RG&E	GINNA STATION	PTLR Revision 1

RCS PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

Responsible Manager

5/24/96

Effective Date

Controlled Copy No. _____

R.E. Ginna Nuclear Power Plant

RCS Pressure and Temperature Limits Report

Revision 1

Per letter from J.A. Mitchell, NRC, to R.C. Mecredy, RG&E, Subject: "R.E. Ginna - Acceptance for Referencing of Pressure Temperature Limits Report, Revision 1 (TAC No. M94770)," this revision of the PTLR is only valid until December 31, 1996 at which time NRC approval of a revised PTLR is required for continued operation.

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 PTLR Figures 1 and 2)

2.2.1 The enable temperature for the Low Temperature Overpressure Protection System is 328°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 References 6 and 7, 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 limits.

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 266.5°F for 32 EFPY per References 6, 7, and 8.

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 22 EFPY for the limiting reactor vessel material.

5.0 REFERENCES

1. Letter from R.C. Mecredy, RG&E, to A.R. Johnson, NRC, Subject: "Application for Amendment to Facility Operating License, Revision to Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," dated April 22, 1996.

2. WCAP-14040, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," Revision 1, December 1994 as approved by letter from C.I. Grimes, NRC to R.H. Newton, WOG, Subject: "Acceptance for Referencing of Topical Report WCAP-14040, Revision 1", (TAC No. M91749), dated October 16, 1995.

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.

6. WCAP-13902, "Analysis of Capsule S from the Rochester Gas and Electric Corporation R.E. Ginna Reactor Vessel Radiation Surveillance Program," dated December 1993.

7. Letter from A.R. Johnson, NRC, to R.C. Mecredy, RG&E, Subject: "R.E. Ginna Nuclear Power Plant - Pressurized Thermal Shock Evaluation (TAC No. M93827)," dated March 22, 1996.

8. Letter from Westinghouse Electric Corporation (SE-REA-96-072) to G.J. Wrobel, RG&E, Subject: "Vessel Fluence Re-evaluation for R.E. Ginna," dated April 16, 1996.

