RG&E	GINNA STATION	PTLR Revision 0

RCS PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

Responsible Manager

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R.E. Ginna Nuclear Power Plant

RCS Pressure and Temperature Limits Report

Revision 0

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, 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) (Reference 1)

2.2.1 The enable temperature for the Low Temperature Overpressure Protection System is 330°F.

2.3 Low Temperature Overpressure Protection System Setpoints (LCO 3.4.12)

2.3.1 Pressurizer Power Operated Relief Valve Lift Setting Limits (Reference 1)

The lift setting for the pressurizer Power Operated Relief Valves (PORVs) is ≤ 424 psig.

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. 3) 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 4, 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 is 275.2°F for 32 EFPY per Reference 5. (Note - these values are based on Capsule T. The new revised RT_{PTS} values based on Capsule S will be implemented following NRC review of these values).

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 21 EFPY for the limiting reactor vessel material.

5.0 REFERENCES

1. Letter from A.R. Johnson, NRC, to R.C. Mecredy, RG&E, Subject: "Issuance of Amendment No. 48 to Facility Operating License No. DPR-18, R.E. Ginna Nuclear Power Plant (TAC No. M79828)," dated March 6, 1992.

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

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

5. Letter from George E. Lear, NRC to Roger Kober, RG&E, "Safety Evaluation by Office of Nuclear Reactor Regulation Regarding Projected Values of Material Properties for Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events, Rochester Gas and Electric Company, R.E. Ginna Nuclear Power Plant Docket No. 50-244 (TAC No. 59956)," dated November 17, 1986.

