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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REVISION TO REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

ROCHESTER GAS AND ELECTRIC CORPORATION

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

1.0 INTRODUCTION

By Amendment No. 64 (Reference 7), issued on May 23, 1996 to Facility Operating License No. DPR-18, R. E. Ginna Nuclear Power Plant, the NRC approved changes to Section 5.6.6 of the Administrative Controls Technical Specifications (TS) to incorporate a reference to the methodology for determining pressure temperature (P/T) and low-temperature overpressure protection (LTOP) system limits and a Pressure Temperature Limits Report (PTLR). However, the transmittal letter stipulated that the approval was granted only until December 31, 1996. The December 31, 1996, date was later extended to December 31, 1997 (References 9 and 13), the date at which time the current P/T curves would expire.

By letter dated September 13, 1996 (Reference 8), RG&E submitted another application for an amendment to update the Ginna Station PTLR and revise the necessary section of the Administrative Control TS. The revised PTLR updated the reactor vessel material information, the P/T limit curves and the LTOP set points for Ginna.

As a result of questions raised concerning the LTOP enable liquid temperature measurement accuracy, RG&E submitted an application for an amendment dated April 24, 1997, including changes to Revision 2 of the PTLR (Reference 10) which replaced the application of September 13, 1996, (Reference 8). Also, as a result of further questions raised by the staff, RG&E submitted a letter of clarification (Reference 11), of their letter of April 24, 1997 (Reference 10).

Applying the LTOP instrument and calculational uncertainties required by the approved methodology resulted in an LTOP setpoint that would have established an operating window that would have been too narrow to permit reasonable system makeup and pressure control. To prevent these difficulties, by letter dated June 12, 1997 (Reference 12), RG&E requested to use ASME Code Case N-514, "Low Temperature Overpressure Protection," which designates the allowable pressure as 110 percent of that specified by 10 CFR Part 50, Appendix G and provides a different method for calculating the LTOP enable temperature.

Subsequently, by letter dated July 28, 1997 (Reference 14), the Commission granted Ginna an exemption from the requirements of 10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation" and allowed the plant to use the methodology in the ASME Code Case. Based on this exemption and further questions by the NRC staff, RG&E superseded the previous request for amendment and supporting documentation with an application for an amendment dated September 29, 1997, including changes to Revision 2 of the PTLR (Reference 15), which replaced the application of September 13, 1996, and April 24, 1997. By letter dated October 8, 1997, the licensee provided corrections to the information presented in the September 29, 1997 submittal.

This safety evaluation addresses the methodology for calculating the newly established P/T and LTOP system limits.

1.1 Neutron Fluence

The neutron fluence values provided by the licensee are used in determining the P/T limit curves. The fluence factors used to generate the values provided in PTLR, Revision 1 in March were estimates and not based on the methodology required by WCAP-14040-NP-A.

When the staff approved a pressurized thermal shock (PTS) evaluation for Ginna, the use of new fluence values and additional surveillance capsule chemistry data impacted that evaluation. Therefore, in June 1996, RG&E generated new fluence values and the neutron transport calculations were submitted. The revised submittals were reviewed and evaluated for acceptability of the fluence values for the pressure vessel. The neutron transport calculations were used to update the reactor vessel material information and the P/T limit curves.

1.2 Pressure Temperature Limits

The NRC recently reviewed and approved a PTS evaluation for Ginna Nuclear Power Plant by letter dated March 1996. They also reviewed the licensee's PTLR in May 1996. However, as previously stated, the use of new fluence values and additional surveillance capsule chemistry data impacted that evaluation. The impact on that evaluation is documented in WCAP-14684, and contains, in part, an update of all reactor vessel material information and an update of the P/T limit curves based on the WCAP-14040-NP-A methodology. The information in this safety evaluation focuses on the major differences between the information provided in the September 1997 submittal, to that provided in the March and May 1996 submittals.