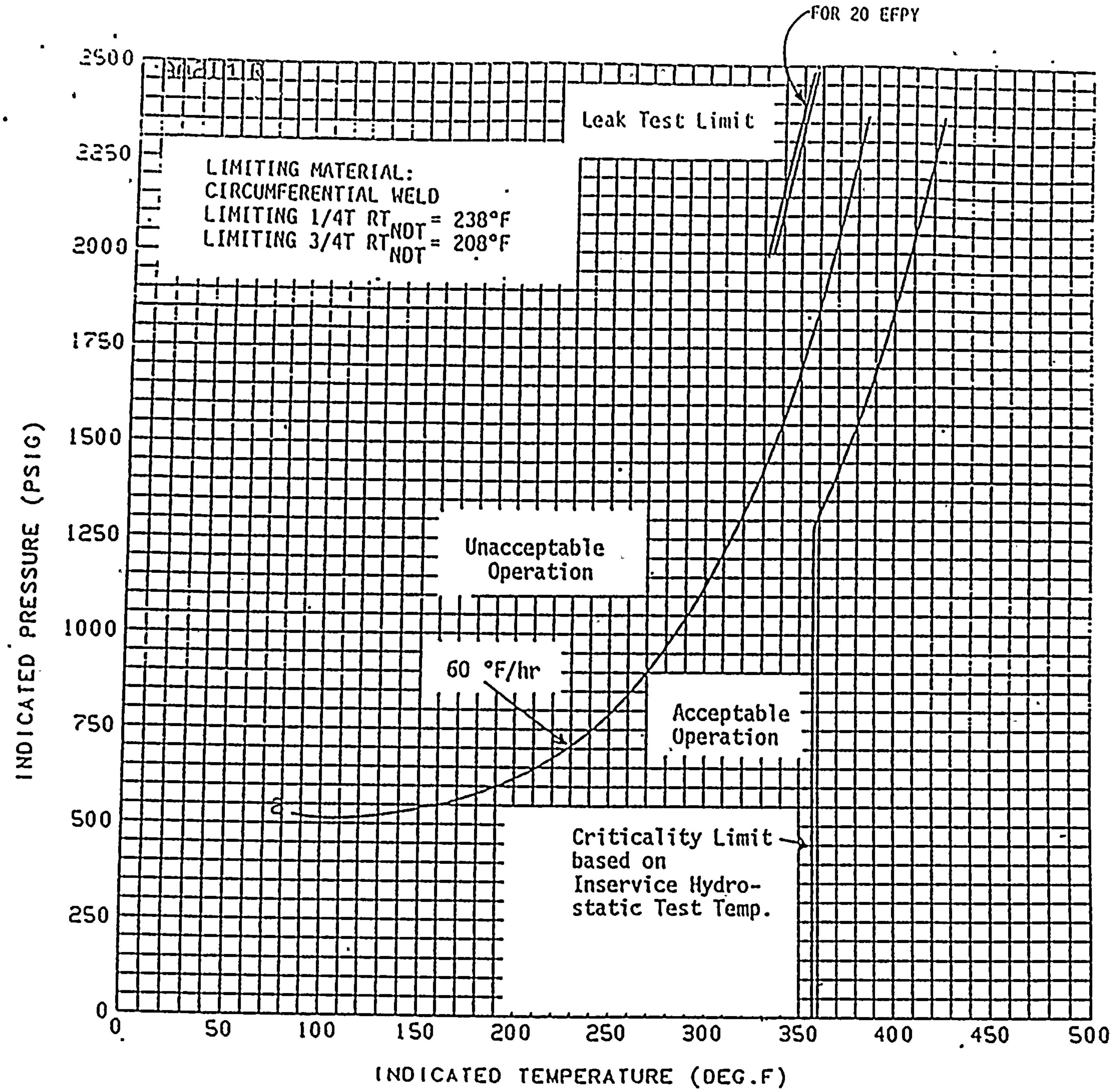

FIGURE 1

REACTOR VESSEL HEATUP LIMITATIONS
APPLICABLE FOR THE FIRST 22 EFPY

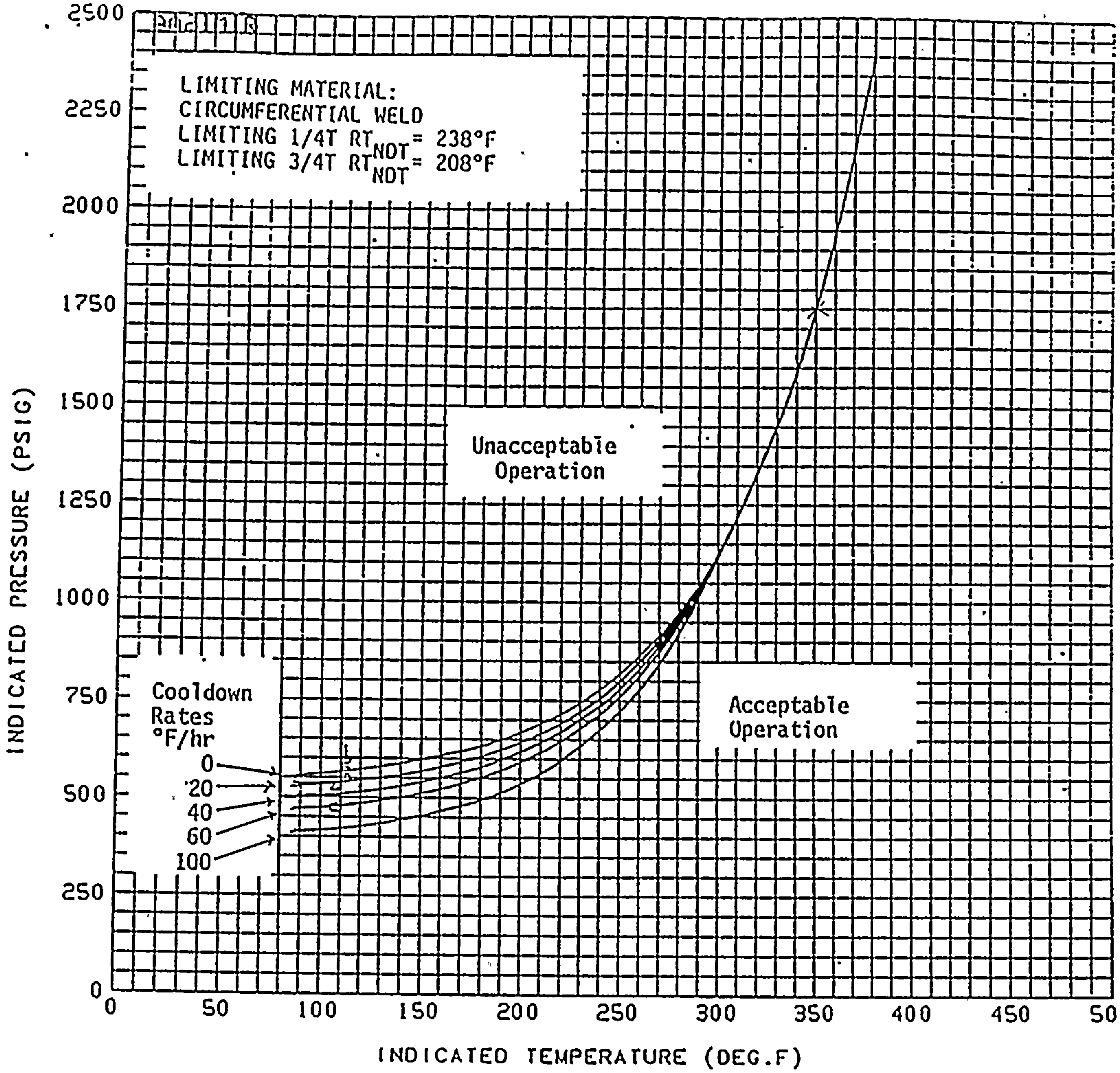

FIGURE 2

REACTOR VESSEL COOLDOWN LIMITATIONS
APPLICABLE FOR THE FIRST 22 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)	0.612
R	257°	3.00	2.7 (removed)	1.242
T	67°	1.85	7 (removed)	2.128
S	57°	1.74	17 (removed)	4.180
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 6 with ENDFB-VI adjustment per Reference 8.

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	0.612	26	25	1
	R	1.242	32	25	7
	T	2.128	37	30	7
	S	4.180	42	42	0
Intermediate Shell	V	0.612	37	0	37
	R	1.242	46	0	46
	T	2.128	52	0	52
	S	4.180	50	60	1
Weld Metal	V	0.612	135	140	5
	R	1.242	168	165	3
	T	2.128	191	150	41
	S	4.180	218	205	13
HAZ Metal	V	0.612	---	0	---
	R	1.242	---	90	---
	T	2.128	---	100	---
	S	4.180	---	95	---

(a) Reference 6 with ENDFB-VI adjustment per Reference 8.

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	0.612	0.862	25	21.55	0.743
	R	1.242	1.060	25	26.5	1.124
	T	2.128	1.205	30	36.15	1.452
	S	4.180	1.366	42	57.37	1.866
				Sum:	141.57	5.185
	Chemistry Factor = 136.39°F					
Intermediate Shell	V	0.612	0.862	0	0	0.743
	R	1.242	1.060	0	0	1.124
	T	2.128	1.205	0	0	1.452
	S	4.180	1.366	60	81.96	1.866
				Sum:	81.96	5.185
