April 7, 1977

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Docket No. 50-244

Rochester Gas & Electric Corporation
ATTN: Mr. Leon D. White, Jr.
Vice President
Electric & Steam Production
89 East Avenue
Rochester, New York 14604

Gentlemen:

The Commission has issued the enclosed Amendment No. 12 to Provisional Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant. This amendment consists of changes to the Technical Specifications in response to your application dated March 10, 1975, as supplemented April 8, 1976 and applications dated January 30 and August 10, 1976.

This amendment changes the Technical Specifications by revising the heatup and cooldown limit curves for the reactor vessel based on the results of tests and analyses performed on irradiated reactor pressure vessel material surveillance specimens contained in Capsule R which was withdrawn from the Ginna vessel. It also changes the organization charts in the Technical Specifications and moves the requirement for logging of control rod positions under certain conditions from column 6 to column 3 on page 3.5-4, Table 3.5-1.

Copies of the Safety Evaluation and Federal Register Notice are also enclosed.

Sincerely,

/s/

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 12 to DPR-18
2. Safety Evaluation
3. Federal Register Notice

OFFICE➤	DOR:ORB-1	DOR:ORB-1	OELD	DOR:ORB-1		
SURNAME➤	SSheppard	TVWambach:esp	A. Mitchell	ASchwencer		
DATE➤	3/7/77	3/7/77	4/5/77	4/7/77		

March 14, 1977

Note to Thomas V. Wambach

GINNA
SAFETY EVALUATION FOR O.L. AMENDMENT TO REVISE THE HEATUP AND COOLDOWN LIMIT CURVES IN TECHNICAL SPECIFICATIONS

This safety evaluation is inadequate because it does not provide enough information for the reasonable well informed lay person to determine whether the conclusions drawn in the document are properly supported. Background material on the origin and purpose of the tests should be provided. Is this one in a series of tests? What did the first one show? When is the next one expected? The reader is advised that the "heatup and cooldown pressure-temperature limitations for the reactor vessel were revised for 10.6 effective full power years ...". However, no information is provided on what the license currently provides in this regard nor the significance (if any) of the change. Inclusion of some of the introductory material in the November 1974 Westinghouse Report would help substantially.

I note also that this package includes a letter dated August 10, 1976 attaching a request to revise the requirement for control rod misalignment monitor operability. Am I correct in assuming this letter was inadvertently included and is not dealt with in this amendment?

Auburn L. Mitchell

Auburn L. Mitchell, Attorney
Office of the Executive Legal Director

Auburn:
I've added a "discussion" section to the SER to respond to your 1st paragraph and a section on Control Rod Misalignment Monitors to respond to your 2nd ¶.

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

April 7, 1977

Docket No. 50-244

Rochester Gas & Electric Corporation
ATTN: Mr. Leon D. White, Jr.
Vice President
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Rochester, New York 14604

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Division of Operating Reactors

Enclosures:
1. Amendment No. 12 to DPR-18
2. Safety Evaluation
3. Federal Register Notice

cc: Lex K. Larson, Esquire
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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 PROVISIONAL OPERATING LICENSE

Amendment No. 12
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 March 10, 1975, as supplemented April 8, 1976, and applications dated January 30 and August 10, 1976, comply 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;

   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 Appendices A and B, as revised through Amendment No. 12 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 7, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 12

PROVISIONAL OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

Revise Appendix A as follows:

Remove pages:

3.1-5
3.1-6
3.1-7
3.1-8
3.1-11
3.1-12 - Change of Figure 3.1-1
3.1-13 - Change of Figure 3.1-2
3.1-14 - Deletion of Figure 3.1-3
3.1-15 - Deletion of Figure 3.1-4
3.1-16 - Deletion of Table 3.1-1
3.1-19
3.5-4
4.3-1
6.2-2 - Change Figure 6.2-1
6.2-3 - Change Figure 6.2-2

Insert identically numbered pages.

3.1.2 <u>Heatup and Cooldown Limit Curves for Normal Operation</u>

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 10.6 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 200°F/hr. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

<u>Basis:</u> <u>Fracture Toughness Properties</u>

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 10.6 effective full power years (EFPY). The 10.6 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>1 Mev) fluence at 1/4 thickness and 3/4 thickness vessel locations is given as a function of full power service life in Reference (5). Using the applicable fluence at the end of the 10.6 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 second capsule of 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 early in the service life of the reactor vessel, note Radiation Surveillance Section in FSAR. 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.

