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VISSING,G.S.

SUBJECT: Provides revised pressure & temperature limits rept to complete all outstanding util commitments.

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ROBERT C. MECREDY
Vice President
Nuclear Operations

December 30, 1996

U.S. Nuclear Regulatory Commission
Document Control Desk
Attn: Guy S. Vissing
Project Directorate I-1
Washington, D.C. 20555

Subject: Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)
Rochester Gas & Electric Corporation
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

References:

(a) Letter from R.C. Mecredy, RG&E, to G.S. Vissing, NRC, *Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)*, dated December 2, 1996.

(b) Letter from G.S. Vissing, NRC, to R.C. Mecredy, RG&E, *R.E. Ginna Ginna - Acceptance of Request to Extend Time for Approval of Revision of Pressure and Temperature Limits Report (PTLR) (TAC No. M97313)*, dated December 10, 1996.

(c) Letter from R.C. Mecredy, RG&E, to G.S. Vissing, NRC, *Request to Use ASME Code Case N-514 in the Determination of Low Temperature Overpressure Protection (LTOP)*, dated December 18, 1996.

(d) Letter from R.C. Mecredy, RG&E, to G.S. Vissing, NRC, *Application for Amendment to Facility Operating License, Revision to Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)*, dated September 13, 1996.

Dear Mr. Vissing:

By Reference (a), RG&E requested an extension to the NRC's acceptance of the current Ginna Station PTLR from December 31, 1996 to July 1, 1997, for reasons as stated in the letter. The NRC approved this request in Reference (b) provided that RG&E submit the following by December 31, 1996:

a. Request to allow use of ASME Code Case N-514; and

b. Revised PTLR incorporating all NRC comments received to date.

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The request to use ASME Code Case N-514 was provided by Reference (c). Therefore, the purpose of this letter is to provide the revised PTLR to complete all outstanding RG&E commitments. The revised PTLR is attached. There is only one change to the PTLR since that provided in Reference (d). This change revises the 3/4-T ART value listed in Figures 1 and 2 and Table 6 to 196.9°F from 196°F. This is due to a round-off error and does not affect the heatup/cooldown curves. All other sections of the PTLR remain unchanged. The cover sheet and pages 6, 7, and 12 are denoted as Rev. 2, 12/96. All other pages are identical to the September 13, 1996 submittal.

Please contact George Wrobel, Manager of Nuclear Safety and Licensing at (716) 724-8070 if you have further questions.

Very truly yours,

Robert C. Mecredy

MDF\898

xc: U.S. Nuclear Regulatory Commission
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Ginna Senior Resident Inspector

RG&E	GINNA STATION	PTLR Revision 2, 12/96

RCS PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

____
Responsible Manager

____
Effective Date

Controlled Copy No. _____

R.E. Ginna Nuclear Power Plant

RCS Pressure and Temperature Limits Report

Revision 2

Note: This report is not part of the Technical Specifications. This report is referenced in the Technical Specifications.

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1.0 RCS Pressure and Temperature Limits Report (PTLR)

This Pressure and Temperature Limits Report (PTLR) for Ginna Station has been prepared in accordance with the requirements of Technical Specification 5.6.6. Revisions to the PTLR shall be provided to the NRC after issuance.

The Technical Specifications addressed in this report are listed below:

3.4.3	RCS Pressure and Temperature (P/T) Limits
3.4.6	RCS Loops - MODE 4
3.4.7	RCS Loops - MODE 5, Loops Filled
3.4.10	Pressurizer Safety Valves
3.4.12	Low Temperature Overpressure Protection (LTOP) System

2.0 OPERATING LIMITS

The cycle-specific parameter limits for the specifications listed in Section 1.0 are presented in the following subsections. All changes to these limits must be developed using the NRC approved methodologies specified in Technical Specification 5.6.6. These limits have been determined such that all applicable limits of the safety analysis are met. All items that appear in capitalized type are defined in Technical Specification 1.1, "Definitions."

2.1 RCS Pressure and Temperature Limits (LCO 3.4.3 and LCO 3.4.12) (Reference 1)

2.1.1 The RCS temperature rate-of-change limits are:

a. A maximum heatup of 60°F per hour.

b. A maximum cooldown of 100°F per hour.

