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

November 30, 1994

Dr. Robert C. Mecredy
Vice President, Nuclear Production
Rochester Gas and Electric Corporation
89 East Avenue
Rochester, NY 14649

SUBJECT: R. E. GINNA NUCLEAR POWER STATION - REQUEST FOR ADDITIONAL INFORMATION DEALING WITH REGULATORY GUIDE 1.97 PROVISIONS FOR NUCLEAR INSTRUMENTATION (TAC NO. M90036)

Dear Dr. Mecredy:

The NRC issued a safety evaluation (SE) with an attached Technical Evaluation Report (TER) on December 4, 1990, and issued a supplemental safety evaluation (SSE) with a TER on February 24, 1993. The SE/TER and the SSE/TER found you conform to Regulatory Guide (RG) 1.97 guidance, or have provided an acceptable justification for deviations from RG 1.97 guidance, except for instrumentation associated with post-accident neutron flux monitoring.

The NRC staff and consultant, Brookhaven National Laboratory, have completed a preliminary review of the information in Attachment 2 of Rochester Gas and Electric's (RG&E) submittal, dated May 16, 1991, dealing with the neutron flux monitoring instrumentation needed to meet conditions discussed in RG 1.97. Although you have submitted information previously regarding conformance to RG 1.97, additional information is needed to complete the review.

RG&E submitted information regarding conformance to RG 1.97 by letters dated May 6 and 16, 1991, June 17, 1991, March 13, 1992, May 8, 1992, and October 14, 1992. Additional information was also submitted in an NRC meeting with you on September 16, 1992, at NRC Headquarters. Commitments and conclusions were published in an NRC Meeting Summary dated November 24, 1992, however further information is required.

The NRC staff has prepared a request for additional information (Enclosure 1). Please provide a response within 60 days of receipt of this letter.

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November 30, 1994

This requirement affects fewer than 10 respondents and, therefore, is not subject of Office of Management and Budget review under P.L. 96-511.

Sincerely,

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

Docket No. 50-244

Enclosures: 1. Request for Additional Information
2. List of References

cc w/encls: See next page

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REQUEST FOR ADDITIONAL INFORMATION REQUIRED FOR THE REVIEW OF THE

MAY 16, 1991, SUBMITTAL CONCERNING THE ADEQUACY OF EXISTING NEUTRON FLUX

ROCHESTER GAS AND ELECTRIC CORPORATION

R.E. GINNA NUCLEAR POWER STATION

INSTRUMENTATION FOR THE REQUIREMENTS OF NUREG-0737, SUPPLEMENT 1

DOCKET NO. 50-244

1. A major difference between the existing neutron flux measurement and the proposed temperature measurements is the additional delay introduced, during a reactivity insertion accident, by the time required for the temperature measurement to detect sensible heat. For slow reactivity insertion rates this delay can become substantial.

   In order to evaluate this effect, provide a quantitative estimate of the time delay in identifying a situation in which the reactivity is increasing (assuming, e.g., a constant reactivity insertion rate) for a complete range of reactivity insertions. Provide a detailed evaluation of the effect of this delay on plant safety analyses, accident consequences and required operator action, relative to the case where the increasing neutron flux is detected by the flux instrumentation at ~ 10^{-5}% power.Any comparison to the excore neutron flux instrumentation should only be made for the condition where the water level is measured to be above the hot leg, and the neutron flux provides a proportional indication of the core neutron flux.

2. Describe any unique plant-specific design features or operating conditions that support the use of temperature measurements for criticality, rather than the existing neutron flux instrumentation.

3. Since the temperature measurements only determine that a critical state exists and sufficient power is being generated to be measured on the temperature instrumentation, describe how the proposed temperature measurements will determine the subcritical states of the core as suggested in Section-3.

4. Regulatory Guide 1.97, Rev. 3 recommends measurements that: a) provide a direct measurement of the desired variable (flux in the case of criticality) and b) minimize the development of conditions which could cause the measurements to give anomalous readings that would be potentially confusing to the operator. NRC staff recognizes your discussion, in previous RG&E submittals, of Emergency Operating Procedure (EOP) instructions involving use of core exit thermocouples; however, additional information is required. Please discuss in detail the ability of the core exit thermocouples and the hot and cold leg temperature measurements to provide an accurate indication of criticality in the presence of large uncontrolled and potentially unknown variations in the core flow and heat removal rate during accident conditions.

Enclosure 1

5. In certain situations, the critical boron versus fuel burnup curve is used to determine if the coolant boron concentration is adequate to insure subcriticality during accident conditions. The NRC staff is aware of the information RG&E submitted previously concerning design basis accident (DBA) range requirements. In addition to this information, how does the critical boron versus fuel burnup curve account for the range of beyond DBA core conditions?

