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.

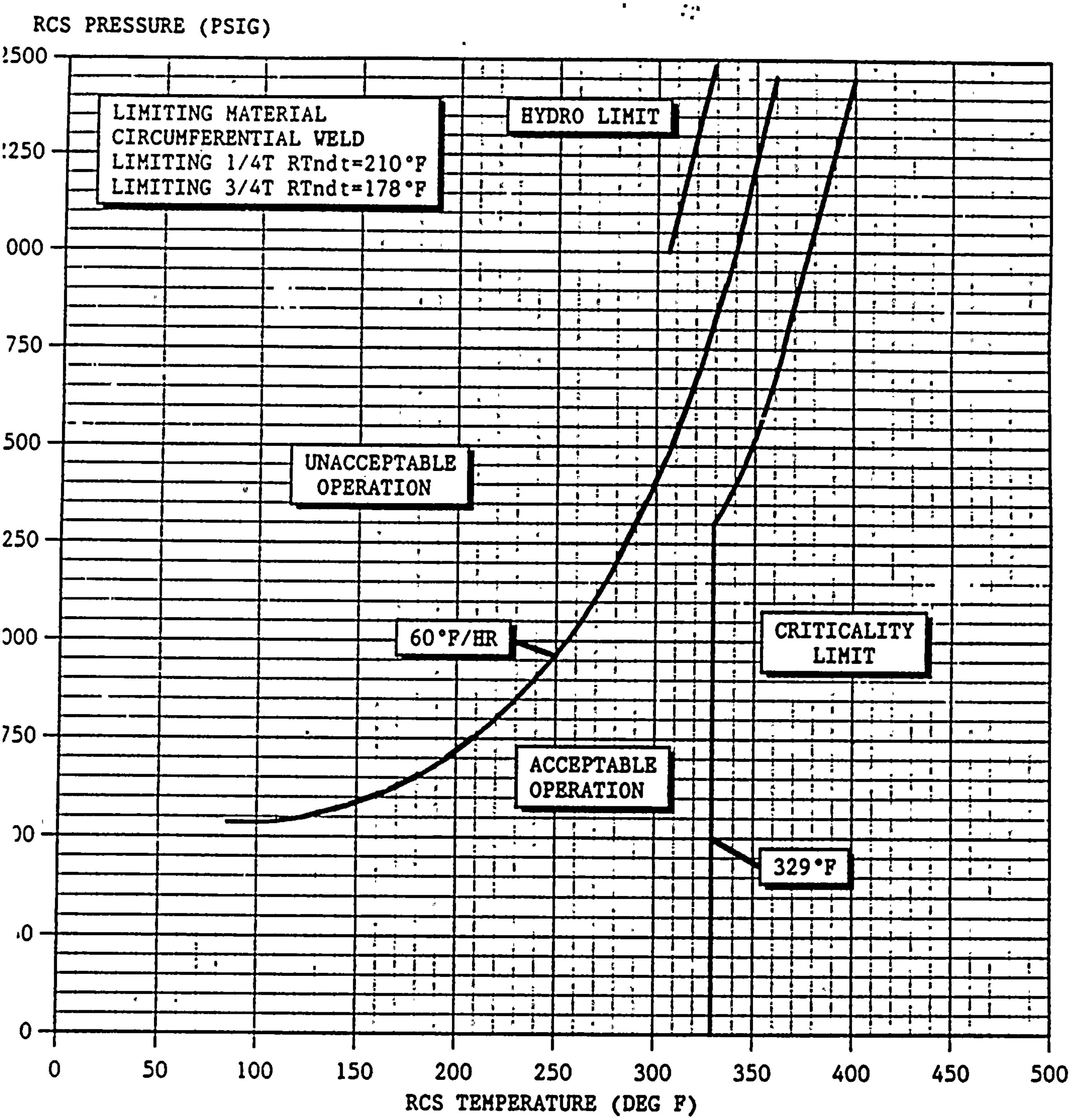

Figure 3.1-1: Ginna Reactor Vessel Heatup Limitations Applicable for the first 21 EFPY using Reg Guide 1.99, Rev. 2

