Docket No. 50-244 March 6, 1992

Dr. Robert C. Mecredy
Vice President, Nuclear Production
Rochester Gas & Electric Corporation
89 East Avenue
Rochester, New York 14649

Dear Dr. Mecredy:

SUBJECT: ISSUANCE OF AMENDMENT NO. 48 TO FACILITY OPERATING LICENSE NO. DPR-18 - R. E. GINNA NUCLEAR POWER PLANT (TAC NO. M79828)

The Commission has issued the enclosed Amendment No. 48 to Facility Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant. This amendment is in response to your application dated February 15, 1991, as supplemented on May 14, 1991.

This amendment revises the pressure-temperature (P-T) limits for the reactor coolant system, in the Ginna Technical Specifications (TS), during heatup, cooldown, leak test, and criticality. The revised P-T limits were developed by you to comply with the NRC position on radiation embrittlement of reactor vessel materials and its effect on plant operations, outlined in Regulatory Guide (RG) 1.99, Revision 2, and Generic Letter 88-11 guidance. The revised P-T limits also considered a re-evaluation of the low temperature overpressurization protection system (LTOPS) setpoint.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly FEDERAL REGISTER notice.

Sincerely,
Original signed by Morton B. Fairtile
for Allen Johnson, Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 48 to License No. DPR-18
2. Safety Evaluation

cc w/enclosures:
See next page

*See previous concurrence

OFC	:LA:PDI-3	:PM:PDI-3	:EMCB*	:SRXB*	:OGC*
NAME	:MRushbrook	:AJohnson:sk	:JTWiggins	:RJones	:MZobler
DATE	:3/5/92	:3/5/92	: 2/12/92	: 2/13/92	: 2/19/92

OFC	:(A)D:PDI-3	:	:	:	:
NAME	:VNerses	:	:	:	:
DATE	:3/6/92	:	:	:	:

OFFICIAL RECORD COPY / Document Name: GI AMEND M79828

DF01
1/1

9203120288 920306
PDR ADOCK 05000244
P PDR

Dr. Robert C. Mecredy Ginna

cc:

Thomas A. Moslak, Senior Resident Inspector
R.E. Ginna Plant
U.S. Nuclear Regulatory Commission
1503 Lake Road
Ontario, New York 14519

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

Ms. Donna Ross
Division of Policy Analysis & Planning
New York State Energy Office
Agency Building 2
Empire State Plaza
Albany, New York 12223

Charlie Donaldson, Esq.
Assistant Attorney General
New York Department of Law
120 Broadway
New York, New York 10271

Nicholas S. Reynolds
Winston & Strawn
1400 L St. N.W.
Washington, DC 20005-3502

Ms. Thelma Wideman
Director, Wayne County Emergency
Management Office
Wayne County Emergency Operations Center
7370 Route 31
Lyons, New York 14489

Ms. Susan Perry
Administrator, Monroe County Office of
Emergency Preparedness
111 West Fall Road
Rochester, New York 14620

AMENDMENT NO. 48 TO DPR-18 - R. E. GINNA NUCLEAR POWER PLANT DATED March 6, 1992

DISTRIBUTION:
Docket File 50-244
NRC PDR
Local PDR
PDI-3 Reading
S. Varga
J. Calvo
M. Rushbrook
A. Johnson
V. Nerses
OGC - 15 B18
Dennis Hagan - MNBB 3206
E. Jordan - MNBB 3701
B. Grimes - 9 A2
G. Hill (4) - P1-37
Wanda Jones - MNBB 7103
ACRS (10) - P-315
PA - 2 G5
OC/LFMB - MNBB 11104
C. Grimes - 11F23
J. Linville, Region I
R. Lobel
C. Y. Cheng
R. Jones
J. Tsao - EMCB, 7D4
C. Liang - SRXB, 8E23

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

ROCHESTER GAS AND ELECTRIC CORPORATION

DOCKET NO. 50-244

R. E. GINNA NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 48
License No. DPR-18

1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:

   A. The application for amendment filed by the Rochester Gas and Electric Corporation (the licensee) dated February 15, 1991, as supplemented on May 14, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-18 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 48 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Victor Nerses

Victor Nerses, Acting Director
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 6, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 48

FACILITY OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
3.1-10	3.1-10
------	3.1-10a*
3.1-11	3.1-11
3.1-12	3.1-12
3.1-13	3.1-13
3.1-14	3.1-14
3.1-15	-------
3.1-16	-------
3.1-16a	-------
3.1-17	-------
3.1-18	-------
3.3-4a	3.3-4a
------	3.3-4b*
3.3-13	3.3-13
3.3-14	3.3-14
------	3.3-14a*
3.15-1	3.15-1
3.15-2	3.15-2

*Denotes new page.