Heatup and Cooldown Limit Curves

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

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).

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 Case 1514)

(4) W. S. Hazelton, S. L. Anderson, and S. E. Yanichko, WCAP-7924, "Basis for Heatup and Cooldown Limit Curves"

(5) WCAP-8421, "Analysis of Capsule R from the Rochester Gas and Electric Corporation R. E. Ginna Unit No. 1 Reactor Vessel Radiation Surveillance Program"

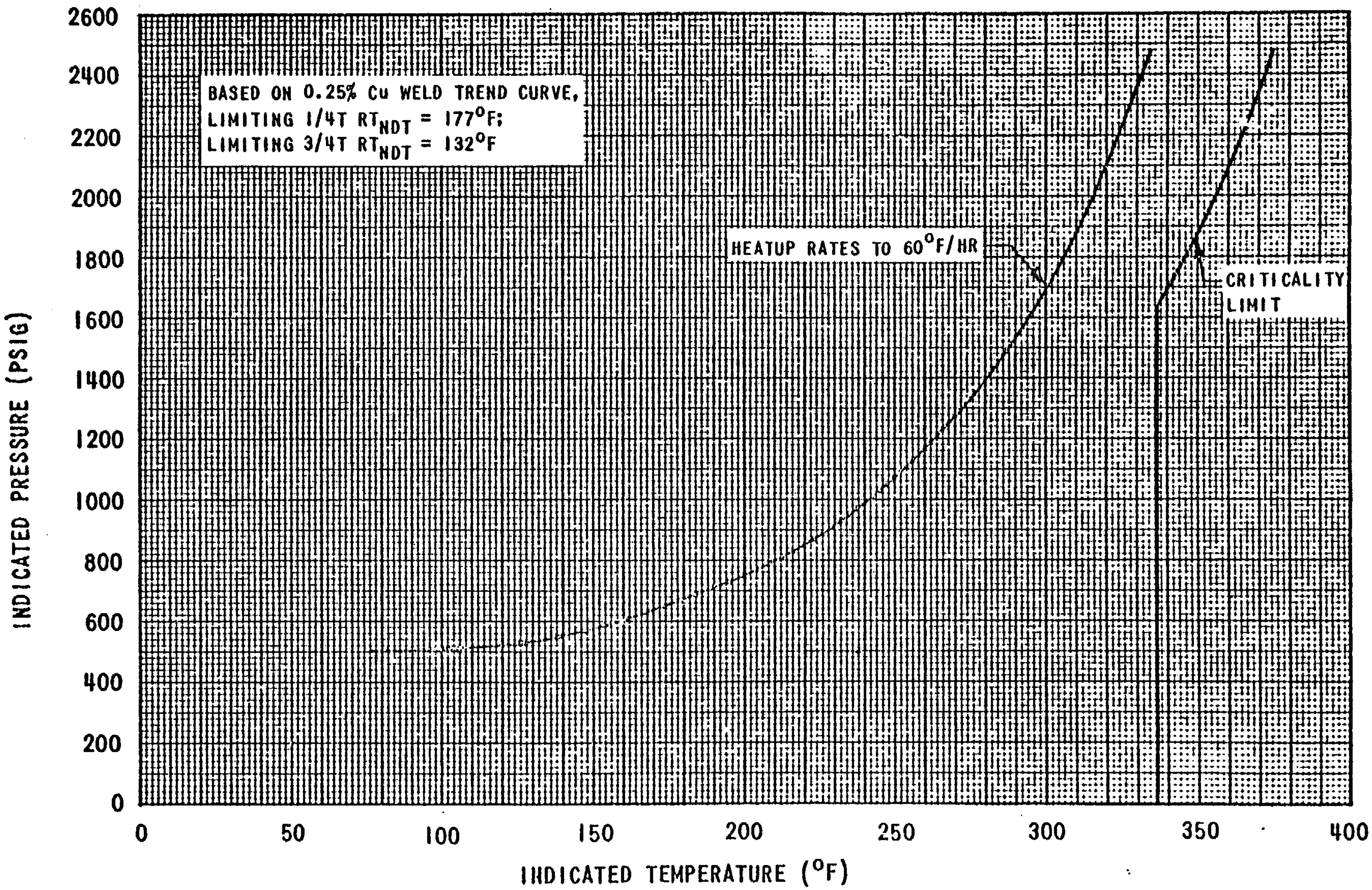

Figure 3.1-1 Reactor Coolant System Heatup Limitations Applicable for 10.6 Effective Full Power Years. $T_{ERROR} = 10^{o}F$, $P_{ERROR} = 60$ PSI