6. In previous correspondence with the NRC, RG&E indicated the qualified temperature limits of the plant core exit thermocouples to range from 0 to 2300 °F, the hot leg temperature measurements to range from 0 to 700 °F, and the cold leg temperatures to range from 0 to 700 °F. Please confirm these temperature measurement ranges and explain how criticality will be determined when the plant is outside these limits?

7. NRC staff acknowledges RG&E submittals with information concerning EOP instructions involving use of core exit thermocouples. Additionally, under what specific conditions will the neutron flux instrumentation and the (core exit thermocouple and hot and cold leg) temperature measurements be used to determine criticality? If the neutron flux instrumentation will not be used during conditions of a hostile environment, how will these conditions be identified? How will it be assured that the Category 3 neutron flux instrumentation is not used under conditions for which the instrumentation system is not qualified?

8. Have any special interpretations been made in the application of the Westinghouse Owners Group Emergency Response Guidelines to accommodate the use of the temperature measurements for the subcriticality function?

9. The Chapter 3 evaluation of the beyond DBAs considered the loss of reactor coolant, loss of secondary coolant and steam generator tube rupture events. How are the other beyond DBAs included in the safety evaluation?

10. Discuss how the proposed core exit thermocouple and the hot and cold leg temperature measurements satisfy the very strong recommendation of ANSI/ANS-4.5 that: a) the criticality measurements should be made with a flux detector which spans the range from 1×10^{-8} to 1×10^{-3} of full power or an equivalent or better alternative and b) to the extent possible, the selected measured variables shall be those that most directly monitor subcriticality.

    Any comparison to the excore neutron flux instrumentation should only be made for the condition when the water level is measured to be above the hot leg, and the neutron flux provides a proportional indication of the core neutron flux.

11. Describe the method used to determine the specific threshold values for the (core exit thermocouple and hot and cold leg) temperature measurements and the boron concentration that are used to protect from the effects of reactivity insertion events.

12. In the analysis of beyond design basis events, how are events other than loss-of-coolant accident secondary break and steam generator tube rupture accounted for?

REFERENCES

(1) NRC Letter (Enclosed SSE/Attached TER), "Emergency Response Capability - Conformance to Regulatory Guide 1.97, Revision 3, February 24, 1993 (TAC No. M80439)."

(2) NRC Meeting Summary, "Summary of Meeting with Rochester Gas and Electric Corporation on Emergency Response Capability of September 16, 1992," November 24, 1992.

(3) RG&E Letter, "Emergency Response Capability/NUREG 0737, Supplement 1," October 14, 1992.

(4) NRC Letter, "Emergency Response Capability - Request for Additional Information (TAC No. M80439)," July 7, 1992.

(5) RG&E Letter, "NUREG-0737 Supplement 1/Regulatory Guide 1.97," May 8, 1992.

(6) RG&E Letter, "NUREG-0737 Supplement 1/Regulatory Guide 1.97: Comparison of Ginna Post Accident Instrumentation," March 13, 1992.

(7) RG&E Letter, "Environmental Qualification of Containment Air Temperature RTD's," June 17, 1991.

(8) RG&E Letter, "NUREG-0737, Supplement 1/Regulatory Guide 1.97 Clarifications," May 16, 1991.

(9) RG&E Letter, "NUREG-0737, Supplement 1/Regulatory Guide 1.97," May 6, 1991.

(10) NRC Letter, "Ginna Station's Conformance to Regulatory Guide 1.97, Revision 3," March 22, 1991.

(11) NRC Letter (Enclosed SE/Attached TER), "Emergency Response Capability - Conformance to Regulatory Guide 1.97, Revision 3, December 4, 1990 (TAC No. 51093)."

(12) RG&E Letter, "Regulatory Guide 1.97 Conformance - Emergency Response Capability," July 13, 1990.

(13) NRC Letter, "Emergency Response Capability - Conformance to Regulatory Guide 1.97, Revision 3," February 20, 1990.

(14) RG&E Letter, "Regulatory Guide 1.97 Review," June 16, 1986.

(15) NRC Letter, RG&E, "Regulatory Guide 1.97, Emergency Response Capability," April 14, 1986.

(16) RG&E Letter, "USNRC Regulatory Guide 1.97," February 28, 1985.

(17) RG&E Letter, "NUREG-0737, Supplement 1," January 31, 1984.

(18) NRC Letter, Generic Letter 82-33, "Supplement No. 1 to NUREG-0737-Requirements for Emergency Response Capability," December 17, 1982.

cc:

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