3.1.2 Heatup and Cooldown Limit Curves for Normal Operation

3.1.2.1 The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.1-1 and 3.1-2 for the first 21.0 effective full power years.

a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. The heatup and cooldown rates shall not exceed 60°F/hr and 100°/hr, respectively. Limit lines for cooldown rates between those presented may be obtained by interpolation.

b. Figures 3.1-1 and 3.1-2 define limits to assure prevention of non-ductile failure only. The limit lines shown in Figures 3.1-1 and 3.1-2 shall be recalculated periodically using methods discussed in the Basis Section.

c. If the limits on Figures 3.1-1 and 3.1-2 are exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; and either

   1) within 6 hours, perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System and determine that the Reactor Coolant System remains acceptable for continued operation, or

2) within 6 hours be in at least HOT SHUTDOWN, and within the next 30 hours reduce RCS temperature and pressure to less than 200°F and 500 psig, respectively.

3.1.2.2 The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator vessel is below 70°F.

3.1.2.3 The pressurizer heatup and cooldown rates shall not exceed 100°F/hr and 200°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

Basis: **Fracture Toughness Properties**

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the Summer 1965 Section III of the ASME Boiler and Pressure Vessel Code, Reference (1), and ASTM E185, Reference (2), and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1972 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code, Reference (3) and the calculation methods described in Reference (4). The results are reported in Reference (5) for Capsule T.

The heatup and cooldown curves are based on nominal pressure-temperature indications. Sufficient conservatism exists in the algorithm from which the curves were derived to account for instrument uncertainties.

TEXT DELETED

The temperature requirements for the steam generator corresponds with the measured NDT for the shell of the steam generator.

A temperature difference of 320°F between the pressurizer and reactor coolant system maintains stresses within the pressurizer spray nozzle below design limits.

(1) ASME Boiler and Pressure Vessel Code Section III (Summer 1965)

(2) ASTM E185 Surveillance Tests on Structural Materials in Nuclear Reactors

(3) ASME Boiler and Pressure Vessel Code, Section III, Summer 1972 Addenda (note Code Class 1514)

(4) Regulatory Guide 1.99, Rev. 2, May 1988, "Radiation Embrittlement of Reactor Vessel Materials."

(5) Westinghouse Report, "Rochester Gas and Electric Reactor Vessel Life Attainment Plan", dated March 1990.

