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. 15
License No. DPR-18

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

   A. The application for amendment by Rochester Gas and Electric Corporation (the licensee) dated December 8, 1982 and supplemented October 10, 1983, August 8, 1984 and August 19, 1985 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;

   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:

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(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 15 , 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 its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Morton B. Fairtile

Morton B. Fairtile, Project Manager
Project Directorate #1
Division of PWR Licensing-A

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 12, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 15

FACILITY OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
3.1-10 to 3.1-16	3.1-10 to 3.1-16
--	3.1-16a
3.1-17 and 3.1-18	3.1-17 and 3.1-18
4.3-1	4.3-1
--	4.3-1a

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°F/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.

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).

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} at the end of 21.0 effective full power years (EFPY). The 21.0 EFPY period is chosen such that the limiting RT_{NDT} at the 1/4 T location in the core region is higher than the RT_{NDT} of the limiting unirradiated material. This service period assures that all components in the Reactor Coolant System will be operated conservatively in accordance with Code recommendations.

The highest RT_{NDT} of the core region material is determined by adding the radiation induced ΔRT_{NDT} for the applicable time period to the original RT_{NDT} shown in Reference (5). The fast neutron (E>1Mev) fluence at 1/4 thickness and 3/4 thickness vessel locations is given as a function of full power service life in Reference (5) and (6). Using the applicable fluence at the end of the 21.0 EFPY period for 1/4 thickness and the copper content of the material in question, the ΔRT_{NDT} is obtained from Reference (5). The ΔRT_{NDT} is more conservative than the value obtained from the third capsule of the radiation surveillance program.

Values of ΔRT_{NDT} determined in this manner will be used until more results from the material surveillance program, when evaluated according to ASTM E185, are available. The next capsule will be removed at approximately 17 EFPY (see Technical Specification 4.3.1). The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is greater than the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

<u>Heatup and Cooldown Limit Curves</u>

Allowable pressure temperature relationships for various heatup and cooldown rates are calculated using methods derived from Non-Mandatory Appendix G in Section III of the ASME Boiler and Pressure Vessel Code and discussed in detail in Reference (4). The approach specifies that the allowable total stress intensity factor (K_I) at any time during heatup or cooldown cannot be greater than that shown in the K_{IR}

curve for the metal temperature at that time. Furthermore, the approach applies explicit safety factors of 2.0 and 1.25* on stress intensity factors induced by pressure and thermal gradients, respectively. Thus, the governing equation for the heatup-cooldown analysis is:

(1) $2\ K_{Im} + 1.25\ K_{It} \leq K_{IR}$

where: K_{Im} is the stress intensity factor caused by membrane (pressure) stress.

K_{It} is the stress intensity factor caused by the thermal gradients.

K_{IR} is provided by the Code as a function of temperature relative to the RT_{NDT} of the material.

During the heatup analysis, Equation (1) is evaluated for two distinct situations.

First, allowable pressure-temperature relationships are developed for steady state (i.e., zero rate of

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* The 1.25 safety factor on K_{It} represents additional conservatism above Code requirements.

change of temperature) conditions assuming the presence of the code reference 1/4 T deep flaw at the ID of the pressure vessel. Due to the fact that during heatup the thermal gradients in the vessel wall tend to produce compressive stresses at the 1/4 T location, the tensile stresses induced by internal pressure are somewhat alleviated. Thus, a pressure-temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the 1/4 T location is treated as the governing factor.

The second portion of the heatup analysis concerns the calculation of pressure temperature limitations for the case in which the 3/4 T location becomes the controlling factor. Unlike the situation at the 1/4 T location, at the 3/4 T position (i.e., the tip of the 1/4 T deep OD flaw) the thermal gradients established during heatup produce stresses which are tensile in nature; and, thus, tend to reinforce the pressure stresses present. These thermal stresses are, of course, dependent on both the rate of heatup and the time (or water temperature) along the heatup ramp. Furthermore, since the thermal stresses at 3/4 T are tensile and increase with increasing heatup rate, a lower bound curve similar to that described in the preceding paragraph cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady state and finite heatup rate situations, the final limit curves are produced in the following fashion. First, a composite curve is constructed based on a point by point comparison of the steady state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the two values taken from the curves under construction. The composite curve is then adjusted to allow for possible errors in the pressure and temperature sensing instruments.

The use of the composite curve is mandatory in setting heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling analysis switches from the OD to the ID location; and the pressure limit must, at all times, be based on the most conservative case.

The cooldown analysis proceeds in the same fashion as that for heatup with the exception that the controlling location is always at the ID. The thermal gradients induced during cooldown tend to produce tensile stresses at the ID position and compressive stresses at the OD position. Thus, the ID flaw is clearly the worst case.

As in the case of heatup, allowable pressure-temperature relations are generated for both steady state and finite cooldown rate situations. Composite limit curves are then constructed for each cooldown rate of interest. Again adjustments are made to account for pressure and temperature instrumentation error.