The staff evaluates the P/T limits based on the following NRC regulations and guidance: Appendix G to 10 CFR Part 50, Generic Letters (GLs) 88-11 and 92-01; Regulatory Guide (RG) 1.99, Rev. 2; and Standard Review Plan (SRP) Section 5.3.2. Appendix G to 10 CFR Part 50 requires that P/T limits for the reactor vessel must be at least as conservative as those obtained by Appendix G to Section III of the ASME Code. GL 88-11 requires that licensees use the methods in RG 1.99, Rev. 2, to predict the effect of neutron irradiation by calculating an adjusted reference temperature (ART) of reactor vessel

material. The ART is defined as the sum of the initial reference temperature (RT_{ndt}) of the material, the mean value of the adjustment in reference temperature to account for neutron irradiation, and a margin to account for uncertainties in the predication method. The mean value of the adjustment in the reference temperature is calculated from the product of a chemistry factor and a fluence factor. The chemistry factor may be calculated using credible surveillance data, obtained by the licensee's surveillance program, as directed by Position 2 of RG 1.99, Rev. 2. If credible surveillance data is not available, the chemistry factor is calculated from Table 1 of RG 1.99, Rev. 2 as a function of the amounts of copper and nickel in the vessel material. GL 92-01 requires licensees to submit reactor vessel materials data, which the staff uses in the review of the P/T limits submittals. SRP 5.3.2. provides guidance on calculation of the P/T limits using linear elastic fracture mechanics methodology specified in Appendix G to Section III of the ASME Code. The linear elastic fracture mechanics methodology postulates sharp surface defects that are normal to the direction of maximum stress and have a depth of one-fourth of the reactor vessel beltline thickness (1/4T) and a length of 1.5 times the beltline thickness. The critical locations in the vessel for this methodology are the 1/4T and 3/4T locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively.

1.3 Low Pressure Overpressure Protection Setpoints

The changes provided in these submittals included revisions to the methodology for calculating the LTOP system enable temperature and actuation setpoint and establishing new values for these limits in accordance with the revised methodology. The revisions to the LTOP methodology were necessary in order to incorporate techniques presented in ASME Code Case N-514 for calculating enable temperature and lift setpoint and also to incorporate instrumentation uncertainties associated with the enable temperature.

The LTOP system mitigates overpressure transients at low temperatures so that the integrity of the reactor coolant pressure boundary is not compromised by violating 10 CFR Part 50, Appendix G. Ginna's LTOP system uses the pressurizer power-operated relief valves (PORVs) to accomplish this function. The system is manually enabled by operators and uses a single setpoint as the lift pressure for the PORVs. The design basis of Ginna's LTOP system considers both mass-addition and heat-addition transients during water solid reactor coolant system (RCS) conditions. The mass-addition analyses account for the injection from all three charging pumps. The heat-addition analyses accounts for heat input from the secondary sides of both steam generators (SGs) into the RCS, upon starting a single reactor coolant pump (RCP). The heat-addition transient analyses assume the SGs secondary side temperatures are 50°F higher than the RCS temperature. In addition, mass-addition analyses were performed for cases of injection from a single safety injection pump with the RCS vented through a 1.1 in² vent.

Ginna's proposed LTOP enable temperature and actuation setpoint were established using a plant-specific methodology in combination with ASME Code Case N-514. The plant-specific methodology is addressed in this safety evaluation.

2.0 EVALUATIONS

2.1 Neutron Fluence Evaluation

The fluence values were estimated using the two dimensional discrete ordinates code DORT in (r,θ) geometry and the BUGLE-93 cross section library which is based on the ENDF/B-VI data. Both forward and adjoint calculations were carried out with a P_3 approximation for the anisotropic scattering and a S_8 order of angular quadrature. Cycle specific sources were used for the adjoint calculations which were obtained from cycle reload reports for the past 25 cycles and the upcoming 26th cycle. The calculational methodology is acceptable because it complies with staff recommendations.

The four surveillance capsules which have been removed from Ginna were reanalyzed. The dosimetry suggested a measurement to calculation bias by about three percent higher than the analytical value. Thus, the calculated fluence values were increased by three percent for the estimation of the P/T limit curves. This is acceptable because the proposed value is conservative. Therefore, we conclude that the proposed fluence values for the estimation of the Ginna P/T limit curves are acceptable because they are conservative.