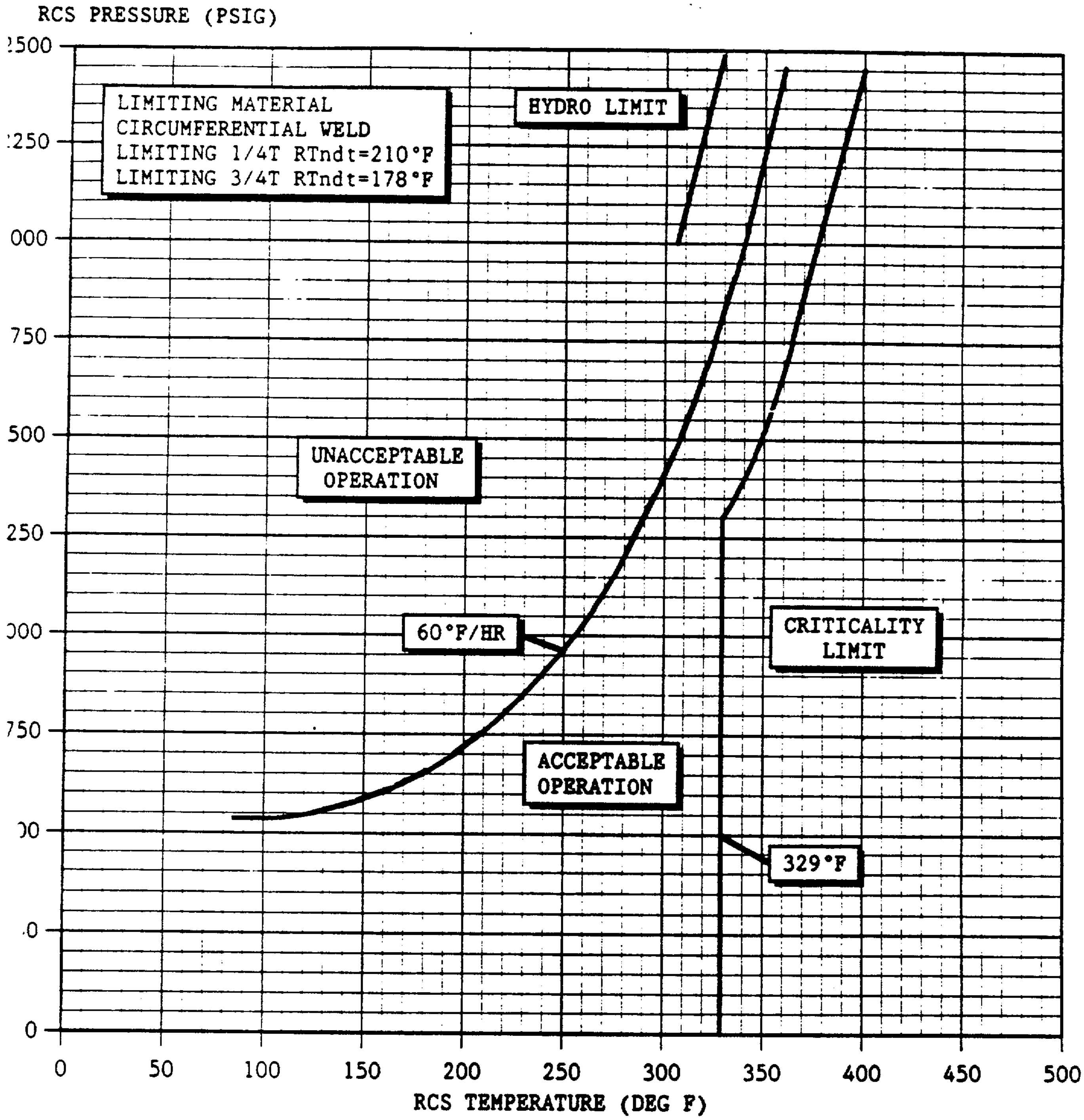

Figure 3.1-1: Ginna Reactor Vessel Heatup Limitations Applicable for the first 21 EFPY using Reg Guide 1.99, Rev. 2