2.1.2 The RCS P/T limits for heatup and cooldown are specified by Figures 1 and 2, respectively.

2.1.3 The minimum boltup temperature, using the methodology of Reference 2, Section 2.7, is 60°F.

2.2 Low Temperature Overpressure Protection System Enable Temperature (LCOs 3.4.6, 3.4.7, 3.4.10 and 3.4.12) (Methodology of Reference 3, Attachment II, Section 3.4 using 1/4T RT_{NDT} value from Reference 1).

2.2.1 The enable temperature for the Low Temperature Overpressure Protection System is 322°F.

2.3 Low Temperature Overpressure Protection System Setpoints (LCO 3.4.12)

2.3.1 Pressurizer Power Operated Relief Valve Lift Setting Limits (Methodology of Reference 3, Attachment II as calculated in Reference 4, Attachment IV)

The lift setting for the pressurizer Power Operated Relief Valves (PORVs) is ≤ 411 psig (includes instrument uncertainty).

3.0 REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM

The reactor vessel material irradiation surveillance specimens shall be removed and examined to determine changes in material properties. The removal schedule is provided in Table 1. The results of these examinations shall be used to update Figures 1 and 2.

The pressure vessel steel surveillance program (Ref. 5) is in compliance with Appendix H to 10 CFR 50, entitled, "Reactor Vessel Radiation Surveillance Program." The material test requirements and the acceptance standard utilize the reference nil-ductility temperature, RT_{NDT}, which is determined in accordance with ASTM E208. The empirical relationship between RT_{NDT} and the fracture toughness of the reactor vessel steel is developed in accordance with Appendix G, "Protection Against Non-Ductile Failure," to section III of the ASME Boiler and Pressure Vessel Code. The surveillance capsule removal schedule meets the requirements of ASTM E185-82.

As shown by Reference 1 (specifically its Reference 51), the reactor vessel material irradiation surveillance specimens indicate that the surveillance data meets the credibility discussion presented in Regulatory Guide 1.99 revision 2 where:

1. The capsule materials represent the limiting reactor vessel material.

2. Charpy energy vs. temperature plots scatter are small enough to permit determination of 30 ft-lb temperature and upper shelf energy unambiguously.

3. The scatter of ΔRT_{NDT} values are within the best fit scatter limits as shown on Table 2. The only exception is with respect to the Intermediate Shell which is not the limiting reactor vessel material.

4. The Charpy specimen irradiation temperature matches the reactor vessel surface interface temperature within ± 25°F.

5. The surveillance data falls within the scatter band of the material database.

4.0 SUPPLEMENTAL DATA INFORMATION AND DATA TABLES

4.1 The RT_{PTS} value for Ginna Station limiting beltline material is 259.1°F for 32 EFPY per Reference 1.

4.2 Tables

Table 2 contains a comparison of measured surveillance material 30 ft-lb transition temperature shifts and upper shelf energy decreases with Regulatory Guide 1.99, Revision 2 predictions.

Table 3 shows calculations of the surveillance material chemistry factors using surveillance capsule data.

Table 4 provides the reactor vessel toughness data.

Table 5 provides a summary of the fluence values used in the generation of the heatup and cooldown limit curves.

Table 6 shows example calculations of the ART values at 24 EFPY for the limiting reactor vessel material.

5.0 REFERENCES

1. WCAP-14684, "R.E. Ginna Heatup and Cooldown Limit Curves for Normal Operation," dated June 1996.

2. WCAP-14040, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," Revision 2, January 1996.

3. Letter from R.C. Mecredy, RG&E, to A.R. Johnson, NRC, Subject: "Technical Specification Improvement Program, Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," dated December 8, 1995.

4. Letter from R.C. Mecredy, RG&E, to A.R. Johnson, NRC, Subject: "Application for Amendment to Facility Operating License, Methodology for Low Temperature Overpressure Protection (LTOP) Limits," dated February 9, 1996.