2.2 Pressure-Temperature Limits Evaluation

The methodology specified in the licensee's September 1996, submittal and again in the September 1997, submittal for calculating the P/T limit curves, including the adjusted reference temperature, is similar to that previously used except that the neutron fluence values, which are used in determining the P/T limit curves, were calculated using a different methodology. In addition, additional surveillance chemistry data became available. Since the surveillance weld was fabricated using the same heat number (weld wire heat 61782) as the limiting Ginna beltline weld, SA-847, the surveillance data may be used to calculate the best estimate chemistry of the beltline weld and the chemistry factor.

In WCAP-14684, an additional 8 values for the copper and nickel content of the surveillance weld were provided by the licensee. This altered the mean copper and nickel value for the surveillance weld to 0.236% and 0.517%, respectively. Using the methodology described in RG 1.99, Rev. 2, the chemistry factor for the surveillance weld was determined to be 159.4°F.

The data for the limiting weld in the Ginna reactor vessel comes primarily from four sources: (1) a weld dropout which includes 12 data points; (2) a weld dropout which contains 29 data points; (3) surveillance data which has 10 data points; and, (4) a weld qualification test which has 1 data point. The only new data since the March 22, 1996, evaluation, are the 8 data points from the surveillance weld. The mean values for the copper and nickel for each of these sources are (1) 0.204% and 0.489%, respectively for the weld dropout with 12 data points; (2) 0.270% and 0.586%, respectively for the weld dropout with 29 data points; (3) 0.236% and 0.517%, respectively for the surveillance data; (4) 0.31% and 0.64%, respectively for the weld qualification test. All of the above data is for welds fabricated using heat number 61782 weld wire. By averaging all of the individual data points (i.e., 52 data points) which are included in the determination of the mean values above, a mean copper value of 0.249 and a mean nickel value of 0.551 is obtained. By averaging the mean values from the four sources (i.e., the mean of the means), one obtains a mean copper value of 0.255% and a mean nickel value of 0.558%. In the PTLR, the licensee used 0.25% and 0.56% for the values of copper and nickel in the circumferential weld seam SA-847, respectively.

In an evaluation dated March 22, 1996, the NRC indicated that a simple average of the copper data does not always represent a best estimate of the copper in a weld since there could be large coil-to-coil variability in the amount of copper because the copper coating can vary on the filler wire. In addition, the staff indicated that it had discussed the coil-to-coil variability concern with the licensee and that the licensee indicated that it could not accurately determine how many coils were used to fabricate the four welds that are included in the database. As a result, the staff evaluated various values for the mean copper content for the vessel weld (i.e., circumferential weld seam SA-847). If the number of coils in the samples are not known, the mean of the means approach is the preferred approach for determining the best estimate of copper and nickel in the limiting weld. The staff, however, concludes that the use of 0.25% for the percentage of copper and 0.56% for percentage of nickel in the limiting weld is acceptable in this case. This conclusion is based on the limited number of data points from the weld qualification test (i.e., one data point with 0.31% copper). Because the mean value of copper for the weld qualification test is significantly greater than the mean value from the other three sources of data, the one data point from the weld qualification test, which is also the mean value, significantly affects the best estimate value when the mean of the means approach is used. Since more samples were obtained for the other 3 sources of data, the staff concludes that the use of 0.25% for the percentage of copper and 0.56% for the percentage of nickel in the limiting weld is acceptable. The chemistry factor for a weld with 0.25% copper and 0.56% nickel is 170.4°F.

Since the mean values for the copper and nickel in the surveillance weld changed, and the mean values for the copper and nickel in the circumferential weld seam SA-847 were modified, as discussed above, the ART and P/T limit curves were modified by the licensee to reflect the additional data and analysis. The staff reviewed the licensee's results using the above referenced regulations and guidance and concludes that the licensee's ART values and P/T curves are acceptable based on the present data.

The proposed ART values and P/T limits are acceptable based on the neutron fluence values and chemistry data provided by the licensee. Since this conclusion is dependent upon the available chemistry and surveillance data, it is subject to change when additional data become available.