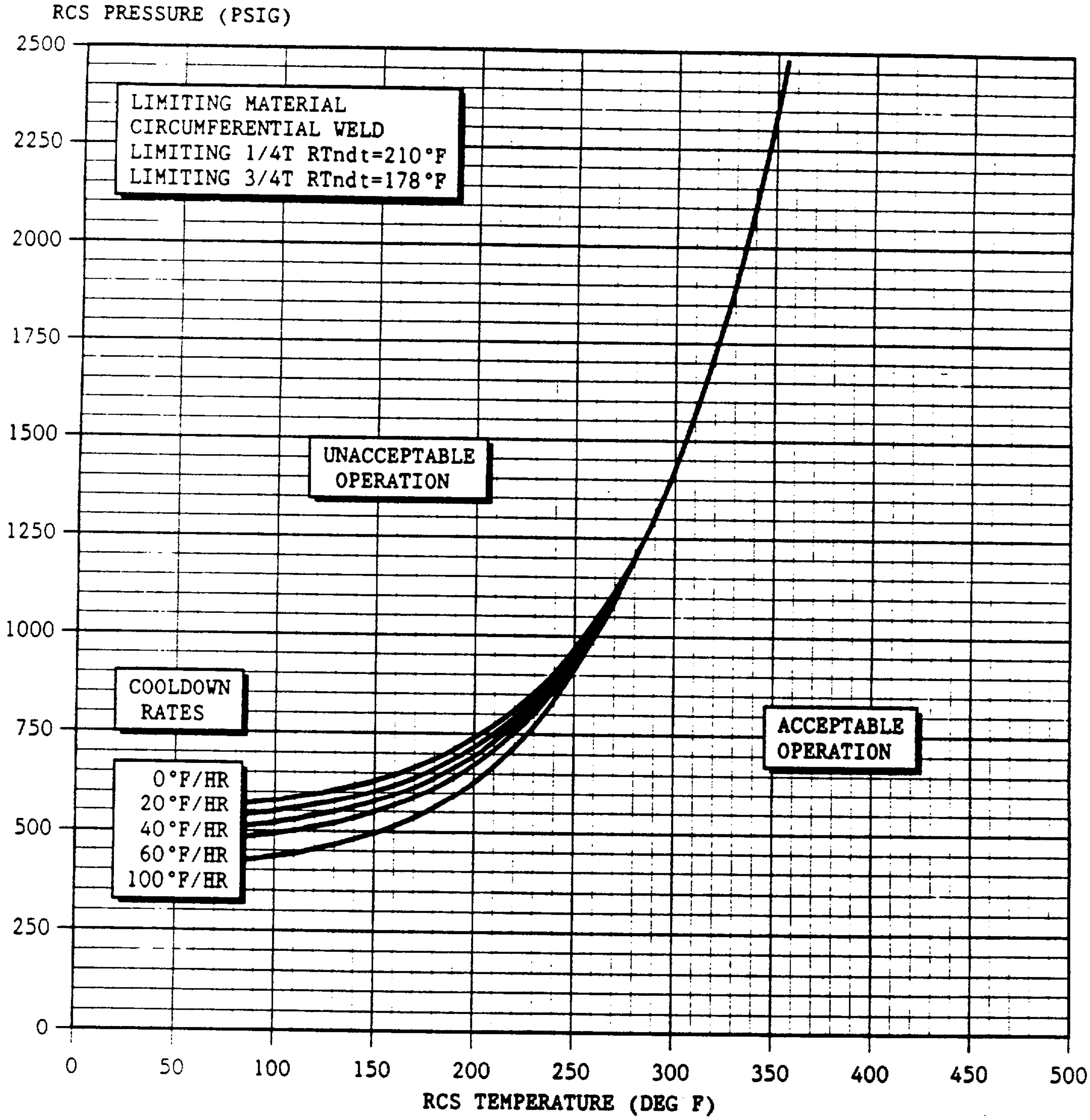

Figure 3.1-2: **Ginna Reactor Vessel Cooldown Limitations Applicable for the first 21 EFPY using Reg Guide 1.99, Rev. 2.**

3.3.1.7 Except during diesel generator load and safeguard sequence testing or when the vessel head is removed, or the steam generator primary system manway is open, no more than one safety injection pump shall be operable whenever the overpressurization protection is provided by a RCS vent of ≥ 1.1 sq. in. (3.15.1.b).

3.3.1.7.1 Whenever only one safety injection pump may be operable by 3.3.1.7, at least two of the three safety injection pumps shall be verified inoperable, as defined in the Basis for this section, a minimum of once per twelve hours.

3.3.1.8 Except during diesel generator load and safeguard sequence testing or when the vessel head is removed, or the steam generator primary system manway is open, all three safety injection pumps shall be inoperable and safety injection discharge paths to the RCS isolated whenever overpressure protection is provided by the pressurizer PORVs (3.15.1.a).

3.3.1.8.1 Whenever safety injection pumps are required to be inoperable by 3.3.1.8, the safety injection pumps shall be verified inoperable, as defined in the Basis of this section, a minimum of once per twelve hours. Similarly safety injection discharge paths to the RCS shall be verified to be isolated a minimum of once per twelve hours.

3.3.1.8.2 The requirements of 3.3.1.8 may be modified to allow operation of one SI pump provided the associated paths to the RCS are isolated by A.C. power being removed to the discharge MOVs in the closed position, or the manual isolation valves closed. Isolation of the discharge paths shall be verified at least once per 12 hours.

