RGE
Always at Your Service

Robert C. Mecredy
Vice President
Nuclear Operations

February 24, 2004

Mr. Robert L. Clark
Office of Nuclear Regulatory Regulation
U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Subject: Supplementary Information Associated with the
Revision to Safety Limits and Instrumentation Setpoints
Rochester Gas and Electric Corporation
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Reference: Letter from Robert C. Mecredy (RG&E) to Guy S. Vissing (NRC), "Application for Amendment to Facility Operating License Revision to Safety Limits and Instrumentation Setpoints", dated April 9, 2002.

Dear Mr. Clark:

In the above Reference, RG&E submitted a proposed change to the Improved Technical Specifications associated with the Safety Limits and Instrumentation Setpoints requirements. Subsequent to the submittal, as the result of recent discussions with the NRC staff, RG&E would like to provide the attached additional information associated with the instrument setpoints.

I declare under penalty of perjury under the laws of the United States of America that I am authorized by RG&E to make this submittal and that the foregoing is true and correct.

Any questions concerning this submittal should be directed to Thomas Harding at (585) 771-3384.

Very truly yours,

Executed on February 24, 2004.

Robert C. Mecredy

Attachment - Instrument Setpoint Supplemental Information

A001

An equal opportunity employer

89 East Avenue | Rochester, NY 14649
tel (585) 546-2700
www.rge.com

1000951

An Energy East Company

xc: Mr. Robert Clark (Mail Stop O-8-C2)
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852

Regional Administrator, Region 1
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

U.S. NRC Ginna Senior Resident Inspector

Peter R. Smith
New York State Energy, Research, and Development Authority
17 Columbia Circle
Albany, NY 12203-6399

Mr. Paul Eddy
NYS Department of Public Service
3 Empire Plaza, 10th Floor
Albany, NY 12223-1350

R.E. Ginna Nuclear Power Plant
Instrument Setpoint Supplemental Information

The following additional information is being provided in response to staff comments associated with the review of the license amendment request dated April 9, 2002, with regards to determining limiting safety system settings (LSSS) and Allowable Values (AVs). The staff comments were discussed on February 11, 2004 by telephone and are listed below in italic. The participants in the conference call were:

RG&E: Tom Harding
John Guider
Ted Quinn (BusinessAmerica Corporation)

NRC: Bob Clark
Paul Rebstock
Chris Gratton

1. *All setpoint calculation assumptions must be adequately justified. For example:*

   - *The fact that the environment is controlled does not automatically indicate that the temperature effect is negligible. Show that the temperature range of the controlled environment does indeed result in negligible impact upon a device, given the sensitivity of the particular device and given the expected range of the temperature rise inside the cabinet or panel in which the device is located.*

   - *Power supply effects are not obviously negligible. Show that the anticipated power supply variation and the device sensitivity to it do indeed result in a negligible contribution to the device uncertainty.*

   - *Citations of "experience" as a justification for an assumption must be traceable to specific examples. Since the calculation addresses events and conditions that may rarely occur, the statistical significance of the cited experience must be explained.*

Response: Temperature affects are included in As-Found / As-Left Data for normal temperature changes, with the 95/95 calculation of drift per Generic Letter 91-04, and over the extended cycles that Ginna included. It is not the temperature rise within the cabinet, but the normal variation in temperature that the instrument experiences over the calibration cycles that causes drift which is measured and included. This is allowed for in ANSI/ISA-RP67.04-Part II-1994. Normal cabinet temperatures are generally less than 90°F, due to the design of the cabinets. The rack instrumentation temperature range is 40 - 120°F. Normal variations in control room temperature (70-78°F) are minor and would have an insignificant affect on instrument uncertainty.

Power supply effects are also included in As-Found / As - Left Data as normal variations over time. For Technical Specification instrument buses these are small variations, as they are controlled tightly and included in the plant specific drift values. Technical Specification instrument bus requirements are 113 VAC to 123 VAC (refer to the Bases for Technical Specification 3.8.9, Distribution Systems - MODES 1, 2, 3, and 4). Normal variations in voltage are < 0.3 VAC and would have an insignificant affect on instrument uncertainty. Typical instrumentation AC voltage requirements are 118 VAC +/-10%.

There are two kinds of assumptions, those that are verifiable, and those that use "good engineering judgement" per ANSI/ISA-S67.04-Part I-1994 and Regulatory Guide 1.105 Rev 3, and HICB 12. The guidance is to provide plant verified documentation for assumptions when possible. Documentation is not required to be provided for all assumptions. Providing plant specific drift data for Technical Specification setpoints is a valuable tool in verifying the performance and operating characteristics of the Ginna instruments under normal operating conditions. The assumptions which have been listed in the various calculations have been reviewed to confirm that the basis for good engineering judgement is defendable.