5. WCAP-7254, "Rochester Gas and Electric, Robert E. Ginna Unit No. 1 Reactor Vessel Radiation Surveillance Program," May 1969.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: CIRCUMFERENTIAL WELD SA-847
LIMITING ART VALUES AT 24 EFPY: 1/4T, 232°F
3/4T, 196.9°F

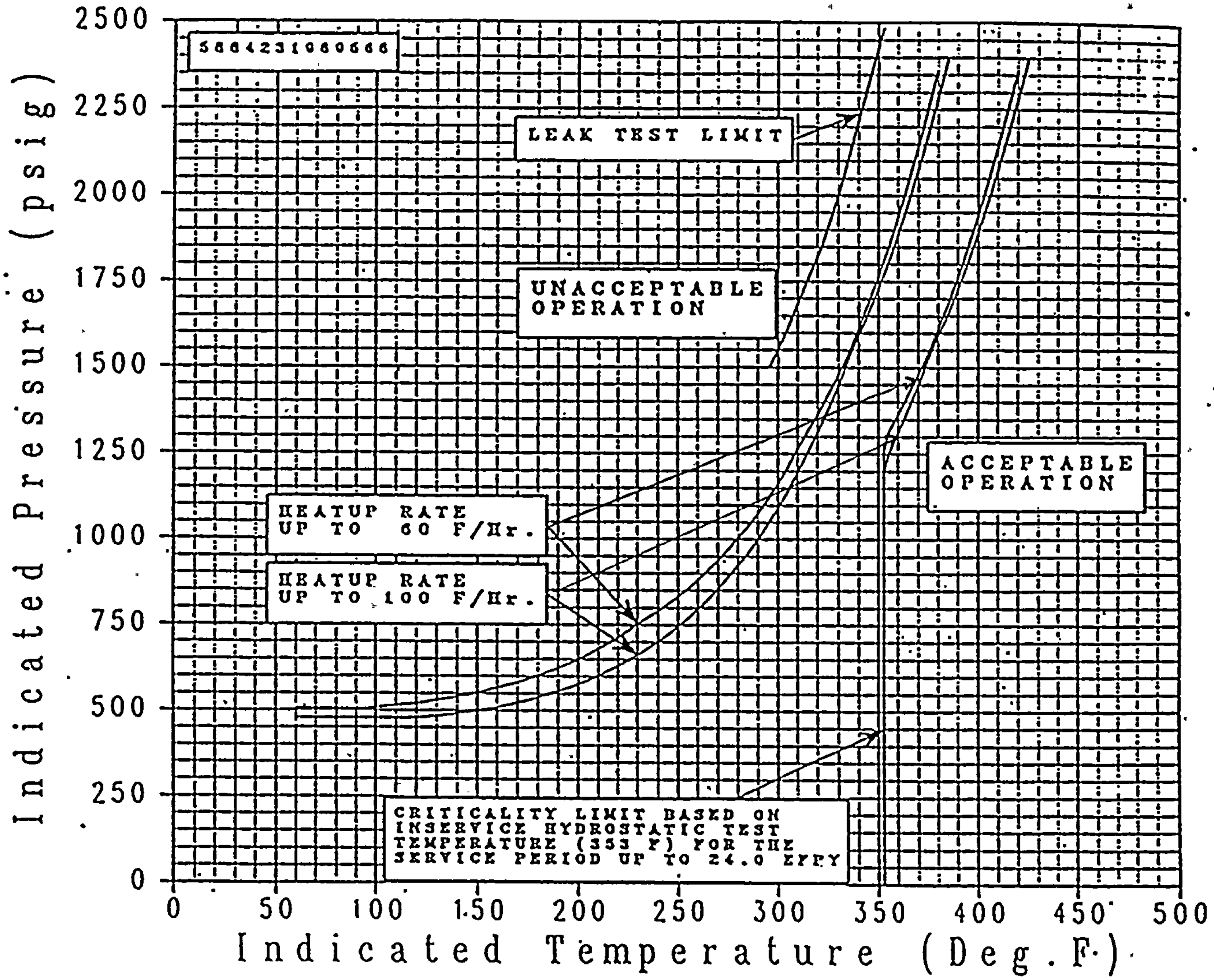

FIGURE 1

REACTOR VESSEL HEATUP LIMITATIONS
APPLICABLE FOR THE FIRST 24 EFPY

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: CIRCUMFERENTIAL WELD SA-847
LIMITING ART VALUES AT 24 EFPY: 1/4T, 232°F
3/4T, 196.9°F