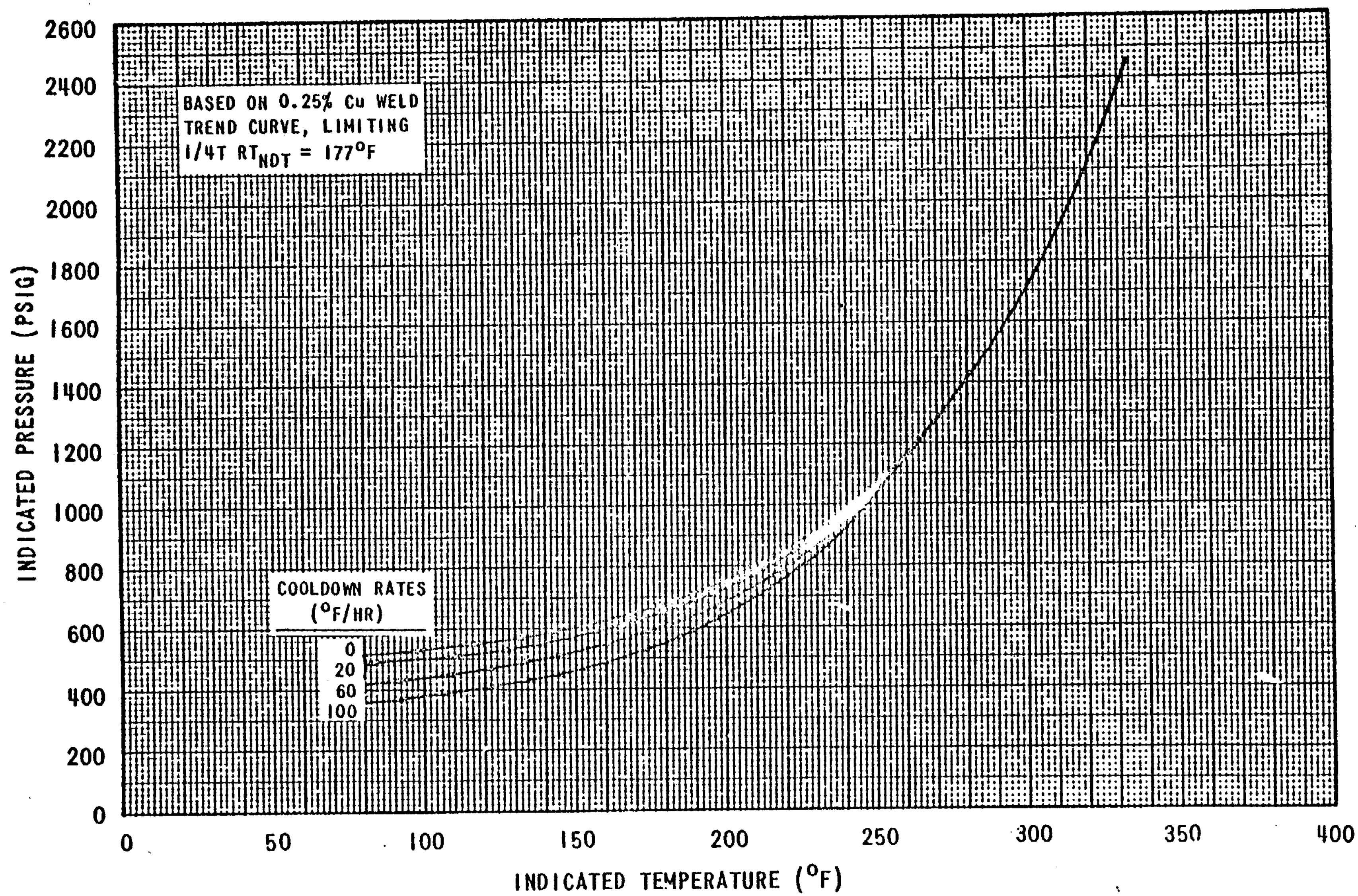

Figure 3.1-2 Reactor Coolant System Cooldown Limitations Applicable for 10.6 Effective Full Power Years. T_{ERROR} = 10°F, P_{ERROR} = 60 PSI