The use of the composite curve in the cooldown analysis is necessary because system control is based on a measurement of reactor coolant temperature; whereas, the limiting pressure is calculated using the material temperature at the tip of the assumed reference flaw. During cooldown, the 1/4 T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition is, of course, not true for the steady state situation. It follows that the ΔT induced during cooldown results in a calculated higher K_{IR} for finite cooldown rates than for steady state under certain conditions.

Because operation control is on coolant temperature and cooldown rate may vary during the cooldown transient, the limit curves shown in Figure 3.1-2 represent a composite curve consisting of the more conservative values calculated for steady state and the specific cooling rate shown.

Details of these calculations are provided in Reference (4).

The temperature requirement 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 thermal 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) W.S. Hazelton, S.L. Anderson, and S.E. Yanichko, WCAP-7924, "Basis for Heatup and Cooldown Limit Curves"

(5) Analysis of Capsule T from the Rochester Gas and Electric Corporation R.E. Ginna Nuclear Plant Reactor Vessel Radiation Surveillance Program (WCAP-10086)

(6) Letter, R.W. Kober, RG&E to J.A. Zwolinski, USNRC, August 19, 1985

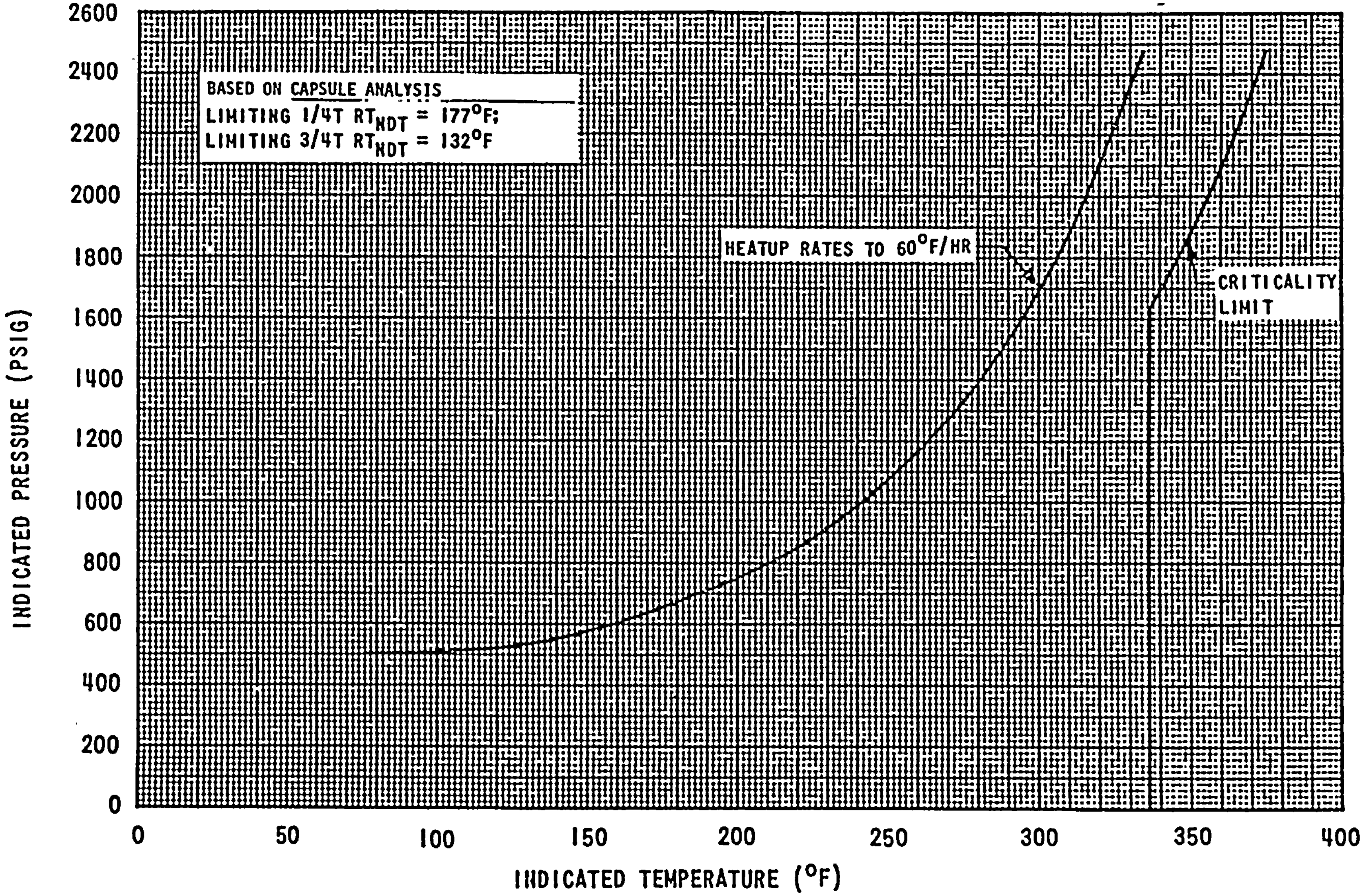

Figure 3.1-1 Reactor Coolant System Heatup Limitations Applicable for 21.0 Effective Full Power Years. $T_{ERROR} = 10^{o}F$, $P_{ERROR} = 60$ PSI