2.3 Low Pressure Overpressure Protection Set Points Evaluation

2.3.1 LTOP Enable Temperature

The LTOP enable temperature is the temperature below which the LTOP system is required to be operable. The licensee proposed to revise the LTOP enable temperature methodology to: 1) account for instrument uncertainties associated

with the instrumentation used to enable the LTOP system and, 2) implement the ASME Code Case methodology of using an enable RCS liquid temperature corresponding to the reactor vessel ¼-t metal temperature of RT_{NDT} + 50 or 200°F, whichever is greater. Therefore, the licensee proposed to calculate the enable temperature as RT_{NDT} + 50°F + $E_{instrument}$ + delta-$T_{¼-t}$. In this calculation, RT_{NDT} refers to the highest ART for welds or base metals in the beltline region at a distance one-fourth of the vessel section thickness from the vessel inside surface, as determined by RG 1.99, Rev. 2; $E_{instrument}$ refers to instrument error; and delta-$T_{¼-t}$ refers to the temperature difference between the reactor coolant and the metal at a distance one-fourth of the vessel thickness from the inside surface in the vessel beltline region.

The use of 50°F in the above methodology is consistent with ASME Code Case N-514. Accounting for the delta-$T_{¼-t}$ is consistent with Branch Technical Position (BTP) RSB 5-2, which states that the enable temperature is defined as "the water temperature corresponding to the metal temperature...at the beltline location (1/4t or 3/4t) that is controlling in the Appendix G limit calculation." This approach is also consistent with the ASME Code Case. Accounting for instrument uncertainty ensures that the LTOP system is not enabled at temperatures less conservative than is required by the aforementioned documents. Based on the above discussion and the July 1997 letter approving the use of ASME Code Case N-514 at Ginna, the staff finds the licensee's proposed revisions to the LTOP enable temperature methodology acceptable.

In addition to the above revisions, the licensee proposed to add wording to allow the LTOP system to be designed to protect the RCS to 110 percent of the steady-state P/T curve as calculated in accordance with Appendix G to 10 CFR Part 50. This is also consistent with ASME Code Case N-514 and is therefore acceptable based on the approval letter of July 1997.

The licensee proposed an LTOP enable temperature of 322°F. Based on Ginna's vessel material data, the limiting RT_{NDT} is 232°F. Additionally, licensee calculations show that the temperature difference between the coolant and the ¼-t location is 3.2°F and that the uncertainty associated with the RCS cold leg temperature instrumentation used for LTOP is ±21.1°F. Therefore, the minimum allowed LTOP enable temperature for Ginna is 306.3°F (232°F + 50°F + 3.2°F + 21.1°F). For an RT_{NDT} value of 232°F, the proposed LTOP enable temperature of 322°F is conservative with respect to the minimum LTOP enable temperature allowed by ASME Code Case N-514 and the plant's proposed methodology. Therefore, the staff finds this change acceptable.

2.3.2 LTOP Actuation Setpoint

LTOP systems are usually designed to mitigate overpressure transients at low temperatures to prevent violating 10 CFR Part 50, Appendix G P/T limits. ASME Code Case N-514 allows the LTOP system be designed to limit the peak pressure at the controlling location to 110 percent of 10 CFR Part 50 Appendix G P/T limits. Additionally, since overpressure events most likely occur during isothermal conditions in the RCS, the NRC has accepted the use of the steady-state Appendix G limits for the design of LTOP systems. The LTOP actuation

setpoint is the pressure at which the PORVs will lift, when the LTOP system is enabled, to limit the peak RCS pressure during a pressurization transient.

Ginna's LTOP methodology for determining the LTOP actuation setpoint was approved by the staff by letter dated May 1996. Revisions proposed in this request are only intended to incorporate the allowance in ASME Code Case N-514 which permits the actuation setpoint to be determined in a manner which protects 110 percent of the Appendix G P/T limits instead of the actual limits. Since the staff has already approved the Code Case for use at Ginna, these changes are acceptable. Other minor editorial and clarifying changes were proposed. These changes did not affect the methodology for calculating the actuation setpoint and are therefore also acceptable.

The licensee proposed an LTOP actuation setpoint of ≤ 411 psig which was calculated in accordance with the proposed methodology. Licensee analyses related to the LTOP system were submitted with the June 3 and September 29, 1997 letters. In these analyses, a PORV lift setpoint of 430 psig was assumed. This value conservatively accounts for the lift setpoint of 411 psig plus actuation channel uncertainty of 16.95 psig as calculated by the licensee.