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The requirement that the reactor is not to be made critical above and to the left of the criticality limit provides increased assurance that the proper relationship between reactor coolant pressure and temperature will be maintained during system heatup and pressurization. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.

If the specified shutdown margin is maintained, there is no possibility of an accidental criticality as a result of an increase in moderator temperature or a decrease of coolant pressure.

References:

(1) FSAR Table 3.2.1-1

(2) FSAR Figure 3.2.1-8

(3) Amendment 14 to Connecticut Yankee License Application, Docket No. 50-213, March 2, 1967

(4) Southern California Edison Co. filing, Docket No. 50-206, October 19, 1967

(5) FSAR Figure 3.2.1-10

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MIN. OPERABLE CHANNELS	4 MIN. DEGREE OF REDUNDANCY	5 PERMISSABLE BYPASS CONDITIONS	6 OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 5 CANNOT BE MET
11.	Turbine Trip	3	2	2	1		Maintain 50% of rated power
12.	Steam Flow Feedwater flow mismatch with Lo Steam Generator Level	2/loop	1/loop	1/loop	1/loop		Maintain hot shutdown
13.	Lo Lo Steam Generator Water Level	3/loop	2/loop	2/loop	1/loop		Maintain hot shutdown
14.	Undervoltage 4 KV Bus	2/bus	1/bus	1/bus	-*		Maintain hot shutdown
15.	Underfrequency 4 KV Bus	2/bus	1/bus (both busses)	1/bus	-*		Maintain hot shutdown
16.	Control rod misalignment monitors**						
	a) Rod position deviation	1	-*	1 or Log individual rod positions once/hr, and after a load change of 10% or after 30 in. of control rod motion	-*		-**
	b) Quadrant power tilt monitor (upper & lower ex-core neutron detectors)	1	-**	1 or Log individual upper & Lower ion chamber currents once/hr & after a load change of 10% or after 30" of control rod motion	-*		-**

NOTE 1: When block condition exists, maintain normal operation.

F.P. = Full Power

* Not Applicable

** If both rod misalingment monitors (a and b) are inoperable for 2 hours or more, the nuclear overpower trip shall be reset to 93% of rated power in addition to the increased surveillance noted.

*** If a functional unit is operating with the minimum operable channels, the number of channels to trip the reactor

4.3.0 REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM

Applicability:

Applies to the tests of the metallurgical speciments taken from the reactor beltline region.

Objective:

To provide data for the determination of the fracture toughness of the reactor vessel.

Specification:

4.3.1 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 Tested
V	End of 1st core cycle
R	End of 3rd core cycle
T	10 years, at nearest refueling
P	20 years, at nearest refueling
S	30 years, at nearest refueling
N	Standby

