RG AND E A Subsidiary of RGS Energy Group, Inc.

ROCHESTER GAS AND ELECTRIC CORPORATION • 89 EAST AVENUE, ROCHESTER, N.Y. 14649-0001 • 585 546-2700 www.rge.com

ROBERT C. MECREDY
Vice President
Nuclear Operations

May 17, 2002

Mr. Robert L. Clark
Office of Nuclear Regulatory Regulation
U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

SUBJECT: 60 Day Response to NRC Bulletin 2002-01, Subject: *Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity*
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

REFERENCE: (1) Letter from R.C. Mecredy, RG&E, to R.L. Clark, NRC, Subject: *Response to NRC Bulletin 2002-01*, dated March 22, 2002.

Dear Mr. Clark:

In Reference (1), Rochester Gas and Electric Corporation (RG&E) provided the information required within 15 days of the issuance of NRC Bulletin 2002-01. The purpose of this letter is to provide the information required within 60 days of the issuance of NRC Bulletin 2002-01. Specifically, the bulletin stated in the Required Information section:

(3) Within 60 days of the date of this bulletin, all PWR addresses are required to submit to the NRC the following information related to the remainder of the reactor coolant pressure boundary:

A. the basis for concluding that your boric acid inspection program is providing reasonable assurance of compliance with the applicable regulatory requirements discussed in Generic Letter 88-05 and this bulletin. If a documented basis does not exist, provide your plans, if any, for a review of your programs.

The enclosure to this letter provides RG&E's basis for concluding that the boric acid inspection program at Ginna Station is providing reasonable assurance of compliance with the applicable regulatory requirements.

Any questions concerning this issue should be directed to Peter Bamford, Manager Primary/Reactor Systems at (585) 771-3832.

A095

1000473

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.

Very truly yours,

Executed on May 17, 2002

Robert C. Mecredy

Enclosure

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

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

U.S. NRC Ginna Senior Resident Inspector

ENCLOSURE
NRC BULLETIN 2002-01 60 DAY REQUIRED INFORMATION

The following provides RG&E's response to the information required within 60 days of the date of NRC Bulletin 2002-01 related to the boric acid inspection program at Ginna Station.

1. Boric Acid Inspection Program Definition and Responsibility

The program as presently defined at Ginna Station consists of both the routine scheduled inspections and "for cause" inspections of the reactor coolant system (RCS) pressure boundary. Ginna Station procedure A-1407, *Program to Prevent Degradation of Reactor Coolant Pressure Boundary Components from Boric Acid Corrosion*, administratively controls the overall inspection program and provides the methods of inspection. It also contains a listing of the carbon steel components in the RCS pressure boundary that may be susceptible to boric acid corrosion and provides location references and access capabilities for inspection (e.g., inspect certain valves only when the reactor is subcritical). This procedure was developed as a direct result of Generic Letter (GL) 88-05.

For leakage identification (i.e., "for cause" inspections), Ginna Station procedure A-1407 is designed to work with Ginna Station Operations procedures S-12.4, *RCS Leakage Surveillance Record Instructions*, and S-12.2, *Operator Action in the Event of Indication of Significant Increase in Leakage*. Operations personnel normally log RCS leak rate using procedure S-12.4 every shift. This includes documentation of containment sump pump actuations, RCS inventory balances, containment radiation monitor trends, and containment fan cooler condensate collection system activations. The RCS leak rate is also reviewed at each morning's management review meeting. When RCS unidentified leak rate increases to a pre-determined value below that specified in the Ginna Technical Specifications, Operations procedure S-12.2 is used to investigate and determine the location of the leakage.

For scheduled inspections, the ASME Section XI Code requires that VT-2 leakage examinations be performed on Class I components. This examination is performed in accordance with NDE procedure VT-109, *Visual Examination For Leakage*, and Ginna Station procedure PT-7, *ISI System Leakage Test, Reactor Coolant System* with the primary system fully assembled. Normally, this examination is performed at the end of each refueling outage (RFO), and can also be performed during RCS heatup after maintenance that affects the RCS pressure boundary. During this examination, qualified examiners from the Technical Performance and Field Inspection (TPFI) Group and Quality Control (QC) department look not only for signs of RCS leakage but also for signs of boric acid residue and for boric acid corrosion in the vicinity of identified leaks.

Additional informational inspections are also performed inside containment following shutdown for refueling outages. Generally, the QC group, the Mechanical Planning group, and/or radiation "decon" groups enter containment and inspect valves, flanges, etc. This ensures that leaking valves are identified early in the outage and the appropriate work orders written. Beginning with the recently completed 2002 refueling outage, this inspection was supplemented by documented inspections performed by TPFI personnel as a program improvement.

2. Inspection Scope and Frequency

As described above, the inspection program consists of both routine scheduled inspections and "for cause" inspections. For scheduled inspections, Ginna Station procedure PT-7 is performed during RCS heatup after each RFO with the primary system fully assembled. This inspection covers the full extent of the RCS pressure boundary as defined in ASME Section XI IWB 5000, and occurs at several RCS pressure plateaus (i.e., 300-350 psig, 1000-1600 psig, and 2235 psig). Observed leakage in any amount identified during PT-7 is documented and evaluated to NDE procedure VT-109.