	Chemistry Factor = 15.81°F					
Weld Metal	V	0.612	0.862	158.2	136.37	0.743
	R	1.242	1.060	186.45	197.64	1.124
	T	2.128	1.205	169.5	204.25	1.452
	S	4.180	1.366	231.65	316.43	1.866
				Sum:	854.69	5.185
	Chemistry Factor = 164.8°F					

NOTES:

(a) Reference 6 with ENDFB-VI adjusted per Reference 8.

(b) ΔRT_{NDT} for weld material is the adjusted value using the 1.13 ratioing factor per Reference 7 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 7.

TABLE 5 Reactor Vessel Surface Fluence Values at 17 and 32 EFPY[a] x 10^{19} (n/cm^2, E > 1.0 MeV)				
EFPY	0°	15°	30°	45°
17	2.26	1.38	0.997	0.886
32	3.94	2.44	1.81	1.65

(a) Reference 6 with ENDFB-VI adjustment per Reference 8.

TABLE 6 Calculation of Adjusted Reference Temperatures at 22 EFPY for the Limiting Reactor Vessel Material		
Parameter	Values	
Operating Time	22 EFPY	
Material	Circ. Weld	Circ. Weld
Location	1/4-T	3/4-T
Chemistry Factor (CF), °F[b]	164.8	164.8
Fluence (f), $+10^{19}$ n/cm^2 (E > 1.0 MeV)[a]	1.91	0.875
Fluence Factor (FF)	1.177	0.963
ΔRT_{NDT} = CF x FF, °F	193.97	158.70
Initial RT_{NDT} (I), °F	-4.8	-4.8
Margin (M), °F[a]	48.3	48.3
ART = I + (CFxFF) + M, °F[a][c]	~ 238°F	~ 203°F

NOTES:

(a) Value calculated using Table 5 values.

(b) Values from Table 3.

(c) ART value is a conservative approximate per adjustment of ENDFB-VI.