The licensee performed analyses for two categories of mass-addition transients. The first category assumed the RCS was water solid and relied on a pressurizer PORV to mitigate the overpressurization event. For this category, the licensee identified the limiting scenario as the injection from all three charging pumps with the RCS at 60°F and both RCPs running. The 60°F temperature is the lowest temperature at which the reactor head is allowed to be bolted and corresponds to the most limiting (lowest) Appendix G P/T curves pressure. Assuming all RCPs running maximizes the dynamic head effect in the analyses which in turn maximizes the pressures achieved. This scenario resulted in a maximum pressure of 587.4 psia which was less than the allowed limit of 608.7 psia (110% of the 540 psig P/T limit curve value at 60°F).

The second mass-addition category assumed the RCS was vented through a 1.1 in^2 vent and relied on this vent for overpressure mitigation. For this category, two analyses were performed to bound the temperatures at which the system is allowed to be in this configuration (i.e., with the vent established). One analysis was performed at the 60°F boltup temperature discussed above while the other was performed at 212°F - the maximum temperature at which the vent could be established. Both analyses assumed that the RCPs were not operating. For these analyses, the assumption of the RCPs not running is consistent with plant operating configurations since the pumps are not allowed to be run with the vents open. Peak pressures achieved for these scenarios were 413.5 psia and 396.7 psia for the 60°F and 212°F cases, respectively. These pressures were below the respective pressure limits of 608.7 psia and 780.3 psia.

The licensee also performed analyses for heat addition cases as described in the proposed methodology. The licensee performed a total of five heat addition transients with different RCS initial temperatures to bound the LTOP region. All heat addition transients were run with the RCS in a water solid condition. For these analyses, the secondary system was assumed to be 50°F

higher than the RCS. One RCP is assumed to start and consequent heat addition from both SGs is accounted for. Expansion of the reactor coolant results in pressurization of the RCS and actuation of the LTOP system. The licensee identified the most limiting heat addition transient as the case with the RCS initially at 60°F and 329.7 psia. This analysis resulted in a peak pressure of 551.3 psia which was below the limit of 608.7 psia.

The licensee further confirmed that the residual heat removal (RHR) system pressure limit was not challenged in any of the cases analyzed. The heat addition case at 280°F was determined to be the most limiting from an RHR system pressure perspective. This case resulted in a peak RHR system pressure of 663.66 psia which was below the limit of 674.7 psia.

For all cases analyzed, the licensee conservatively assumed one PORV failed and, therefore, credited only one PORV for pressure relief. Additionally, the licensee did not credit the RHR system relief valves for mitigating the pressure transient.

Based on the above discussion, the staff finds acceptable the licensee's proposed LTOP actuation setpoint of ≤ 411 psig.

3.0 CONCLUSIONS

Based on the staff evaluation discussed in Section 2.0 above, we conclude that the material properties, the neutron fluence, and the methodologies for determining the P/T limit curves are acceptable. The staff has reviewed the licensee's proposed revisions to the methodology for calculating the LTOP system's enable temperature and actuation setpoint. The staff finds these changes consistent with BTP RSB 5-2 and ASME Code Case N-514 which was approved for use at Ginna in a July 28, 1997 letter; and therefore, acceptable. The staff has also reviewed the licensee's analyses related to the proposed enable temperature of 322°F and actuation setpoint of ≤ 411 psig.

The staff finds that the licensee's analyses were performed in a manner consistent with the proposed methodology. The staff further agrees with the licensee's conclusion that the analyses conservatively demonstrated that the RCS and RHR system pressure limits will be met with these settings and, therefore, finds the settings acceptable.

4.0 REFERENCES

1. Amendment No. 48 to Facility Operating License No. DPR-18, R. E. Ginna Nuclear Power Plant, March 6, 1992.

2. Letter from C. I. Grimes, NRC, To R. A. Newton, Westinghouse Electric corporation, "Acceptance for Referencing of Topical Report WCAP-14040, Revision 1, 'Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,'" October 16, 1995. (Also known as WCAP 14040-NP-A).