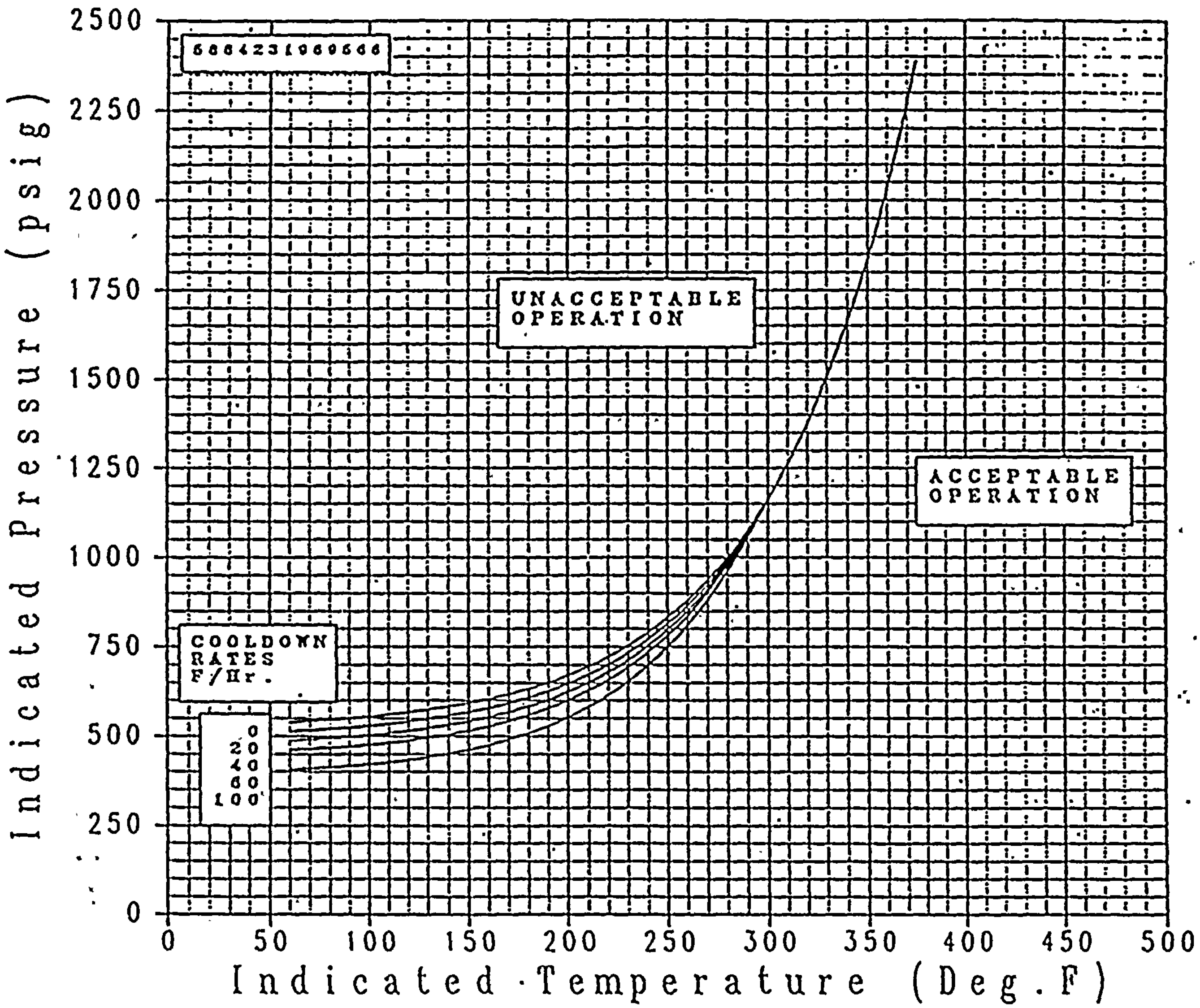

FIGURE 2

REACTOR VESSEL COOLDOWN LIMITATIONS
APPLICABLE FOR THE FIRST 24 EFPY

Table 1

Surveillance Capsule Removal Schedule

Capsule	Vessel Location (deg.)	Capsule Lead Factor	Removal Schedule[a]	Capsule Fluence E19(n/cm^2)[c]
V	77°	2.99	1.6 (removed)	.5028
R	257°	3.00	2.7 (removed)	1.105
T	67°	1.85	7 (removed)	1.864
S	57°	1.74	17 (removed)	3.746
N	237°	1.74	TBD[b]	TBD[b]
P	247°	1.9	Standby	N/A

NOTES:

(a) Effective Full Power Years (EFPY).

