UNITED STATES
NUCLEAR REGULATORY COMMISSION
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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 & ELECTRIC CORPORATION

R.E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

1.0 INTRODUCTION

In a letter dated April 22, 1996, the Rochester Gas & Electric Corporation (the licensee) provided a License Amendment Request (LAR) that proposes to revise the R.E. Ginna Nuclear Power Plant (Ginna) Pressure and Temperature Limit Report (PTLR). The PTLR revision updates the reactor vessel material information, the pressure-temperature (P-T) limit curves and the Low Temperature Overpressure Protection (LTOP) set points for Ginna. The reactor vessel material information is used to calculate the P-T limit curves. The P-T limit curves are used to determine the LTOP set points. By a letter dated May 5, 1995, supplemented by letters dated July 17, 1995, December 8, 1995, February 9, 1996, March 15, 1996, and April 22, 1996, RG&E submitted a specific methodology for determining Low Temperature Overpressure Protection (LTOP) setpoints at the Ginna Nuclear Power Plant for NRC staff review and approval. The methodology presented in this submittal has been used by the licensee to develop LTOP setpoints in its PTLR.

The staff evaluates the P-T limits based on the following NRC regulations and guidance: Appendix G to 10 CFR Part 50; Generic Letters (GL) 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 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 prediction method. The mean value of the adjustment in 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 Regulatory Guide (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 1-1/2 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.

The licensee's P-T limit curves were developed using the methodology of WCAP-14040 except that neutron fluence values were not calculated using the ENDFB-VI methodology. On the date of the submittal, the fluence calculation for 22 effective full power years of operation (EFPYs) was not available. Therefore, the licensee proposed to increase the previously used fluence by 7%. The P/T curves and the LTOP limit determination were based on this 7% fluence increase. The P-T limit curves are applicable for 22 effective full power years (EFPY), or until the summer of 1998. The P-T limits also include a leak test limit curve that is applicable to 20 EFPY, to be used during the 1996 Refueling Outage.

Since the P-T limit curves were not calculated using the neutron fluence methodology of WCAP-14040, the licensee will provide a revised LAR by the end of 1996 that includes an evaluation of the effect of the ENDFB-VI methodology on the P-T limit curves.

2.0 EVALUATION

2.1 Neutron Fluence

The licensee is requesting extension of the P/T curves and LTOP limits to 22 EFPYs of operation. Current limits are applicable to 21 EFPYs. The neutron fluence at 22 EFPYs is 1.91×10^{19} n/cm^2 (E>1.0MeV) at 1/4T in the peak radial direction (0°). This includes a 7% increase in the previously approved value. We estimate that with the assumed load factor of 0.85 for the remainder of calendar year 1996 the Ginna plant will increase its neutron fluence by about $.03 \times 10^{19}$ n/cm^2. This is less than the assumed 7% increase on which the P/T curves and the LTOP limits are based. Therefore, for the period for which the SE is valid, the assumed fluence of 1.91×10^{19} n/cm^2 at 1/4T is conservative.

2.2 Reactor Vessel Materials

In a letter to Dr. Robert C. Mecredy from Allen R. Johnson dated March 22, 1996, the staff evaluated the Ginna reactor vessel materials. The staff's March 22, 1996, evaluation concludes that the Ginna surveillance data comply with the credibility criteria of RG 1.99, Rev. 2 and the surveillance data should be used to determine the chemistry factor for the limiting Ginna vessel weld. In the March 22, 1996 evaluation, the staff determined that the ratio of the chemistry factor of the vessel weld to the chemistry factor of the surveillance weld is 1.13. This ratio is used to adjust the surveillance data to account for the difference in chemical composition between the vessel and

surveillance welds.

The licensee used its weld surveillance data to determine the chemistry factor for the limiting Ginna vessel weld. The neutron fluence values for the surveillance capsules were adjusted to account for the difference in the ENDFB-VI methodology required by WCAP-14040 and the methodology used at the time the capsules were evaluated. The surveillance data was also adjusted using the ratio factor of 1.13 to account for the difference in chemical composition between the vessel and surveillance welds. The licensee calculated the chemistry factor using surveillance data, in accordance with the methodology in RG 1.99, Rev. 2. The chemistry factor was multiplied by the fluence factors at the 1/4T and 3/4T wall locations to determine the mean values of the adjustment in reference temperature at 22 EFPY. The fluence factors were calculated using neutron fluence values that the licensee indicates conservatively account for the ENDFB-VI methodology.