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

ATTACHMENT A

R.E.GINNA NUCLEAR POWER PLANT
MANAGEMENT ORGANIZATION CHART

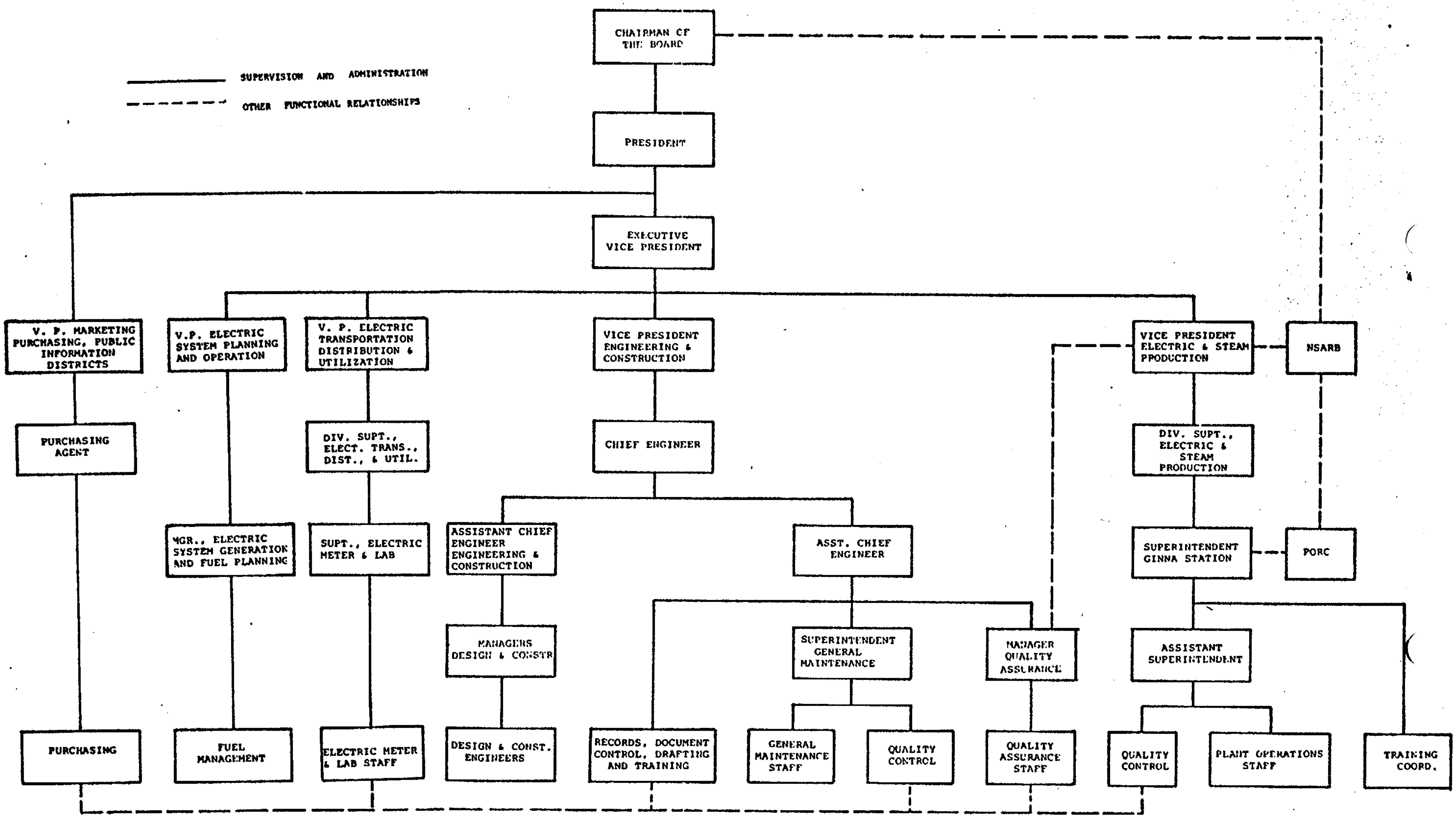

Figure 6.2-1

ATTACHMENT A

ROCHESTER GAS AND ELECTRIC CORPORATION
GINNA STATION ORGANIZATION

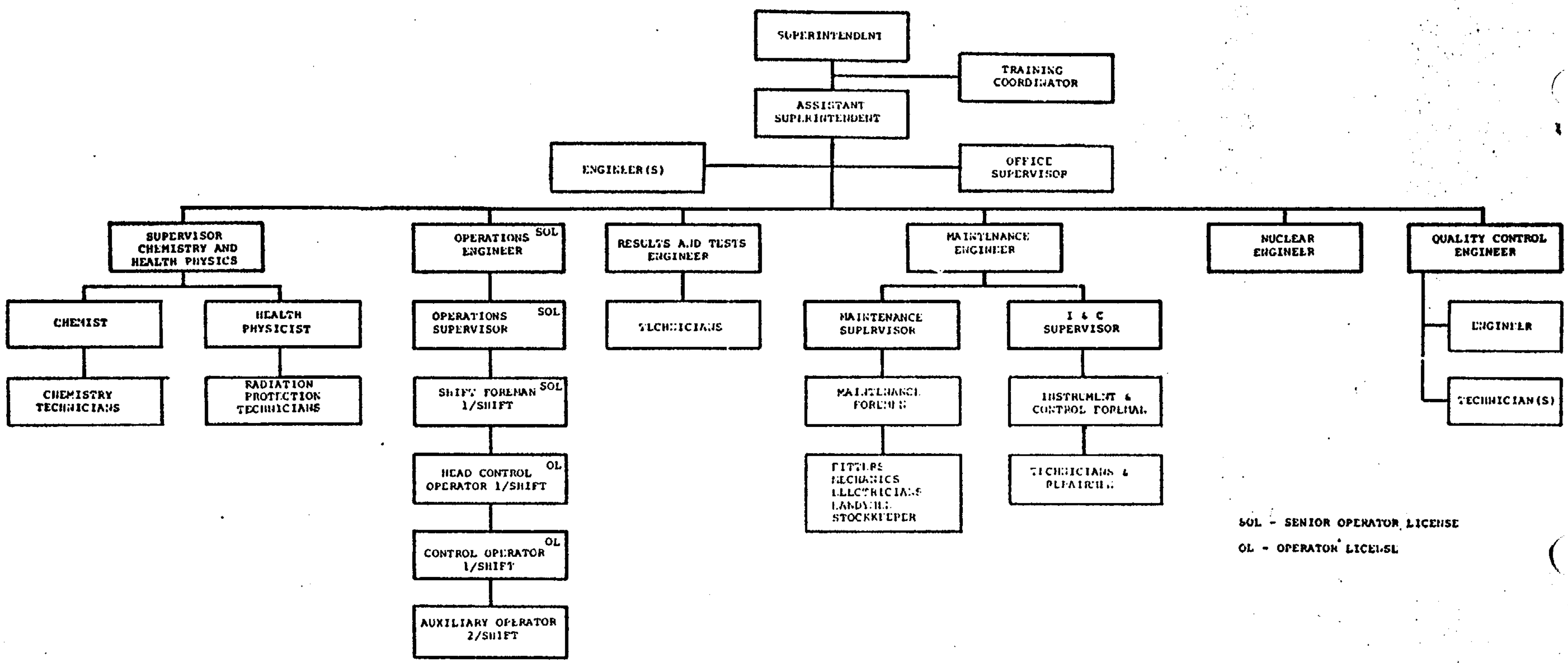

Figure 6.2-2

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 12 TO PROVISIONAL OPERATING LICENSE NO. DPR-18

ROCHESTER GAS AND ELECTRIC CORPORATION

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

Introduction

By application dated March 10, 1975, Rochester Gas and Electric Corporation (RG&E) applied for an amendment to Operating License No. DPR-18 for the Robert E. Ginna Nuclear Power Plant (Ginna) to revise the heatup and cooldown limit curves in the Technical Specifications. This change was based on the results of tests and analyses performed on irradiated reactor pressure vessel material specimens withdrawn from the Ginna vessel in Capsule R reported in WCAP-8421. This application was supplemented by submittal of the results of the Wedge Opening Loading (WOL) testing on April 8, 1976. By application dated January 30, 1976, RG&E has requested amendment of DPR-18 to revise Figures 6.2-1 and 6.2-2 of the Technical Specifications to reflect revisions to the station staff and station management organization. By application dated August 10, 1976, RG&E requested amendment of DPR-18 to remove a reporting requirement for inoperability of one of the two control rod misalignment monitors.

Discussion

Title 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements", requires that pressure-temperature limits be established for reactor coolant system heatup and cooldown operations, inservice leak and hydrostatic tests, and reactor core operation. These limits are required to ensure that the stresses in the reactor vessel remain within acceptable limits. They are intended to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences.

The specific pressure-temperature limits which are initially established depend upon the metallurgical properties of the reactor vessel material and the design service condition. However, the metallurgical properties vary over the lifetime of the reactor vessel because of the effects of neutron irradiation. One principal effect of the neutron irradiation is that it causes the reactor vessel nil ductility temperature (RT_{NDT})1/ to increase or shift with time. The practical results of the RT_{NDT} shift is that, for any given value of reactor pressure, the reactor vessel metal temperature must be maintained at higher values during the heatup and cooldown process. By periodically revising the pressure-temperature limits to account for neutron irradiation induced increases in RT_{NDT}, the stresses in the reactor vessel are maintained within acceptable limits.

