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ROBERT C. MECREDY
Vice President
Nuclear Operations

February 19, 2001

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

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

Ref.(a): Technical Specification 5.6.6

Dear Mr. Vissing:

Ginna Station Technical Specification 5.6.6 requires that RG&E provide the NRC with any revisions of the PTLR. As such, attached please find a copy of Revision 3 of this document which is applicable up to 28 effective full power years (EFPY). Future revisions to this document will be forwarded to the NRC in accordance with the applicable technical specification.

Very truly yours,

Robert C. Mecredy

Attachment

xc: Mr. Guy S. Vissing (Mail Stop 8C2)
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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

U.S. NRC Ginna Senior Resident Inspector

A001

1000263

R.E. Ginna Nuclear Power Plant

RCS Pressure and Temperature Limits Report
PTLR

Revision 3

Responsible Manager: George Wrobel

Effective Date: 2-15-2001
02-15-2001

Controlled Copy No. ________

1.0 RCS Pressure and Temperature Limits Report (PTLR)

This Pressure and Temperature Limits Report (PTLR) for the R.E. Ginna Nuclear Power Plant 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[1]

(LCO 3.4.3)
(LCO 3.4.12)

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 Figure PTLR - 1 and Figure PTLR - 2, respectively.

2.1.3 The minimum boltup temperature, using the methodology of Reference 4, Enclosure 2 is 60°F.

2.2 Low Temperature Overpressure Protection System Enable Temperature[2]

(LCO 3.4.6)
(LCO 3.4.7)
(LCO 3.4.10)
(LCO 3.4.12)

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[3]

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 PTLR - 1. The results of these examinations shall be used to update Figure PTLR - 1 and Figure PTLR - 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 PTLR - 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 $\pm$ 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 256.6°F for 32 EFPY per Reference 1.

4.2 Tables

Table PTLR - 1 contains the location and schedule for the removal of surveillance capsules.

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

Table PTLR - 4 provides the reactor vessel toughness data.

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

Table PTLR - 6 shows example calculations of the ART values at 28 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-NP-A, "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 Guy S Vissing, NRC, Subject: "Application for Amendment to Facility Operating License, Revision to Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR) Administrative controls Requirements," dated September 29, 1997.

4. Letter from R.C. Mecredy, RG&E, to Guy S. Vissing, NRC, "Clarifications to Proposed Low Temperature Overpressure Protection System Technical Specification," dated June 3, 1997.

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

6. Letter from R.C Mecredy, RG&E, to Guy S. Vissing, NRC, "Corrections to Proposed Low Temperature Overpressure Protection System Technical Specification," October 8, 1997.

7. RG&E Design Analysis DA-ME-97-031, "Evaluation of Ginna RCS Coolant Temperature to Support LTOPS Requirements," Revision 0.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: CIRCUMFERENTIAL WELD SA-847
LIMITING ART VALUES AT 28 EFPY: 1/4T, 236°F
3/4T, 204°F