The licensee added the margin value and initial reference temperature that were approved by the staff in the March 22, 1996, letter to the mean value of the adjustment in reference temperature from the surveillance data to calculate the adjusted reference temperatures at the 1/4T and 3/4T wall locations at 22 EFPY for the limiting reactor vessel material. These values are acceptable because they were calculated in accordance with the methodology in RG 1.99, Rev. 2. However, since the calculation is dependent on the neutron fluence values for the surveillance capsules and for the 1/4T and 3/4T locations, this calculation will need to be reviewed after the licensee revises the neutron fluences using the ENDFB-VI methodology.

2.3 P-T Limit Curves

The P-T limit curves were calculated using the methodology of WCAP-14040 except that conservative estimates of neutron fluence values are used instead of the required ENDFB-VI methodology. WCAP-14040 was reviewed by the staff in a letter from C.I. Grimes to R.A. Newton dated October 16, 1995. The staff's review of the WCAP-14040 methodology indicates that the methodology will result in P-T limit curves that will conform to Appendix G of 10 CFR Part 50 and SRP Section 5.3.2. However, the P-T limit curves will need to be reviewed after the licensee revises the neutron fluence values using the ENDFB-VI methodology.

2.4 Methodology For Determining LTOP Setpoints

Attachment II to RG&E's letter dated December 8, 1995, provides a final version of the specific methodology for determining LTOP setpoints for Ginna Nuclear Power Plant. The LTOP is designed to provide the capability, during reactor operation at low temperature conditions, to automatically prevent the RCS pressure from exceeding the applicable limits established by 10 CFR Part 50 Appendix G requirements. The LTOP is manually enabled by reactor operators based on its predetermined enable temperature during reactor startup and shutdown. RG&E's specific methodology for determining LTOP setpoints covers only the use of the pressurizer power operated relieve valves (PORVs) as the means of LTOP. RG&E's specific methodology does not cover a methodology to use relief valves of the RHR system as the means of LTOP.

2.4.1 Transients Considered for LTOP

The licensee's methodology for LTOP includes the consideration of both the heat addition transient and the mass addition transient.

The LTOP is designed to mitigate all potential transients that lead to overpressure of RCS during low temperature operating conditions. Both heat addition and mass addition transients are considered in the design of LTOP. Both of these scenarios assume no RHR System heat removal capability and the RCS in a water-solid condition.

For the heat addition transient, RG&E assumed that a reactor coolant pump in a single loop is started with the RCS temperature as much as 50 F lower than the steam generator (SG) secondary side temperature. This results in a sudden heat input to a water-solid RCS from the SG, creating an increasing pressure transient.

For the mass addition transient, two scenarios are considered in the design of the LTOP system. The first scenario is an inadvertent actuation of the safety injection pumps into the RCS. The second scenario is the simultaneous isolation of the RHR System, isolation of letdown, and failure of the normal charging flow controls to the full flow condition. Either scenario may be eliminated from consideration depending on plant configurations that are restricted by plant technical specifications. Also, various combinations of charging and safety injection flows may be evaluated on a plant specific basis. The resulting mass addition transient causes an increasing pressure transient.

The staff considers the licensee's methodology acceptable because these design criteria ensure that the most limiting transients will be analyzed in the design of LTOP.

2.4.2 Pressure Limits for Design of LTOP

The major function of LTOP is to protect the reactor vessel from a fast-propagating brittle fracture. In order to achieve this purpose, the design of the LTOP is based on satisfying the steady state 10 CFR 50, Appendix G, limit curve. Also, the LTOP provides for an operational consideration to maintain the integrity of the PORV piping by preventing ductile fracture.

The PORV set point is dependent upon two limits. The upper limit is established to protect the reactor vessel from brittle fracture. The lower limit of the PORV set point is based on an operational consideration for maintaining a normal pressure differential across the RCP number 1 seals for proper RCP operation. Where there is insufficient range between the upper and lower pressure limits, the upper pressure limit shall take precedence.

2.4.3 Determination of PORV Setpoints for LTOP

The methodology for developing the PORV setpoints is intended to provide adequate protection for reactor vessel integrity and to maintain proper operational margins. In calculation of the PORV setpoints, a list of plant

parameters and transient conditions listed in Section 3.2.1 of the methodology are considered. This list includes initial RCS and steam generator parameters, PORV size and lifting characteristics, mass and heat input rate to the RCS, and pressure limits to be protected. These parameters are modelled in the BWNT RELAP5/MOD2-B&W computer code, which is documented in a topical report BAW-10164P-A. The computer code calculates the maximum and minimum RCS pressures due to over shoot and under shoot of RCS pressure under various overpressure transient conditions.