The magnitude of the shift in RT_{NDT} is proportional to the integrated amount of neutron irradiation experienced by the reactor vessel. For Ginna the predicted RT_{NDT} shift is given in the Final Facility Description and Safety Analysis Report (FFDSAR). In addition a reactor vessel material surveillance program is established to check the validity of the predicted increases in RT_{NDT}. Surveillance specimens are periodically removed from the reactor vessel for testing and analysis. The results of the tests and analysis are compared with the predicted shifts in RT_{NDT}, then the pressure-temperature limits are revised accordingly.

Analysis of the first reactor vessel material surveillance specimens for Ginna were completed and the results were submitted by report dated May 23, 1973, and supplement dated November 12, 1973. On March 7, 1974, Change No. 11 to the Ginna Technical Specifications was issued which incorporated the change in the pressure-temperature limits resulting from the testing of those first specimens in Capsule V. Those limits were applicable for up to the first seven effective full power years (EFPY) operation. The Ginna reactor vessel has now experienced approximately 4 1/2 EFPY operation. The following evaluation addresses the revised pressure-temperature limits resulting from the testing of the second specimens removed from the Ginna vessel in Capsule R.

1/ RT_{NDT} is the temperature associated with the transition from ductile to brittle fracture mode of failure.

Evaluation

Reactor Vessel Heatup and Cooldown Limits

Capsule R was withdrawn from the Ginna reactor vessel after four years of reactor operation. Reactor pressure vessel material specimens in Capsule R had received a neutron fluence of 7.6 x 10^{18} n/cm^2 at an exposure temperature of less than 579°F. For this same time period the fluence at the inside reactor pressure vessel wall was 3.02 x 10^{18} n/cm^2. In addition to the neutron dosimeters and temperature monitors Capsule R contained (a) tensile and WOL specimens taken from the weldment and two heats of the base metal and (b) Charpy V-notch (C_V) specimens taken from the weldment heat affected zone (HAZ) and the two heats of base metal. For Ginna, the base metal in the reactor vessel beltline region is SA 508 Class 2. The residual element content for the two base metal heats was relatively low with 0.05% and 0.07% Cu and 0.01% P. The weldment had 0.23% Cu and 0.012% P.

As would be expected the weld metal sustained the greatest loss of toughness from neutron radiation with the average 50 ft-lb C_V temperature increased by 175°F and the average upper shelf impact energy decreased from 80 ft-lb to 50 ft-lb compared to the unirradiated weld metal. The average 50 ft-lb C_V temperature shifts for the two base metal heats were 0 and 25°F while the decreases in upper shelf energies were relatively small. The heat-up and cooldown pressure-temperature limitations for the reactor vessel were revised for 10.6 effective full power years based on the methods of Appendix G, Section III of the ASME Code using C_V data representative of the weld metal from Capsule R. In an effort to obtain dynamic fracture toughness data comparable to that used as the basis for the K_{IR} curve in Appendix G, Section III of the ASME Code, the WOL specimens from Capsule R were tested dynamically.

The fracture toughness test results and analysis performed on the surveillance specimens in Capsule R conform to the requirements of Appendix H, 10 CFR 50, "Reactor Vessel Material Surveillance Program Requirements" and ASTM Specification E 185-73, "Surveillance Tests for Nuclear Reactor Vessels." The heatup and cooldown pressure-temperature operating limitations proposed in the revised Technical Specifications for 10.6 EFPY have been determined in accordance with Appendix G, Section III of the ASME Code, "Protection Against Nonductile Failure" and ensure a margin of safety during operation consistent with that required by Appendix G, 10 CFR 50, "Fracture Toughness Requirements".

The heatup and cooldown pressure-temperature operating limitations for 10.6 EFPY proposed by RG&E are acceptable.

Revision of Organization Charts

The Organization Charts in the Technical Specifications have been expanded to show the different groups within the corporate divisions associated with the operation of Ginna and to show more detail in the plant staff organization in the areas of technical and maintenance personnel. This change is purely administrative in nature, does not degrade RG&E's management control and is acceptable.

