dear Dr. Mecredy:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RE: R. E. GINNA NUCLEAR POWER PLANT SERVICE WATER SYSTEM - ROCHESTER GAS AND ELECTRIC CORPORATION'S RESPONSES TO DEFICIENCIES IDENTIFIED IN NRC REPORT 50-244/91-201, JANUARY 30, 1992 (TAC NO. M84947)

Rochester Gas and Electric Corporation (RG&E) has responded to the concerns expressed in the subject NRC inspection report by letters dated January 31, April 6, April 9, May 4, July 2, September 1, and September 30, 1992. RG&E also submitted information pertaining to heat exchanger performance testing in a letter dated June 1, 1992. We have completed our review of the responses cited above and find that additional information is needed to complete our review. While the staff agrees with your position pertaining to the check valve single failure issue and licensing basis, the staff is asking for additional information to better understand the bases for your determination that these issues are not safety significant.

Please respond to the enclosed request for additional information within 60 days of receipt of this letter.

This requirement affects fewer than ten respondents and, therefore, is not subject to Office of Management and Budget review under P.L. 96-511.

Sincerely,
/S/
Allen R. Johnson, Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
Request for Additional Information

cc w/enclosure:
See next page

OFFICE	LA:PDI-3	PM:PDI-3	D:PDI-3		
NAME	TClark	AJohnson:mw	WButler		
DATE	5/24/93	5/24/93	5/24/93	/ /	/ /

OFFICIAL RECORD COPY
FILENAME: A:\GIM84947.RAI

270113

9306030377 930524
PDR ADOCK 05000244
Q PDR

NRC FILE CENTER COPY

DF01
1/1

cc:

Thomas A. Moslak, Senior Resident Inspector
R.E. Ginna Plant
U.S. Nuclear Regulatory Commission
1503 Lake Road
Ontario, New York 14519

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406

Ms. Donna Ross
Division of Policy Analysis & Planning
New York State Energy Office
Agency Building 2
Empire State Plaza
Albany, New York 12223

Charlie Donaldson, Esq.
Assistant Attorney General
New York Department of Law
120 Broadway
New York, New York 10271

Nicholas S. Reynolds
Winston & Strawn
1400 L St. N.W.
Washington, DC 20005-3502

Ms. Thelma Wideman
Director, Wayne County Emergency
Management Office
Wayne County Emergency Operations Center
7370 Route 31
Lyons, New York 14489

Ms. Mary Louise Meisenzahl
Administrator, Monroe County
Office of Emergency Preparedness
111 West Fall Road, Room 11
Rochester, New York 14620

Enclosure

REQUEST FOR ADDITIONAL INFORMATION RE:
GINNA SERVICE WATER SYSTEM ISSUES

The inspection report for the Ginna service water system operational performance inspection (SWSOPI), NRC Report 50-244/91-201, dated January 30, 1992, raised a number of issues that were identified for further staff review. One issue of particular concern is related to the staff's review of the Ginna service water system discussed in the safety evaluation for Systematic Evaluation Program (SEP) Topic IX-3, "Station Service and Cooling Water Systems," dated November 3, 1981. During that review, it was the staff's understanding that the Ginna service water system (SWS) was designed and configured such that: a) non-safety-related loads would be automatically isolated upon initiation of a reactor safeguards actuation signal, b) potential single failure problems were overcome since each safety-related load had a redundant counterpart cooled by the other SWS header, and c) the service water supply loop is normally split into two separate headers with no sizeable cross connections. The staff's understanding with regard to these features was based on the system description that was contained in the Final Safety Analysis Report (FSAR) for the facility which was in error.

As a result of the SWSOPI, the staff learned that the Ginna SWS was in fact normally operated in a cross-connected manner and non-safety-related loads were not automatically isolated upon initiation of a reactor safeguards actuation signal.

Other issues related to the SWS design basis that were identified for staff review during the SWSOPI included: (1) application of the hydraulic model to the SWS; (2) assessment and resolution of the SWS pump discharge check valve single failure vulnerability; (3) basis/criteria for the SWS low pressure setpoint; (4) identification and resolution of potential single failure vulnerabilities that may exist due to the cross connected configuration of the SWS loop; (5) assessment and resolution of preoperational test discrepancies; and (6) assessment/validation of the minimum number of SWS pumps required to be operable to mitigate the consequences of an accident.