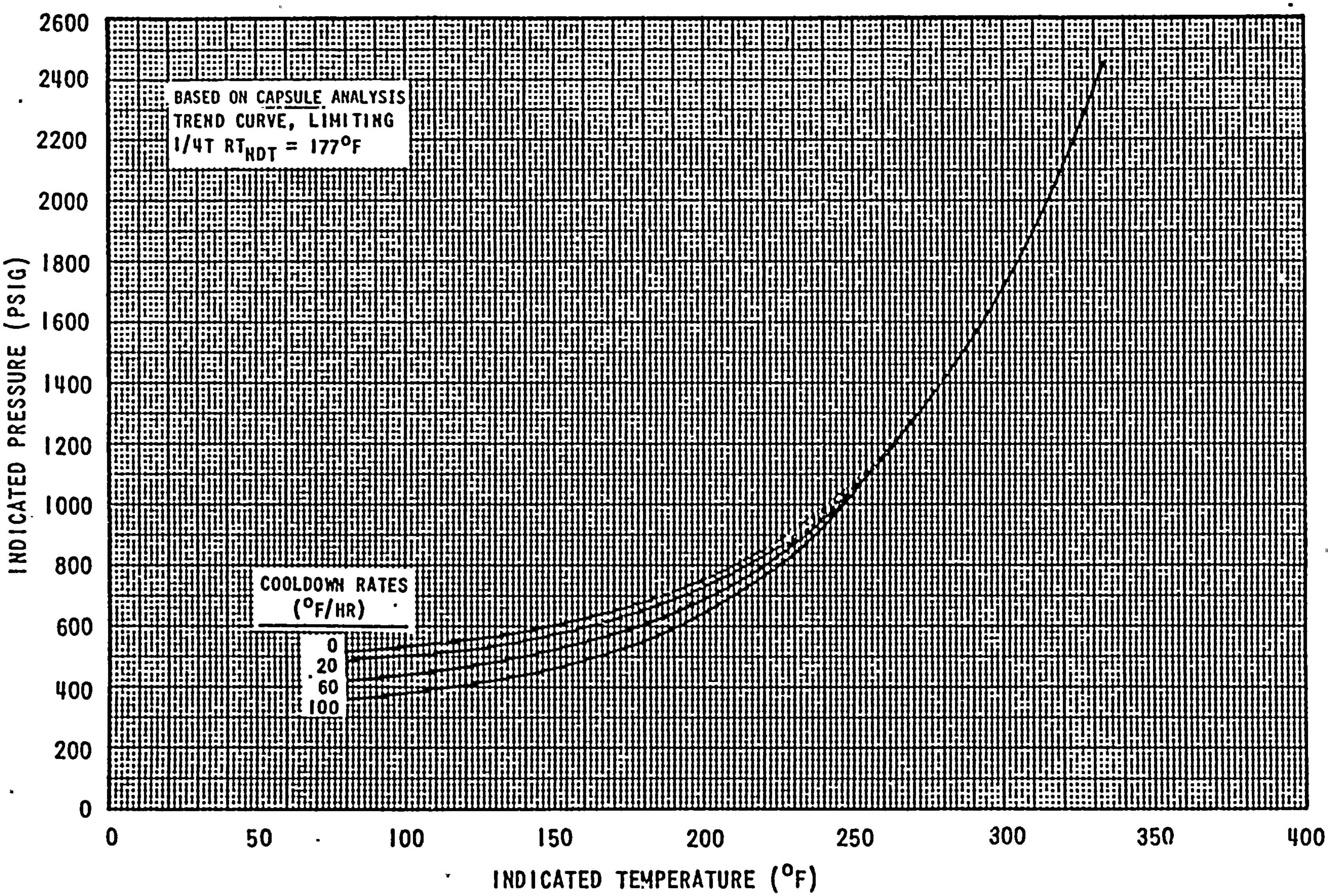

Figure 3.1-2 Reactor Coolant System Cooldown Limitations Applicable for 21.0 Effective Full Power Years. T_{ERROR} = 10°F, P_{ERROR} = 60 PSI

4.3 Reactor Coolant System

Applicability

Applies to surveillance of the reactor coolant system and its components.

Objective

To ensure operability of the reactor coolant system and its components.

Specifications:

4.3.1 Reactor Vessel Material Surveillance Testing

4.3.1.1 The reactor vessel material surveillance testing program is designed to meet the requirements of Appendix H to 10 CFR Part 50. This program consists of the metallurgical specimens receiving the following test: tensile, charpy impact and the WOL test. These tests of the Radiation Capsule Specimens shall be performed as follows:

Capsule	Time Removed For Testing
V	(Removed in 1971)
R	(Removed in 1974)
T	(Removed in 1980)
P	17 EFPY at nearest refueling
S	Standby
N	Standby

4.3.1.2 The report of the Reactor Vessel Material Surveillance shall be written as a Summary Technical Report as required by Appendix H to 10CFR Part 50.

4.3.2 Pressurizer

4.3.2.1 The pressurizer water level shall be verified to be within its limits at least once per 12 hours during power operation and hot shutdown.