Control Rod Misalignment Monitors

The proposed change will not change any action which is to be taken if either or both control rod misalignment monitors are inoperable. If one of the monitors is inoperable, hand logging at the intervals specified in Table 3.5-1 will commence. If both monitors are inoperable, the reduction in overpower trip and the hand logging specified in Table 3.5-1 must be performed. Thus, no change in the type or frequency of monitoring is proposed.

The proposed change would provide that if one of the two monitors is inoperable and hand logging as specified is performed, the occurrence is not interpreted as being reportable pursuant to Technical Specification 6.9.2.b. No change in reporting requirements is proposed for the condition in which both monitors are inoperable.

If one monitor is inoperable and hand logging is performed, it is not considered that plant operation is in a degraded mode. This is due to the fact that one continuous monitor is continuing to operate and based on the assurance provided by the hand logging. This logging will identify slow changes caused by, for example, core burnup with the accompanying slow change in control rod position. The logging after a power change of 10% or more and after control rod movement of 30 inches is required since the possibility of rod misalignment is greater if significant power or rod position changes are occuring than if only those changes due to core burnup or small power changes are occuring. The hand logging requirement provides assurance that rod misalignment does not occur as a result of the power change in excess of 10% or control bank movement in excess of 30 inches. Thus, operability of one monitor in addition to the specified hand logging provides adequate assurance that control rods are not misaligned such that the plant need not be deemed to be in a degraded mode and this condition, therefore, does not fall under the reporting requirements of Specification 6.9.2.b.

The change moves the hand logging requirement from column 6 to column 3 of Table 3.5-1 and is acceptable.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: April 7,1977

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-244

ROCHESTER GAS AND ELECTRIC CORPORATION

NOTICE OF ISSUANCE OF AMENDMENT TO PROVISIONAL OPERATING LICENSE

The Nuclear Regulatory Commission (the Commission) has issued Amendment No. 12 to Provisional Operating License No. DPR-18, issued to Rochester Gas and Electric Corporation, which revised Technical Specifications for operation of the R. E. Ginna Nuclear Power Plant located in Wayne County, New York. The amendment is effective as of its date of issuance.

This amendment changes the Technical Specifications by revising the heatup and cooldown limit curves for the reactor vessel based on the results of tests and analyses performed on irradiated reactor pressure vessel material surveillance specimens contained in Capsule R which was withdrawn from the Ginna vessel. It also changes the organization chart in the Technical Specifications and removes a reporting requirement for inoperability of one of the two control rod misalignment monitors.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated March 10, 1975, as supplemented April 8, 1976, and applications dated January 30, and August 10, 1976, (2) Amendment No. 12 to Provisional License No. DPR-18 and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the Lyons Public Library, 67 Canal Street, Lyons, New York 14489 and at the Rochester Public Library, 115 South Avenue, Rochester, New York 14627. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 7th day of April 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

PRELIMINARY DETERMINATION

NOTICING OF PROPOSED LICENSING AMENDMENT

Licensee: Rochester Gas and Electric Corporation - R. E. Ginna

Request for: Technical Specification Change Request - To update the reactor vessel heatup and cooldown curves in conformance with Appendix G incorporating the results of the second surveillance capsule test results.

Request Date: March 10, 1975

Proposed Action:
( ) Pre-notice Recommended
(X) Post-notice Recommended
( ) Determination delayed pending completion of Safety Evaluation

Basis for Decision: This change involves the periodic updating of reactor vessel heatup and cooldown limitations as required by 10 CFR 50 Appendix G and by the existing technical specifications. The proposed limitations are more restrictive than those in the present specifications. This is normal and results from the radiation induced effects on the properties of the vessel material as measured by testing of the surveillance capsules. Conformance with Appendix G ensures vessel integrity for plant operation. These limits

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CONCURRENCES:		DATE:
1.	T. V. Wambach — T. V. Wambach	3/19/75
2.	RAPurple — R. A. Purple	3/19/75
3.	Karl R. Goller — K. R. Goller	3/19/75
4.	Edward G. Ketchen, Jr. — Office of Executive Legal Director	3/26/75