Rochester Gas and Electric Corporation (the licensee) has responded to the concerns expressed in the NRC inspection report by letters dated January 31, April 6, April 9, May 4, July 2, September 1, and September 30, 1992. The licensee also submitted information pertaining to heat exchanger performance testing in a letter dated June 1, 1992, which is also relevant. We have completed our review of the above cited submittals and find that some additional information is needed to facilitate completion of the staff's review of the Ginna SWS. Also, while the staff agrees with the licensee's position pertaining to the check valve single failure and licensing basis issue, additional information has been requested to better understand the bases for the licensee's determination that these issues are not safety significant.

The licensee is requested to provide complete responses to each of the staff's questions, including assumptions and justifications, in order to expedite review of the Ginna SWSOPI issues.

1. In order for the staff to judge whether one service water pump can provide adequate flow for the system configuration, the specific SWS flow requirements for each safety-related component must be identified. In making this determination, the worst-case conditions must be assumed for each component (not necessarily the design basis accident conditions). Describe specifically how the flow requirements were determined for each component, including assumptions that were made. Also, indicate to what extent vendor concurrences have been obtained for judgements and evaluations pertaining to equipment performance capabilities. Also, given the flow requirements established for the Component Cooling Water (CCW) heat exchangers, discuss any changes in the ability and times required to shutdown and cooldown the plant.

2. Based on recent test results using the most limiting service water pump (corrected for worst-case lake level, water temperature, instrument inaccuracies, and pump degradation allowed under the inservice testing program), annotate on a simplified diagram of the SWS the pump discharge pressure, flow rates through each safety-related component, flow rates through non-safety-related supply lines (as applicable), and the system configuration/alignment that is necessary to satisfy the flow requirements for safety-related equipment. Also, indicate the electrical division (A, B, or non-safety-related) where each component receives power. Describe how this system configuration/alignment will be achieved during an event assuming off-site power is available and also for the condition where off-site power is not available, and identify specific time constraints that must be met in order to satisfy all accident analyses assumptions for heat removal. Provide conservative estimates of how long it will take to complete necessary actions for each case (i.e., off-site power available and off-site power not available). The basis for these estimations should be provided, which may include reference to previous plant experience, plant simulations and exercise walk-throughs.

3. Starting with event initiation, provide bounding assessments (i.e., normal system alignment supplying service water to safety-related components, including the CCW heat exchangers and the spent fuel pool (SFP) heat exchangers, and to non-safety-related components with off-site power available and also for the condition where off-site power is not available) of SWS flow rates over time for single pump operation. The basis for this assessment should be provided, which may include reference to previous plant experience, plant simulations, and exercise walk-throughs. Provide confirmation and vendor concurrence (as appropriate) that SWS pump performance will not be jeopardized by these postulated SWS flow conditions.

4. Describe periodic surveillances, tests, inspections, and training that will be performed to ensure that the SWS will be capable of satisfying its function during an event and to ensure that all assumptions are valid.

5. Confirm that all existing accident analyses, including safe shutdown/Appendix R analyses, have been updated pursuant to 10 CFR 50.59 requirements to account for single service water pump operation, assuming worst-case conditions of lake level, water temperature, instrument inaccuracies and pump degradation. Similarly, confirm that the current submittal on boron dilution requirements only credits a single service water pump (assuming worst-case conditions) in its supporting accident analyses. Also, discuss specifically the effect of having only one service water pump operable has on the ability to mitigate a steam generator tube rupture event.

6. Provide a detailed description of the assumptions and inputs used in the previous/original limiting containment analysis versus assumptions and inputs used in the current (single service water pump) containment analysis. Provide detailed explanations for any differences of inputs and assumptions between the two analyses. Describe what the service water flow rates are for all components (safety-related and non-safety-related) over time and describe in detail how the containment heat removal rates were determined (including justification for all assumptions) for each of these analyses.

7. Confirm that for a steam line break event at hot zero power with off-site power available, that one service water pump is sufficient to mitigate the event. Similar to item 6 (above), provide a detailed explanation (justification) of any differences in assumptions and inputs used in the previous/original limiting analysis versus assumptions and inputs used in the current (single service water pump) analysis. Also, describe what the service water flow rates are for all components (safety-related and non-safety-related) over time and describe in detail how the containment heat removal rates were determined (including justification for all assumptions) for each of these analyses.

8. Describe actions that have been taken to validate the SWS flow model for various configurations of split and cross-connected system operation that may be encountered and, in particular, discuss validation of the flow model with respect to single pump operation for various system operating configurations.

9. Provide the results of the check valve single failure analysis using the validated flow model, including SWS header pressure and SWS flow rates through the failed check valve and all other components assuming that the two most limiting service water pumps are operating. Justify all assumptions. Also, describe the results of the Ginna probabilistic risk assessment (PRA) with regard to failure of a SWS pump discharge check valve, including justification for check valve failure frequencies that were assumed.