With respect to "for cause" inspections, Ginna Station procedure S-12.4 utilizes several methods for leakage detection when the plant is in Modes 1-4. These methods are sensitive to increases in RCS leak rate of approximately 1% of the limit for unidentified leakage as specified in the Ginna Technical Specifications. Refer to the conclusion in Reference (1) for additional details. Ginna Station procedure A-1407 is used as directed by Ginna Operations procedures S-12.4 and S-12.2, when indicated leakage increases and the location of an RCS leak cannot be determined by other means. Procedure A-1407 identifies equipment and the component parts that are constructed of carbon steel near possible leakage points. It also identifies the location and special entry requirements needed to inspect the components when at power. Acceptance criteria for RCS leakage in Modes 1-4 are specified in the Ginna Station Technical Specifications, Section 3.4.13.

3. Obstructions to Visual Inspections

Ginna Station procedure PT-7 is performed following each RFO with the primary system fully assembled. With the exception of the reactor vessel flange, the insulation material at bolted connections is removed and is not installed until after the final inspections are completed. This examination is performed at several RCS pressure plateaus, including full RCS pressure.

While the inspection covered by procedure PT-7 covers the full extent of the RCS pressure boundary as defined in ASME Section XI IWB 5000, there are several areas where the insulation is not normally removed. This includes the reactor vessel head flange and heater penetrations at the bottom of the pressurizer. If leakage were to occur in these locations, evidence of leakage would be seen on the insulation, on the walls in the loop areas, or on the floor directly below the penetrations. Also, the containment A sump is inspected during the final plateau of PT-7 and evidence of leakage on the walls below the reactor nozzles would be visible if RCS leakage were present.

4. **Training**

Ginna Station Mechanical Maintenance (including Planners), Radiation Protection, and Operations personnel are capable of locating and evaluating the condition (wet, dry, or fluffy) and/or age (color) of boric acid deposits. However not all of these personnel have been formally trained in the specific use of procedure A-1407. This situation was identified during a self assessment (see item 6 below) and is being addressed.

TPFI personnel and QC inspectors are qualified in accordance with the ASME Code VT-2 visual inspection standard. The VT-2 examinations are used to determine the system leak test acceptability per the ASME code requirements.

5. **Response to Leakage**

Formal acceptance criteria for RCS leakage during performance of PT-7 is specified in NDE procedure VT-101. These procedures ensure that RCS leakage is within acceptable limits prior to reactor startup.

During plant operation, RCS leakage limits are specified by the Ginna Station Technical Specifications, Section 3.4.13. Unidentified leakage during plant operation is typically less than 10% of the limit of 1 gpm and monitored using procedure S-12.4. When RCS leak rate increases to a pre-determined value below that specified in the Technical Specifications, procedure S-12.2 is used to investigate and determine the location of the leakage. This determination is performed through the use of techniques that do not require the need to enter containment (e.g., trending all available level indicators and equipment start logs). It also includes the use of video cameras at selected locations inside containment.

After all means of leak detection outside containment have been exhausted, and the location of the leak is still not known, a containment entry is made using procedure A-1407 as a reference. Once a leak has been visually identified, corrective actions are taken commensurate with the safety significance of the leak. The assessment includes an evaluation of the impact of boric acid corrosion on carbon steel surfaces. Any identified degradation is entered into the Ginna Station corrective action program.

6. **Review of Program Effectiveness**

Self Assessment #2002-0037, *Effectiveness of the Ginna Station Program For Prevention of Boric Acid Corrosion*, has recently been performed. This assessment concluded that Ginna Station has an effective boric acid corrosion prevention program. It identified strengths of the program, and also areas for improvement which primarily involve enhanced documentation.

Numerous examples of prudent corrective actions for boric acid leaks were identified during the self assessment, indicating an effective program. Examples from past RFOs include correcting leaks at the non-regenerative heat exchanger and "B" residual heat removal (RHR) heat exchanger flanges, and the re-tensioning of the "B" reactor coolant pump main flange bolts. Other examples of an effective program include various stud removal activities for inspection at joints in accordance with ASME Section XI. Refer to Reference (1) for additional examples of the Ginna Station responses to identified RCS leakage and accumulation of boric acid deposits.

7. Assessment Relative to NUREG/CR-5576

On October 10-12, 1989, Ginna Station was audited by the NRC to determine if RG&E had implemented the boric acid corrosion prevention program requirements of GL 88-05. Ginna Station was also one of the survey participants that was included in NUREG/CR-5576. The audit results were found to be acceptable, as documented in Reference (2).

8. References

(1) Letter from R.C. Mecredy, RG&E, to R.L. Clark, NRC, Subject: *Response to NRC Bulletin 2002-01*, dated March 22, 2002.

(2) Letter from A. Johnson, NRC, to R.C. Mecredy, RG&E, Subject: *Prevention of Boric Acid Corrosion at Ginna Nuclear Power Plant*, dated August 7, 1990.