(b) To be determined, there is no current requirement for removal.

(c) Reference 1.

TABLE 2 Surveillance Material 30 ft-lb Transition Temperature Shift					
Material	Capsule	Fluence (x 10^{19} n/cm^2, E > 1.0 MeV)[a]	30 lb-ft Transition Temperature Shift		
			Predicted[a] (°F)	Measured[a] (°F)	Δ (°F)
Lower Shell	V	.5028	26	25	1
	R	1.105	32	25	7
	T	1.864	37	30	7
	S	3.746	42	42	0
Intermediate Shell	V	.5028	37	0	37
	R	1.105	46	0	46
	T	1.864	52	0	52
	S	3.746	59	60	1
Weld Metal	V	.5028	135	140	5
	R	1.105	168	165	3
	T	1.864	191	150	41
	S	3.746	218	205	13
HAZ Metal	V	.5028	---	0	---
	R	1.105	---	90	---
	T	1.864	---	100	---
	S	3.746	---	95	---

(a) Reference 1 (including its Reference 51).

TABLE 3 Calculation of Chemistry Factors Using Surveillance Capsule Data						
Material	Capsule	Fluence (x 10^{19} n/cm^2, E > 1.0 MeV)[a]	FF	ΔRT_{NDT} (°F)[a][b]	FF*ΔRT_{NDT} (°F)	FF^2
Intermediate Shell Forging 05 (Tangential)	V	.5028	.8081	25	20.2	.6530
	R	1.105	1.0279	25	25.7	1.0566
	T	1.864	1.1706	30	35.1	1.3703
	S	3.746	1.3418	42	56.4	1.8004
				Sum:	137.4	4.8803
	Chemistry Factor = 28.2°F					
Intermediate Shell	V	.5028	.8081	0	0	.6530
	R	1.105	1.0279	0	0	1.0566
	T	1.864	1.1706	0	0	1.3703
	S	3.746	1.3418	60	80.5	1.8004
				Sum:	80.5	4.8803
	Chemistry Factor = 16.5°F					
Weld Metal	V	.5028	.8081	149.7	121.0	.6530
	R	1.105	1.0279	176.4	181.3	1.0566
	T	1.864	1.1706	160.4	187.8	1.3703
	S	3.746	1.3418	219.1	294.0	1.8004
				Sum:	854.69	4.8803
	Chemistry Factor = 160.7°F					

NOTES:

(a) Reference 1.

(b) ΔRT_{NDT} for weld material is the adjusted value using the 1.069 ratioing factor per Reference 1 applied to the measured values of Table 2.

TABLE 4 Reactor Vessel Toughness Table (Unirradiated)[a]			
Material Description	Cu (%)	Ni(%)	Initial RT_{NDT}(°F)
Intermediate Shell	.07	.69	20
Lower Shell	.05	.69	40
Circumferential Weld	.25	.56	-4.8

(a) Per Reference 1.

TABLE 5 Reactor Vessel Surface Fluence Values at 19.5 and 32 EFPY[a] x 10^{19} (n/cm^2, E > 1.0 MeV)				
EFPY	0°	15°	30°	45°
19.5	2.32	1.47	1.05	.969
32	3.49	2.20	1.56	1.45

(a) Reference 1.

TABLE 6 Calculation of Adjusted Reference Temperatures at 24 EFPY for the Limiting Reactor Vessel Material		
Parameter	Values	
Operating Time	24 EFPY	
Material	Circ. Weld	Circ. Weld
Location	1/4-T	3/4-T
Chemistry Factor (CF), °F[b]	160.7	160.7
Fluence (f), 10^{19} n/cm^2 (E > 1.0 MeV)[a]	1.85	.851
Fluence Factor (FF)	1.17	.955
ΔRT_{NDT} = CF x FF, °F	188	153.4
Initial RT_{NDT} (I), °F	-4.8	-4.8
Margin (M), °F[a]	48.3	48.3
ART = I + (CFxFF) + M, °F[a][c]	232	196.9

NOTES:

(a) Value calculated using Table 5 values.

(b) Values from Table 3.

(c) Reference 1.