The facility has four service water pumps. Only one is needed during the injection phase, and two are required during the recirculation phase of a postulated loss-of-coolant accident.[8] The control room emergency air treatment system is designed to filter the control room atmosphere during periods when the control room is isolated and to maintain radiation levels in the control room at acceptable levels following the Design Basis Accident.[9] Reactor operation may continue for a limited time while repairs are being made to the air treatment system since it is unlikely that the system would be needed. Technical Specification 3.3.5 applies only to the equipment necessary to filter the control room atmosphere. Equipment necessary to initiate isolation of the control room is covered by another specfication.

The limits for the accumulator pressure and volume assure the required amount of water injection during an accident, and are based on values used for the accident analyses. The indicated level of 50% corresponds to 1108 cubic feet of water in the accumulator and the indicated level of 82% corresponds to 1134 cubic feet.

The limitation of no more than one safety injection pump to be operable when overpressure protection is being provided by a RCS vent of ≥ 1.1 sq. in. insures that the mass addition from the inadvertant operation of safety injection will not result in RHR system pressure exceeding design limits. The limitation on no safety injection pumps operable and the discharge lines isolated when overpressure protection is provided by the pressurizer PORV's removes mass injection from inadvertant safety injection as an event for which this configuration of overpressure protection must be designed to protect.

Inoperability of a safety injection pump may be verified from the main control board with the pump control switch in pull stop, or the pump breaker in the test or racked out position such that the pump could not start from an inadvertant safety injection signal. Isolation of a safety injection pump discharge path to the RCS may be verified from the main control board by the discharge MOV switch position indicating closed, or the discharge valve closed with A.C. power removed, or a manual discharge path isolation valve closed such that operation of the associated safety injection pump would not result in mass injection to the RCS.

The limitation on boric acid storage tank volume is based on the assumption that 2000 gallons of 12% to 13% solution is delivered to the RCS during a large steam line break associated with the recirculation flow to the RWST and the sweep volume in the SI pump suction line and still deliver 2000 gallons to the RCS.

References

(1) Deleted

(2) UFSAR Section 6.3.3.1

(3) UFSAR Section 6.2.2.1

(4) UFSAR Section 15.6.4.3

(5) UFSAR Section 9.2.2.4

(6) UFSAR Section 9.2.2.4

(7) Deleted

(8) UFSAR Section 9.2.1.2

(9) UFSAR Section 6.2.1.1 (Containment Integrity) and UFSAR Section 6.4 (CR Emergency Air Treatment)

(10) Westinghouse Analysis, "Report for the BAST Concentration for R. E. Ginna", August 1985 submitted by RG&E letter from R. W. Kober to H. R. Denton, dated October 16, 1985.

3.15 **Overpressure Protection System**

Applicability

Applies whenever the temperature of one or more of the RCS cold legs is ≤ 330°F, or the Residual Heat Removal System is in operation.

Objective

To prevent overpressurization of the reactor coolant system and the residual heat removal system.

Specification

3.15.1 Except during secondary side hydrostatic tests in which RCS pressure is to be raised above the PORV setpoint, at least one of the following over-pressure protection systems shall be operable:

a. Two pressurizer power operated relief valves (PORVs) with a lift setting of ≤ 424 psig, or

b. A reactor coolant system vent of ≥ 1.1 square inches.

3.15.1.1 With one PORV inoperable, either restore the inoperable PORV to operable status within 7 days or depressurize and vent the RCS through a 1.1 square inch vent(s) within the next 8 hours; maintain the RCS in a vented condition until both PORVs have been restored to operable status.

3.15.1.2 With both PORVs inoperable, depressurize and vent the RCS through a 1.1 square inch vent(s) within 8 hours; maintain the RCS in a vented condition until both PORVs have been restored to operable status.

3.15.1.3 Use of the overpressure protection system to mitigate an RCS or RHRS pressure transient shall be reported in accordance with 6.9.2.

Basis

An RCS vent opening of greater than 1.1 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are ≤ 330°F[1]. This relief capacity will ensure that no overpressurization of the RHR system could occur. The vent opening protects the RCS and RHRS from overpressurization when the transient is limited to either 1) the start of an idle RCP with the secondary water temperature of the steam generator ≤ 50°F above the RCS cold leg temperature or 2) the start of a safety injection pump and its injection into a water solid RCS[1,2].