3. Letter from R. C. Mecredy, Rochester Gas and Electric Corporation (RG&E), to Document Control Desk, NRC, Attention A. R. Johnson,

"Application for Amendment to Facility Operating License Methodology for Low Temperature Overpressure Protection (LTOP) Limits," February 9, 1996.

4. Letter from A. R. Johnson, NRC, to R. C. Mecredy, Rochester Gas and Electric Corporation (RG&E), "R. E. Ginna Nuclear Power Plant - Pressurized Thermal Shock Evaluation (TAC No. M93827)," March 22, 1996.

5. Letter from R. C. Mecredy, Rochester Gas and Electric Corporation (RG&E), to Document Control Desk, NRC, Attention A. R. Johnson, "Application for Amendment to Facility Operating License Revision to Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," April 22, 1996.

6. Letter from J. A. Mitchell, NRC, to R. C. Mecredy, Rochester Gas and Electric Corporation, "R. E. Ginna - Acceptance for Referencing of Pressure Temperature Limits Report, Revision 1 (TAC No. M94770)," May 23, 1996.

7. Amendment No. 64 to Facility Operating License No. DPR-18, R. E. Ginna Nuclear Power Plant, May 23, 1996.

8. Letter from R. C. Mecredy, Rochester Gas and Electric Corporation to G. S. Vissing, NRC, "Application for Amendment to Facility Operating License, Revision to Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR), Rochester Gas and Electric Corporation, R. E. Ginna Nuclear Power Plant, Docket No. 50-244," September 13, 1996. (WCAP 14684 Attached).

9. Letter from G. S. Vissing, NRC, to R. C. Mecredy, Rochester Gas and Electric Corporation, "R. E. Ginna - Acceptance of Request to Extend Time for Approval of Revision of Pressure and Temperature Limits Report (PTLR) (TAC No. M97313)," December 10, 1996.

10. Letter from R. C. Mecredy, Rochester Gas and Electric Corporation to G. S. Vissing, NRC, "Application for Amendment to Facility Operating License, Revision to Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR) Administrative Controls Requirements, Rochester Gas and Electric Corporation, R. E. Ginna Nuclear Power Plant, Docket No. 50-244," April 24, 1997.

11. Letter from R. C. Mecredy, Rochester Gas and Electric Corporation to G. S. Vissing, NRC, "Clarifications to Proposed Low Temperature Overpressure Protection (LTOP) Technical Specification, Rochester Gas and Electric Corporation, R. E. Ginna Nuclear Power Plant, Docket No. 50-244," June 3, 1997.

12. Letter from R. C. Mecredy, Rochester Gas and Electric Corporation, "Request for Exemption from 10 CFR 50.60, Acceptance Criteria for Fracture Prevention for Light Water Nuclear Power Reactors for Normal Operation," June 12, 1997.

13. Letter from G. S. Vissing, NRC to R. C. Mecredy, Rochester Gas and Electric Corporation, "R. E. Ginna - Acceptance of Request to Extend Time for Approval of Revision of Pressure and Temperature Limits Report (PTLR) (TAC No. M98992)," June 26, 1997.

14. Letter from G. S. Vissing, NRC to R. C. Mecredy, Rochester Gas and Electric Corporation, "Exemption from the Requirements of 10 CFR Part 50.60, Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation - R. E. Ginna Nuclear Power Plant (TAC No. M98993)," July 28, 1997.

15. Letter from R. C. Mecredy, Rochester Gas and Electric Corporation to G. S. Vissing, NRC, "Application for Amendment to Facility Operating License, Revision to Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR) Administrative Controls Requirements, Rochester Gas and Electric Corporation, R. E. Ginna Nuclear Power Plant, Docket No. 50-244," September 29, 1997.

16. Letter from R. C. Mecredy, Rochester Gas and Electric Corporation to G. S. Vissing, NRC, "Corrections to Proposed Low Temperature Overpressure Protection Technical Specifications, Rochester Gas and Electric Corporation, R. E. Ginna Nuclear Power Plant, Docket No. 50-244," October 8, 1997.

Principal Contributors: B. Elliot
K. Karwoski
L. Lois
M. Shuaibi
M. Weston
G. Vissing

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