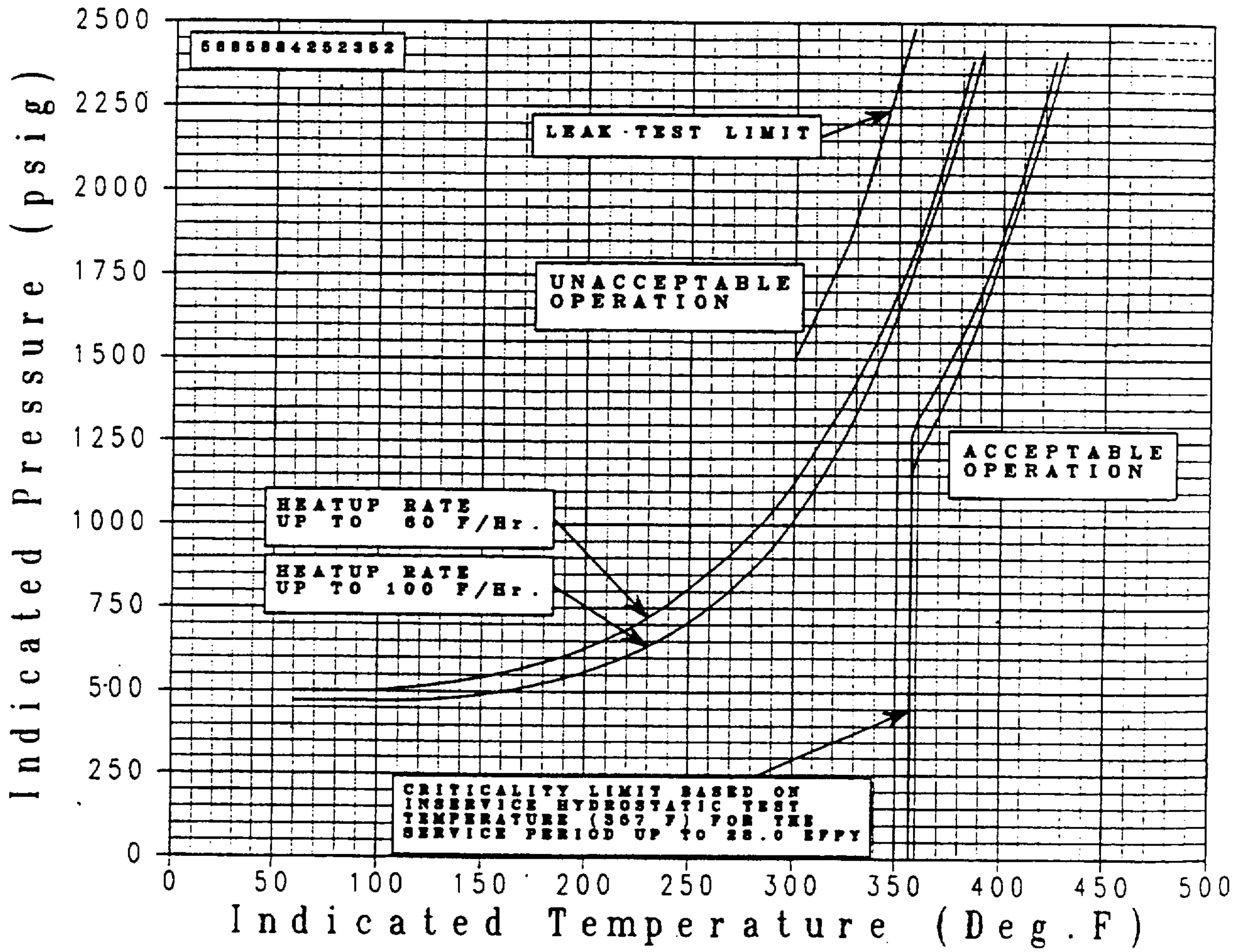

Figure PTLR - 1
R. E. Ginna Reactor Coolant System Heatup Limitations (Heatup Rates of 60 and 100°F) Applicable to 28 EFPY (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: CIRCUMFERENTIAL WELD SA-847
LIMITING ART VALUES AT 28 EFPY: 1/4T, 236°F
3/4T, 204°F