The operability of two pressurizer PORVs ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are ≤ 330°F[2]. This relief capacity will also ensure that no overpressurization of the RHR system could occur. Either PORV has adequate relieving capability to protect the RCS and RHRS from overpressurization when the transient is limited to either 1) the start of an idle RCP with the secondary water temperature of the steam generator ≤ 50°F above the RCS cold leg temperature or 2) charging/letdown mismatch with three charging pumps in operation[3].

References:

(1) L. D. White, Jr., letter to A. Schwencer, NRC, dated July 29, 1977.

(2) SER for SEP Topics V-10.B, V-11.B, VII-3, "Safe Shutdown," dated September 29, 1981.

(3) Westinghouse Report, "R. E. Ginna Low Temperature Overpressure Protection System (LTOPS) Setpoint Phase II Evaluation Final Report," dated February 1991 submitted by letter to Allen R. Johnson, NRC, dated February 15, 1991.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED AMENDMENT NO. 48 TO FACILITY OPERATING LICENSE NO. DPR-18

ROCHESTER GAS AND ELECTRIC CORPORATION

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

INTRODUCTION

1.0 INTRODUCTION

By letter dated February 15, 1991, as supplemented on May 14, 1991, the Rochester Gas and Electric Corporation (the licensee) requested an amendment to Facility Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant. The proposed amendment would revise the pressure/temperature (P/T) limits in the Ginna Technical Specifications (TS), Section 3.1. The revised P/T limits have then re-evaluated the low temperature overpressurization protection system (LTOPS) set point and its associated basis in the Ginna Technical Specifications, Section 3.1.2, 3.3.1, and 3.15.1.

1.1 TS P/T Limits

The proposed P/T limits are valid for 21 effective full power years (EFPY) and were developed using Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Material." Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Effect on Plant Operations," recommends RG 1.99, Rev. 2, be used in calculating P/T limits, for the operation of the reactor coolant system during heatup, cooldown, criticality, and hydrotest.

To evaluate the P/T limits, the staff uses the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; and the ASTM Standards and the ASME Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); RG 1.99, Revision 2; Standard Review Plan (SRP) Section 5.3.2; and Generic Letter 88-11.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide Technical Specifications (TS) for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the TS. The P/T limits are among the limiting conditions of operation in the TS for all commercial nuclear plants in the U.S. Appendices G and H of 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits. An acceptable method for constructing the P/T limits is described in SRP Section 5.3.2.

Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature (RT_{ndt}). Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE). Generic Letter 88-11 requested that licensees and permittees use the methods in RG 1.99, Revision 2, to predict the effect of neutron irradiation on reactor vessel materials. RG 1.99, Revision 2 defines the ART as 1) the sum of unirradiated reference temperature; 2) the increase in reference temperature resulting from neutron irradiation, and 3) a margin to account for uncertainties in the prediction method.

Appendix H of 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline.

1.2 TS LTOP Setpoint

The Reactor Coolant System (RCS) Pressure-Temperature (P/T) limits during plant heatup and cooldown for R. E. Ginna Nuclear Power Plant are specified in Technical Specification (TS) Figures 3.1-1 and 3.1-2. The requirements for low temperature overpressure protection (LTOP) system are specified in TS 3.15. The restriction of the safety injection pump operability is specified in TS 3.3.1 in consistence with the assumption used in the analysis supporting LTOP setpoint.

By letter, dated February 15, 1991 (Ref. 1), Rochester Gas and Electric Corporation, in response to GL 88-11, proposed amendments to the P/T curves in TS Figures 3.1-1 (for heatup) and 3.1-2 (for cooldown), changes in the LTOP system setpoint and restriction of the safety injection pump operability specified in TS 3.3.1 and 3.15.1. The licensee also provided the results of a safety analysis to support its proposed TS changes.

2.0 EVALUATION

2.1 TS P/T Limits

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the Ginna reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Revision 2. The staff has determined that the material with the highest ART at 21 EFPY was the circumferential weld between the intermediate and lower shells (SA-847/WR-19) with 0.25% copper (Cu), 0.55% nickel (Ni), and an initial RT_{ndt} of 0°F.