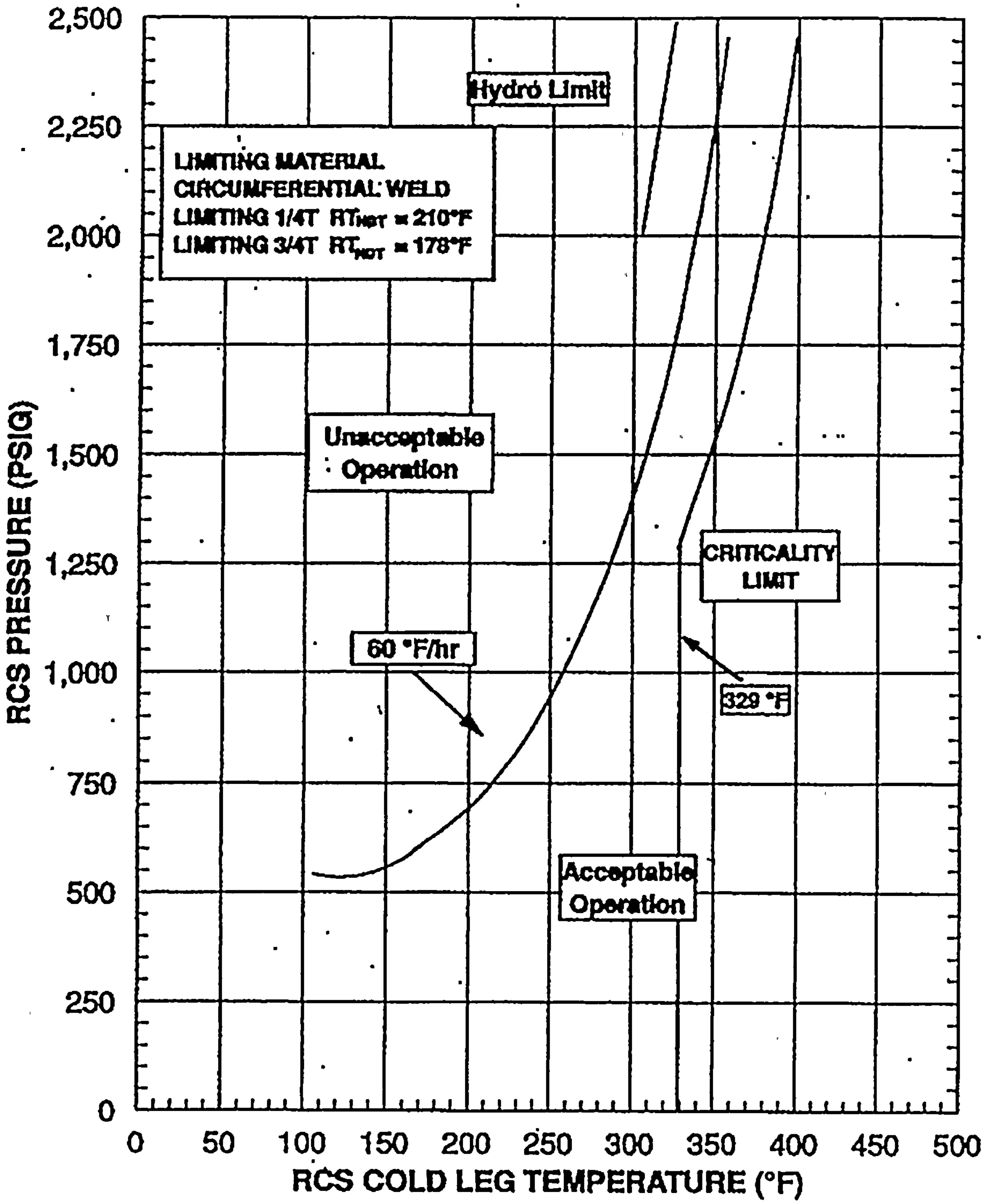

FIGURE 1
REACTOR VESSEL HEATUP LIMITATIONS
APPLICABLE FOR THE FIRST 21 EFPY

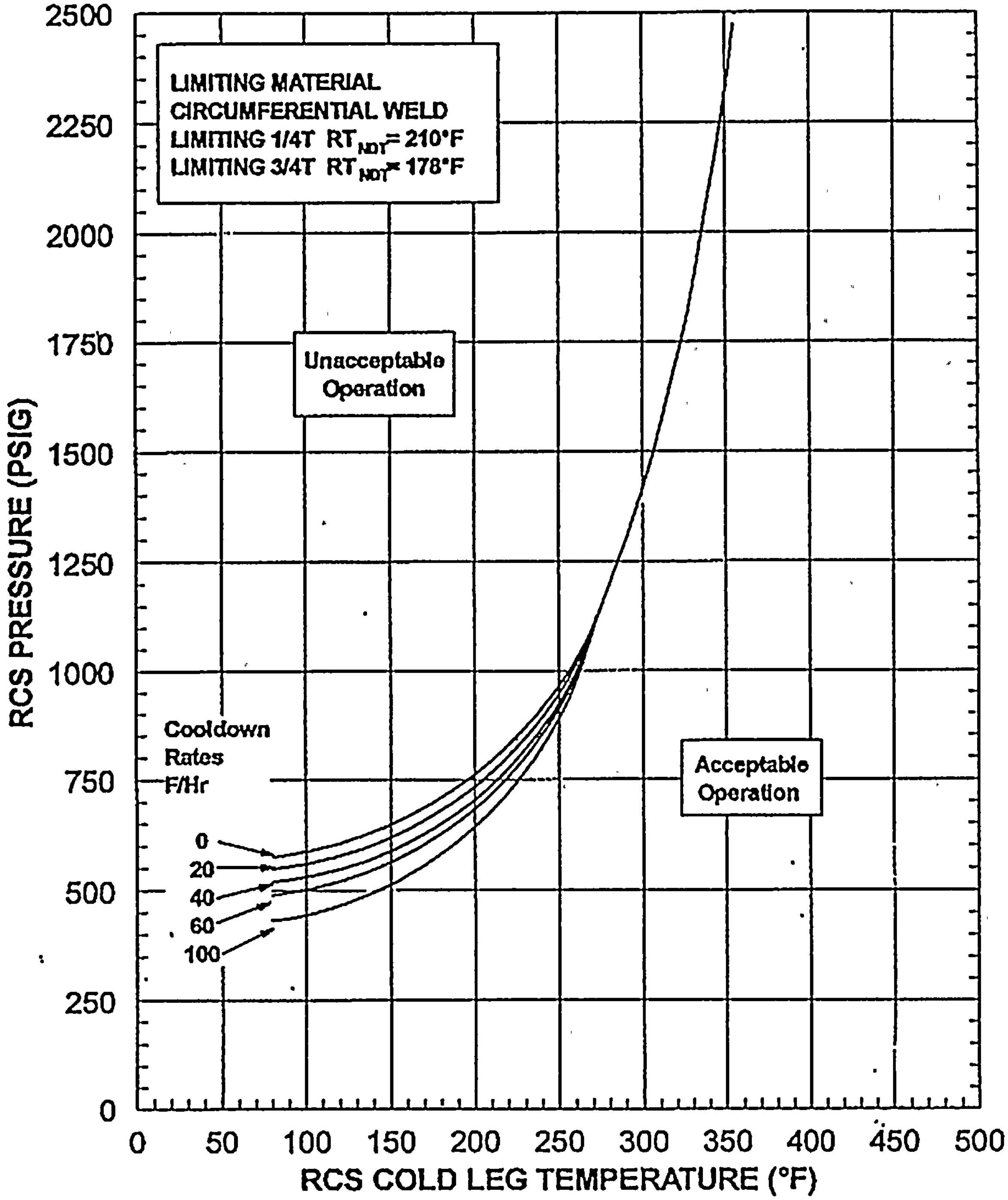

FIGURE 2

REACTOR VESSEL COOLDOWN LIMITATIONS
APPLICABLE FOR THE FIRST 21 EFPY

Table 1

Surveillance Capsule Removal Schedule

Capsule	Vessel Location (deg.)	Capsule Lead Factor	Removal Schedule[a]	Capsule Fluence E19(n/cm^2)
V	77°	2.99	1.6 (removed)	0.556
R	257°	3.00	2.7 (removed)	1.15
T	67°	1.85	7 (removed)	1.97
S	57°	1.74	17 (removed)[c]	3.87
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) Currently under NRC review, not included in current heat up/cooldown curve.