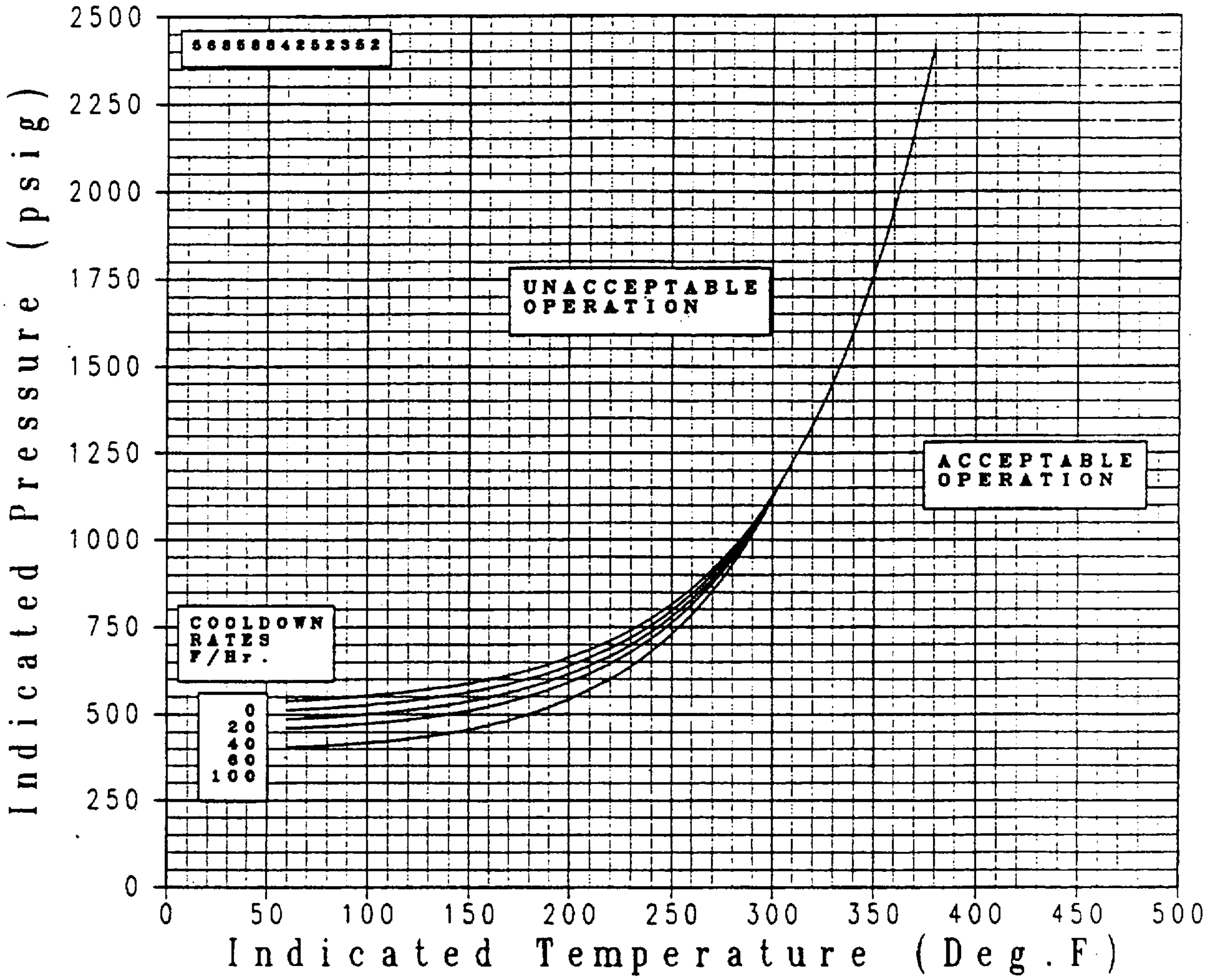

Figure PTLR - 2
R. E. Ginna Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 20 40 60 and 100°F/hr) Applicable to 28 EFPY (Without Margins for Instrumentation Errors)

Table PTLR - 1
Surveillance Capsule Removal Schedule

Capsule	Vessel Location (deg.)	Capsule Lead Factor	Removal Schedule[(a)]	Capsule Fluence E19(n/cm^2)[(b)]
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[(c)]	TBD[(c)]
P	247°	1.9	Standby	N/A

(a) Effective Full Power Years (EFPY).

(b) Reference 1.

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

Table PTLR - 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 PTLR - 3
Calculation of Chemistry Factors Using Surveillance Capsule Data

Material	Capsule	Fluence ($\times 10^{19}$ 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					

(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 PTLR - 2.

Table PTLR - 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 PTLR - 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 PTLR - 6
Calculation of Adjusted Reference Temperatures at 28 EFPY for the Limiting Reactor Vessel Material

Parameter	Values	
Operating Time	28 EFPY	
Material	Circ. Weld	Circ. Weld
Location	1/4-T	3/4-T
Chemistry Factor (CF), °F[a]	160.7	160.7
Fluence (f), 10^{19} n/cm^2 (E > 1.0 MeV)[b]	2.11	.965
Fluence Factor (FF)	1.20	1.00
ΔRT_{NDT} = CF x FF, °F	192.8	160.7
Initial RT_{NDT} (I), °F	−4.8	−4.8
Margin (M), °F[b]	48.3	48.3
ART = I + (CFxFF) + M, °F[b][c]	236.3	204.2

(a) Values from Table PTLR - 3.

(b) Value calculated using Table PTLR - 5 values.

(c) Reference 1.

END NOTES

1. (Reference 1)
2. (Methodology of Reference 3, Attachment VI and Reference 6, as calculated in Reference 7.)
3. (Methodology of Reference 3, Attachment VI and Reference 6, as calculated in Reference 3, Attachment VII.)