The licensee has removed three surveillance capsules from Ginna. The results from capsules V, R, and T were published in Westinghouse reports FR-RA-1, WCAP-8421, and WCAP-10086, respectively. All surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline material, weld SA-847/WR-19, the staff calculated the ART to be 210°F at 1/4T (T = reactor vessel beltline thickness) and 178.0°F for 3/4T at 21 EFPY. The staff used a neutron fluence of 1.57E19 n/cm^2 at 1/4T and 7.20E18 n/cm^2 at 3/4T. The ART was determined by the least squares extrapolation method using the Ginna surveillance data. The least squares method is described in Section 2.1 of RG 1.99, Revision 2.

The licensee used the method in RG 1.99, Revision 2, to calculate an ART of 210°F at 21 EFPY at 1/4T for the same limiting weld metal. Substituting the ART of 210°F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on RT_{ndt} for the reactor vessel closure flange materials. Section IV.A.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the RT_{ndt} of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Based on the flange RT_{ndt} of 60°F, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

Section IV.B of Appendix G requires that the predicted Charpy USE at end of life be above 50 ft-lb. The staff has identified weld SA-847/WR-19, a Linde 80 weld, as the limiting material which has the lowest USE of all reactor vessel beltline materials. Based on data from surveillance capsule T and on RG 1.99, Revision 2, the staff has calculated USE at current EFPY (about 15.2) to be 54.2 ft-lb. The USE at 21 EFPY is calculated to be 52.2 ft-lb, and the End-of-Life (EOL) USE is expected to be 49.2 ft-lb. The licensee has also predicted that the USE of the limiting material at EOL to be below 50 ft-lb as shown in the Babcock & Wilcox (B&W) report, BAW-1803. The licensee is a member of the B & W Owners Group (B&WOG) established to study and resolve the low USE issue. The staff is following the progress of the B&WOG's study.

2.2 <u>TS LTOP Setpoint</u>

In the current TS, when the RCS cold leg temperature is less than or equal to 330°F, LTOP protection is provided by either PORVs with a lift setting of less than or equal to 435 psig, or a RCS vent of greater than or equal to 1.1 square inches. The operability of two PORVs or an RCS vent of greater than or equal to 1.1 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when the RCS cold leg temperature is less than or equal to 330°F. Each PORV has adequate relieving capability to protect the RCS against P/T limits when the transient is limited to either (1) the start of an idle RCS with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperature or (2) the inadvertent actuation of a safety injection pump and its injection into a water solid RCS. Consistent with the above assumption, the current TS 3.3.1 permits one operable safety injection pump when the RCS cold leg temperature is less than or equal to 330°F.

The licensee indicated in its letter, dated February 15, 1991, there is no safety relief valve at the suction side of the RHR system to protect the RHR system from potential overpressurization, thus the LTOP system also protects the RHR system from overpressurization when the RHR system is connected to the RCS. The licensee also determined that the allowable peak RCS pressure is more limiting for the RHR system protection than that for the protection against the limits of Appendix G to 10 CFR Part 50.

The licensee, in its letter, dated February 15, 1991 (Reference 3), proposed a change of the PORV setpoint from 435 psig to 424 psig. With this reduction of PORV setpoint, the licensee stated that the required LTOP system specified in Technical Specification 3.15.1 is adequate for protection against the limits of Appendix G to 10 CFR Part 50. The design basis transients of either energy addition or mass addition as described above apply. However, the licensee has determined that in order to protect the RHR system from overpressurization by the PORV the design basis mass addition transient needs to be changed to a charging/letdown mismatch with three charging pumps in operation. Consistent with this change, the licensee has proposed Technical Specification 3.3.1.8 to require that all three safety injection pumps shall be inoperable and safety injection discharge flow paths to the RCS isolated whenever overpressure protection is provided by the pressurizer PORV. Also, the licensee proposed that Technical Specification 3.3.1 allows no more than one safety injection pump to be operable when the overpressure protection is provided by a RCS vent of greater than or equal to 1.1 square inch. This is because the results of the licensee's recent analysis indicated that the mass addition from the inadvertent operation of a safety injection pump will not result in RHR system pressure exceeding allowable limits when overpressure protection is being provided by a RCS vent of greater than or equal to 1.1 square inch.