Either of the two PORVs with its setpoint determined by this process would protect upper P/T limits assuming a single failure of the other PORV.

Section 3.2.5 of the methodology originally indicated that, since the P/T limits are conservatively determined, the uncertainties in the pressure and temperature instrumentation utilized by the LTOP would not be explicitly accounted for in the selection of the PORV setpoints for LTOP. The staff concluded that this approach was not acceptable. In response to the staff request, RG&E in its letter dated December 8, 1995, modified Section 3.2.5 to indicate that the uncertainties in the pressure and temperature instrumentation utilized by the LTOP would be accounted for consistent with the methodology of the Instrument Society of America (ISA) Standard 67.04-1994. We consider that the RG&E proposed change of methodology regarding its treatment of instrumentation uncertainties is acceptable.

2.4.4 Enable Temperature for LTOP

The enable temperature is the temperature below which the LTOP is required to function. This temperature is currently determined per Branch Technical Position RSB 5-2 which is an attachment to Standard Review Plan (SRP) 5.2.2. It is the water temperature corresponding to a metal temperature of at least RT_{NDT} + 90 F at the reactor vessel beltline location (1/4t or 3/4t). Above this temperature, brittle fracture of the reactor vessel is not expected. RG&E's methodology of determining the enabled temperature is consistent with the staff position stated in BTP RSB 5-2 and, therefore, is acceptable.

2.4.5 Application of ASME Code Case N-514

ASME Code Case N-514 provides 1) selection of the setpoints of LTOP to limit the maximum pressure in the reactor vessel to 110% of the P/T limits determined per the requirement of 10 CFR 50 Appendix G, and 2) selection of the enabled temperature of LTOP using 200 F or the reactor coolant temperature corresponding to a reactor vessel metal temperature of at least RT_{NDT} + 50 F, whichever is greater.

In Section 3.3 of the methodology, RG&E states that the application of ASME Code Case N-514 has not yet been approved by the NRC. In the interim, an exemption to the regulations must be approved by the NRC were the licensee to desire to apply this Code Case in a licensee's design of LTOP. The staff agrees with this statement.

2.5 PTLR Relative to PORV Setpoint

Attachment IV to RG&E's letter dated April 22, 1996, provides a final version of the PTLR developed for the operation at R.E. Ginna Nuclear Power Plant. Using the methodology presented in the Attachment II to RG&E's letter dated December 8, 1995, Section 2.0 of the final revision of the PTLR defines that the enable temperature of the LTOP system is 328 F and the lift setting for the PORVs is less or equal to 411 psig. The staff has evaluated those setpoints with the information provided in Attachment IV to RG&E's letter dated February 9, 1996, and RG&E letter dated March 15, 1996, and finds that they are conservative and will not result in transients that exceed the P/T limits as defined in Figures 1 and 2 of the final revised PTLR in Attachment IV to RG&E's letter dated April 22, 1996.

3.0 CONCLUSIONS

Based on the staff evaluation discussed in Section 2.0 above, we conclude that the material properties, the estimated neutron fluence, and the methodologies for determining the P-T limit curves are acceptable. In addition, the staff concludes that RG&E's methodology for developing LTOP and Section 2.0 of the revised PTLR developed for the operation at R.E. Ginna Nuclear Power Plant as they are presented in Attachment II to RG&E's letter dated December 8, 1995, and Attachment IV to RG&E's letter dated April 22, 1996, respectively are acceptable. This review is applicable only through the end of 1996. Extension of the staff's conclusion of acceptability of RG&E's results through 22 EFPY is dependent upon satisfactory resolution of the fluence issue. In a teleconference on April 25, 1996 the licensee committed to resolve this issue by submitting the full report on P/T and LTOP using the correct fluence value by the end of September 1996, and to modify the P/T curves and the LTOP limits accordingly should any modification be necessary.

Principal Contributors: R. Giardina
B. Elliot

Date:

ensure that these limits are acceptable based on the approved methodology. At that time, RG&E will be required to revise the technical specifications Administrative Controls section to reference the staff approved methodology.

Sincerely,

ORIGINAL SIGNED BY:

Jocelyn A. Mitchell, Acting Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-244

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Sincerely,

Jocelyn A. Mitchell, Acting Director
Project Directorate I-1
Division of Reactor Projects
Office of Nuclear Reactor Regulation

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