TABLE 2

Surveillance Material 30 ft-lb Transition Temperature Shift

Material	Capsule	Fluence (x 10^{19} n/cm^2, E > 1.0 MeV)[b]	30 lb-ft Transition Temperature Shift		
			Predicted[a] (°F)	Measured (°F)	Δ (°F)
Lower Shell	V	.703	26	25	1
	R	1.01	32	25	7
	T	1.75	37	30	7
Intermediate Shell	V	.703	37	0	37
	R	1.01	46	0	46
	T	1.75	52	0	52
Weld Metal	V	.703	135	140	5
	R	1.01	168	165	3
	T	1.75	191	150	41
HAZ Metal	V	.703	---	0	---
	R	1.01	---	90	---
	T	1.75	---	100	---

(a) Based upon Reg. Guide 1.99, Revision 2 predictions

(b) Letter from A. Johnson (NRC) to R. Mecredy (RG&E) "Issuance of Amendment No. 48 to Facility Operating License No. DPR-18, R.E. Ginna Nuclear Power Plant (TAC No. M79828)," dated March 6, 1992 especially Reference (5) of Section 3.1.2.

TABLE 3 Calculation of Chemistry Factors Using Surveillance Capsule Data						
Material	Capsule	Fluence (x 10^{19} n/cm^2, E > 1.0 MeV)	FF	ΔRT_{NDT} (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Intermediate Shell Forging 05 (Tangential)	V	.703	0.901	25	22.5	.812
	R	1.01	1.003	25	25.1	1.006
	T	1.75	1.154	30	34.6	1.332
				Sum:	82.2	3.15
	Chemistry Factor[a] = 26.1					
Intermediate Shell	V	.703	0.901	0	0	.812
	R	1.01	1.003	0	0	1.006
	T	1.75	1.154	0	0	1.332
				Sum:	0.0	3.15
	Chemistry Factor[a] = 0.0					
Weld Metal	V	.703	.901	140	126.1	.812
	R	1.01	1.003	165	165.5	1.006
	T	1.75	1.154	150	173.1	1.332
				Sum:	464.7	3.15
	Chemistry Factor[a] = 147.5°F					

<u>NOTES</u>:

(a) Letter from A. Johnson (NRC) to R. Mecredy (RG&E) "Issuance of Amendment No. 48 to Facility Operating License No. DPR-18, R.E. Ginna Nuclear Power Plant (TAC No. M79828)," dated March 6, 1992 especially Reference (5) of Section 3.1.2.

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	.55	0

(a) Letter from A. Johnson (NRC) to R. Mecredy (RG&E) "Issuance of Amendment No. 48 to Facility Operating License No. DPR-18, R.E. Ginna Nuclear Power Plant (TAC No. M79828)," dated March 6, 1992 especially Reference (5) of Section 3.1.2.

TABLE 5 Reactor Vessel Surface Fluence Values at 7 and 21 EFPY[a] x 10^{19} (n/cm^2, E > 1.0 MeV)				
EFPY	0°	14.5°	30°	44.5°
7	.866	.538	.359	.310
21	2.32[b]	1.42	0.991	.893

(a) WCAP-11026 "R.E. Ginna Reactor Vessel Fluence and RT_{PTS} Evaluations," dated December 1985 Table II.2-3 through II.2-6, as referenced from License Amendment No. 48.

(b) Letter from A. Johnson (NRC) to R. Mecredy (RG&E) "Issuance of Amendment No. 48 to Facility Operating License No. DPR-18, R.E. Ginna Nuclear Power Plant (TAC No. M79828)," dated March 6, 1992 especially Reference (5) of Section 3.1.2.

TABLE 6 Calculation of Adjusted Reference Temperatures at 21 EFPY for the Limiting Reactor Vessel Material		
Parameter	Values	
Operating Time	21 EFPY	
Material	Circ. Weld	Circ. Weld
Location	1/4-T	3/4-T
Chemistry Factor (CF), °F[b]	147.5	147.5
Fluence (f), $\div 10^{19}$ n/cm^2 (E > 1.0 MeV)[a]	1.57	.720
Fluence Factor (FF)[a]	1.125	.908
ΔRT_{NDT} = CF x FF, °F	165.9	133.93
Initial RT_{NDT} (I), °F	0	0
Margin (M), °F[a]	44	44
ART = I + (CFxFF) + M, °F[a]	210	178°F

NOTES:

(a) Value calculated from Table 4 note[a] reference.

(b) Values calculated from Table 3 and Table 4.