The staff has evaluated the licensee's proposed Technical Specification changes regarding LTOP setpoint and the analysis supporting these proposed changes. We have concluded that the proposed Technical Specifications are consistent with the assumptions used in their supporting analysis and therefore are acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (56 FR 33960). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

Based on the staff evaluation in Section 2.0 above, the staff concludes that the licensee's proposed Technical Specification changes, Sections 3.1.2, 3.3.1.7, 3.3.1.8, and 3.15.1, are acceptable.

The staff concludes that the proposed P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 21 EFPY because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The proposed P/T limits also satisfy Generic Letter 88-11 because the method in RG 1.99, Rev. 2 was used to calculate the ART. Hence, the proposed P/T limits may be incorporated into the Ginna Technical Specifications.

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 REFERENCES

1. Regulatory Guide 1.99, Radiation Embrittlement of Reactor Vessel Materials, Revision 2, May 1988.

2. NUREG-0800, Standard Review Plan, Section 5.3.2: Pressure-Temperature.

3. Letter from R. C. Mecredy, Rochester Gas and Electric Corporation, to USNRC, "R. E. Ginna Nuclear Power Plant, Modifications to Low Temperature Overpressure Protection Systems," February 15, 1991.

4. May 14, 1991, Letter from G. J. Wrobel (RG&E) to A. R. Johnson (USNRC), Subject: Additional Information on Ginna.

5. T. R. Mager et al., "Analysis of Capsule V from the Rochester Gas and Electric R. E. Ginna Unit No. 1 Reactor Vessel Radiation Surveillance Program," Westinghouse Nuclear Energy Systems, FP-RA-1, April 1, 1973.

6. S. E. Yanichko et al., "Analysis of Capsule R from the Rochester Gas and Electric R. E. Ginna Unit No. 1 Reactor Vessel Radiation Surveillance Program," Westinghouse Electric Corporation, WCAP-8421, November 1974.

7. S. E. Yanichko et al., "Analysis of Capsule T from the Rochester Gas and Electric R. E. Ginna Unit No. 1 Reactor Vessel Radiation Surveillance Program," Westinghouse Electric Corporation, WCAP-10086, April 1982.

Principal Contributors: John Tsao
C. Liang

Dated: March 6, 1992

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

November 19, 1991

MEMORANDUM FOR: Sholly Coordinator

FROM: Allen Johnson, Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II

SUBJECT: REQUEST FOR PUBLICATION IN BI-WEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE (TAC NO. M77515)

Rochester Gas and Electric Corporation, Docket No. 50-244, R. E. Ginna Nuclear Power Plant, Wayne County, New York

Date of application for amendment: June 29, 1990, as supplemented on June 20, 1991.

Brief description of amendment: The proposed amendment would revise Technical Specifications, Action Statements, Surveillance Requirements, and Basis Sections for the operability and testing requirements of the Ginna plant auxiliary electrical systems. The proposed amendment is a result of a station modification which incorporates additional availability of offsite electrical power from a second transmission source for the operation of plant auxiliaries. The supplemental information submitted on June 20, 1991 clarified information in the application. The information did not change the scope of the amendment request or the proposed determination of no significant hazards consideration.

Date of issuance: November 19, 1991

Effective date: November 19, 1991

Amendment No.: 47

Facility Operating License No. DRP-18: Amendment revised the Technical Specifications.

OP-1

DFX

Date of initial notice in FEDERAL REGISTER: October 17, 1990 (55 FR 42100)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 19, 1991.

No Significant hazards consideration comments received: No

Local Public Document Room location: Rochester Public Library, 115 South Avenue, Rochester, New York 14610.

Original signed by
Allen Johnson, Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II

DISTRIBUTION: Docket File, PDI-3 rf, WButler, AJohnson, OGC, MRushbrook

OFC	:PDI-3/LA	:PDI-3/PM	:OGC	:PDI-3/DIR	:
NAME	:MRushbrook	:AJohnson:mw	:CBarth	:WButler	:
DATE	:11/8/91	:11/8/91	:11/8/91	:11/19/91	:

OFFICIAL RECORD COPY
Document Name: SHOLLY GINNA TAC NO.77515