It is recognized that, by implementing the plant specific drift methodology described in ANSI/ISA-RP67.04-Part II-1994, Appendix E, Ginna is adopting a conservative approach to the development of instrument drift that is NRC approved in Generic Letter 91-04 as updated in HICB-12 and Regulatory Guide 1.105 Revision 3.

The derivation of drift, using this methodology, is much different from the vendor process that develops drift in a classical manner, under laboratory conditions. The value of plant specific drift includes normal variations in temperature and power supply as well as changes in Measurement and Test Equipment (M&TE) used and variations in how the technicians calibrate and record both as-found and as-left values. Using the formula for determination of 95/95 drift, the result provides conservative values as compared to those recommended by manufacturers and are more representative of the operation of that equipment in the specific nuclear application.

2. *It is not clear why instrument uncertainties should not need to be considered if they are less than the associated calibration tolerances. For example, an instrument uncertainty of 1 unit would certainly not be negligible when combined with a calibration tolerance of 1.1 units. The net uncertainty in such a case would be SRSS (1.0, 1.1) = 1.49. In addition, it is noted that the cited provision in RP67.04 applies only to RA, not to other components of the instrument uncertainty.*

Response: There are two issues here. The first is the issue of the use of either accuracy or calibration tolerance. There are a specific set of notes in ANSI/ISA-RP67.04-Part

II-1994, Section 6.2.6.2, that allows this, with the basic conditions of accuracy being verified. If a calibration is performed that meets the requirements for verification of the elements of accuracy including reference basic accuracy, repeatability, hysterisis, and deadband, then the ISA methodology allows the use of the larger of either accuracy or calibration tolerance. This methodology has been adopted at Ginna Station. The second consideration is basic statistical analysis. We need to ensure that our calculation is conservative using SRSS for those values that are random, independent and normal, and algebraic for those that are not. Per the ISA standard, the larger values are always overpowering in an SRSS calculation.

3. *Some of the values in the TLU computation in Section 10.2 (DA EE-92-041-21) do not correspond with the values derived earlier in the calculation. For example, the M&TEU, REU, and DU values presented on page 29 in the TLU computation for the Accident Indicator do not correspond with the values derived on pages 25 and 26. Licensee must either explain why the values should be different, or confirm that these and all other such errors have been located and corrected in all calculations used in support of the requested TS changes.*

Response: The values in the TLU computation for EOP Setpoint M.13 in Section 10.2 of calculation DA EE-92-041-21 do not correspond with the values derived earlier in the setpoint analysis due to the calibration tolerances being more conservative than the instrument inaccuracies (see NRC question 2). For this specific procedural calculation the instrument accuracies were excluded. ANSI/ISA-RP67.04-Part II-1994, Section 6.2.6.2 "Calibration Tolerance", states that "If the method of calibration or performance monitoring verifies all attributes of the reference accuracy and the calibration tolerance is larger than the reference accuracy, the larger value for the calibration tolerance may be substituted for the reference accuracy in the setpoint uncertainty calculation as opposed to inclusion of the calibration tolerance as a separate term" (i.e., including both the calibration tolerance and accuracy terms are not required). In cases where both the calibration tolerance and reference accuracy are both included, additional conservatism results.

4. *Proposed TS Table 3.3.5-1 refers to AVs defined outside the TS. The AV must be specified in the TS.*

Response: The setpoints for the Containment Radiation Instrumentation are not included in the proposed TS Table 3.3.5-1, since these setpoints are not in the existing TS. These instrumentation are not credited in the LOCA accident analyses dose calculations. That is, these functions are a backup to the Containment Isolation function for conservatism, but are not modeled within the analyses.

The high radiation containment ventilation isolation signal during a fuel handling accident in containment is credited in the current accident analysis, though the actual trip setpoint is not. The setpoints for these isolation functions are determined and adjusted in accordance with the methodology and parameters in the Offsite Dose Calculation Manual (ODCM) for 10 CFR Part 20 releases, as required by Technical Specification 5.5.4, Radioactive Effluent Controls Program, and as described in the UFSAR (Section 11.5.2.2.4). The 10 CFR Part 20 release limits are extremely conservative when compared to the 10 CFR Part 100 limits. Actual field setpoints are set at 40 % of the 10 CFR 20 limits, for additional conservatism. Also, as described in the RG&E submittal of May 21, 2003, for the change in dose methodology to Alternate Source Term, the isolation of containment following a fuel handling accident will no longer be credited. The Containment Radiation Instrumentation setpoints are contained within numerous documents within RG&E, including station procedures (such as calibration and setpoint procedures). The affected procedures and the UFSAR are controlled under 10 CFR 50.59. RG&E considers this sufficient control of the setpoints.