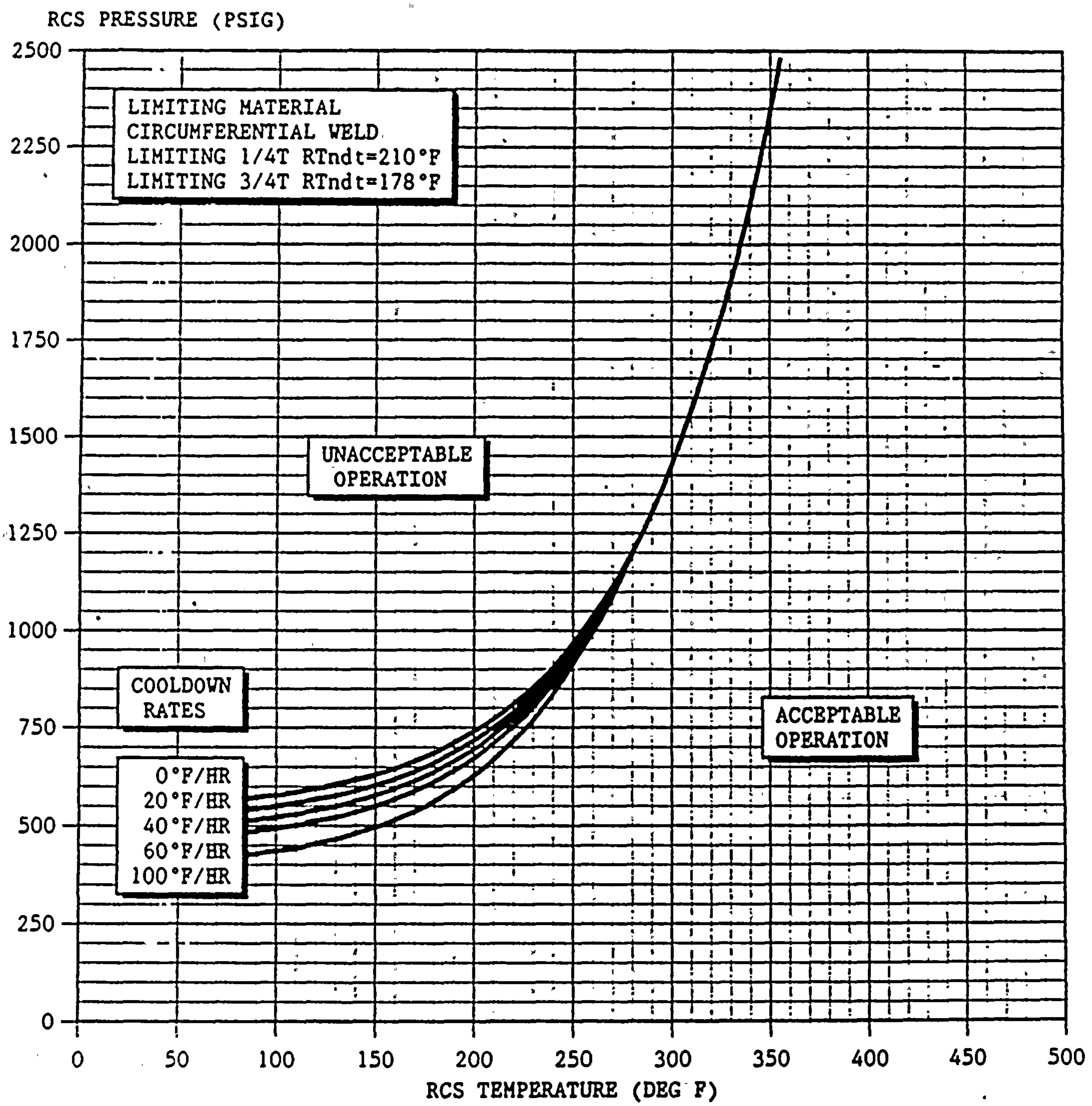

Figure 3.1-2: Ginna Reactor Vessel Cooldown Limitations Applicable for the first 21 EFPY using Reg Guide 1.99, Rev. 2.

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].

<u>References</u>:

(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.