10. Describe periodic maintenance, surveillance, test and inspection activities that are performed that provide assurance that a SWS pump discharge check valve will function properly. Also, describe how the failure of a pump discharge check valve to function will be identified and addressed by the operators, including a description of training that has been provided and periodic training that will be provided in the future. Provide confirmation that operator action is sufficient to ensure that a SWS pump discharge check valve failure will remain bounded by the accident analyses that have been performed.

11. Provide the conclusions and recommendations of the revised single failure analysis. Discuss corrective actions that are being taken to address vulnerabilities. Also, confirm that the revised single failure analysis encompasses all operating configurations of the SWS that are allowed and include a description of the specific configurations that were considered to be applicable in this regard.

12. Identify which safety-related components do not need service water cooling during the injection phase of an accident and provide justification for this position, including vendor concurrence. The justification should include consideration for small break loss of coolant accident (SBLOCA) conditions, where pumps may be required to operate at or near shut-off head conditions. Provide a conservative estimate for when service water cooling must be restored to this equipment and describe how this will be accomplished. The basis for these estimations and actions should be provided, which may include reference to test data, previous plant experience, plant simulations and exercise walk-throughs.

13. Similar to item 12 (above), provide justification and vendor concurrence for the position that component cooling is not required during the injection phase of an accident for the RHR pump mechanical seal coolers and bearing water jackets, the safety injection pump mechanical seal coolers, and the containment spray pump mechanical seal coolers.

14. Describe how long the emergency diesel generator (EDG) coolers have been in service and what periodic inspections and preventive maintenance activities are typically performed relative to these coolers, including frequencies.

15. Assuming the maximum decay heat load in the spent fuel pool, describe how long service water to the SFP heat exchanger can be isolated, including supporting justification. Also, describe alternative measures that can be taken that are included in emergency procedures for cooling the spent fuel pool during an event.

16. Define the SWS header low pressure setpoints for all system configurations and discuss the basis for the setpoints that have been established.

17. Confirm that, for various modes of SWS operation, with valves 4760 and 4669 in the open position (diesel generator service water cross-connect valves), including both split and cross-connected header configuration, that a passive failure will not jeopardize operability of both diesel

generators. Describe the specific SWS configurations that were considered in this regard.

18. Given the system constraints and limitations that must be satisfied during an event with only one service water pump available, identify any changes that should be made to the existing Technical Specification LCOs and surveillance requirements. For example, the current Technical Specifications only require one loop header to be operable and does not specifically require that a service water pump be operable in each loop header, which is not consistent with the FSAR description that credits both loop headers as being available for redundancy of cooling capability. Also, requirements for split vs. cross-connected loop header operation are not stipulated.

19. Supplemental response to GL 89-13 dated June 1, 1992, only lists one SFP heat exchanger as "critical." This is not consistent with the FSAR description of critical heat loads which lists two heat exchangers. Explain.

20. Identify specifically what service water flow rates are required to be established through each heat exchanger being tested during heat exchanger performance testing that is periodically performed. Also, state the worst-case fouling factors that are assumed for each of these heat exchangers and the bases for the values being credited.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 24, 1993

Docket No. 50-244

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

Dear Dr. Mecredy:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RE: R. E. GINNA NUCLEAR POWER PLANT SERVICE WATER SYSTEM - ROCHESTER GAS AND ELECTRIC CORPORATION'S RESPONSES TO DEFICIENCIES IDENTIFIED IN NRC REPORT 50-244/91-201, JANUARY 30, 1992 (TAC NO. M84947)

Rochester Gas and Electric Corporation (RG&E) has responded to the concerns expressed in the subject NRC inspection report by letters dated January 31, April 6, April 9, May 4, July 2, September 1, and September 30, 1992. RG&E also submitted information pertaining to heat exchanger performance testing in a letter dated June 1, 1992. We have completed our review of the responses cited above and find that additional information is needed to complete our review. While the staff agrees with your position pertaining to the check valve single failure issue and licensing basis, the staff is asking for additional information to better understand the bases for your determination that these issues are not safety significant.

Please respond to the enclosed request for additional information within 60 days of receipt of this letter.

This requirement affects fewer than ten respondents and, therefore, is not subject to Office of Management and Budget review under P.L. 96-511.

Sincerely,

Allen R. Johnson

Allen R. Johnson, Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
Request for Additional
Information

cc w/enclosure:
See next page