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May 19, 2009

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: **R.E. Ginna Nuclear Power Plant**
Docket No. 50-244

Transmittal of RCS Pressure and Temperature Limits Report (PTLR)

In accordance with the R.E. Ginna Nuclear Power Plant Improved Technical Specification 5.6.6, which requires the submittal of revisions to the PTLR, the attached report is hereby submitted.

There are no new commitments being made in this submittal. Should you have questions regarding the information in this submittal, please contact Thomas Harding at (585) 771-5219 or Thomas.HardingJr@Constellation.com.

Very truly yours,

Thomas L. Harding

Attachment: Ginna PTLR, Revision 5

cc: S. J. Collins, NRC
D.V. Pickett, NRC
Resident Inspector, NRC (Ginna)

A001
NRR

WPLNRC-1002135

Constellation Energy
R.E. Ginna Nuclear Power Plant

RCS Pressure and Temperature Limits Report
PTLR

Revision 5

Responsible Manager: [signature]

Effective Date: 5/14/09

Controlled Copy No. _______

Record Cat.# 4.43.3

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. Reference 1 calculates Pressure/ Temperature Limits out to 52 EFPY pre-Extended Power Uprate (EPU). Reference 9 determines the data in Reference 1 is valid out to 47.3 EFPY post-EPU. The titles and labels in the PTLR will show the 47.3 EFPY.

2.1 RCS Pressure and Temperature Limits

(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. These curves are based on Reference 1 as modified in Reference 12 to include instrument errors.

2.1.3 The minimum boltup temperature, using the methodology of Reference 4, Enclosure 2 is 60°F (Reference 12).

2.2 Low Temperature Overpressure Protection System Enable Temperature (Calculated in Reference 12)

(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 (See Reference 12)

The lift setting for the pressurizer Power Operated Relief Valves (PORVs) is ≤ 410 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 as modified by Ref. 1) 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 7 (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 uses RG 1.99 Rev. 2 Regulatory Position 1.1.

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 53 EFPY post-EPU for Ginna Station limiting beltline material is 273.1°F for welds and 116.4°F for forgings per Reference 12.

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 47.3 EFPY for the limiting reactor vessel material.

5.0 REFERENCES

1. WCAP-15885, Revision 0, "R. E. Ginna Heatup and Cooldown Limit Curves for Normal Operation," dated July 2002.

2. WCAP-14040-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," Revision 4, May 2004.

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

8. Letter from M. Korsnick, CEG, to US NRC Document Control Desk, Subject: R. E. Ginna Nuclear Power Plant, Licensee Amendment Request Regarding Extended Power Power Uprate. (Attachment 5 - Licensing Report), dated July 7, 2005.

9. CN-RCDA-04-149, Revision 2, "Ginna Extended Power Uprate Program Reactor Vessel Integrity Evaluations."

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

11. BAW-1803, Revision 1, "Correlations for Predicting the Effects of Neutron Radiation on Linde 80 Submerged-Arc Welds," dated May 1991.

12. DA-ME-08-020, Revision 1, "Pressure Temperature Limit Report (PTLR) Supporting Analysis," dated March 23, 2009.

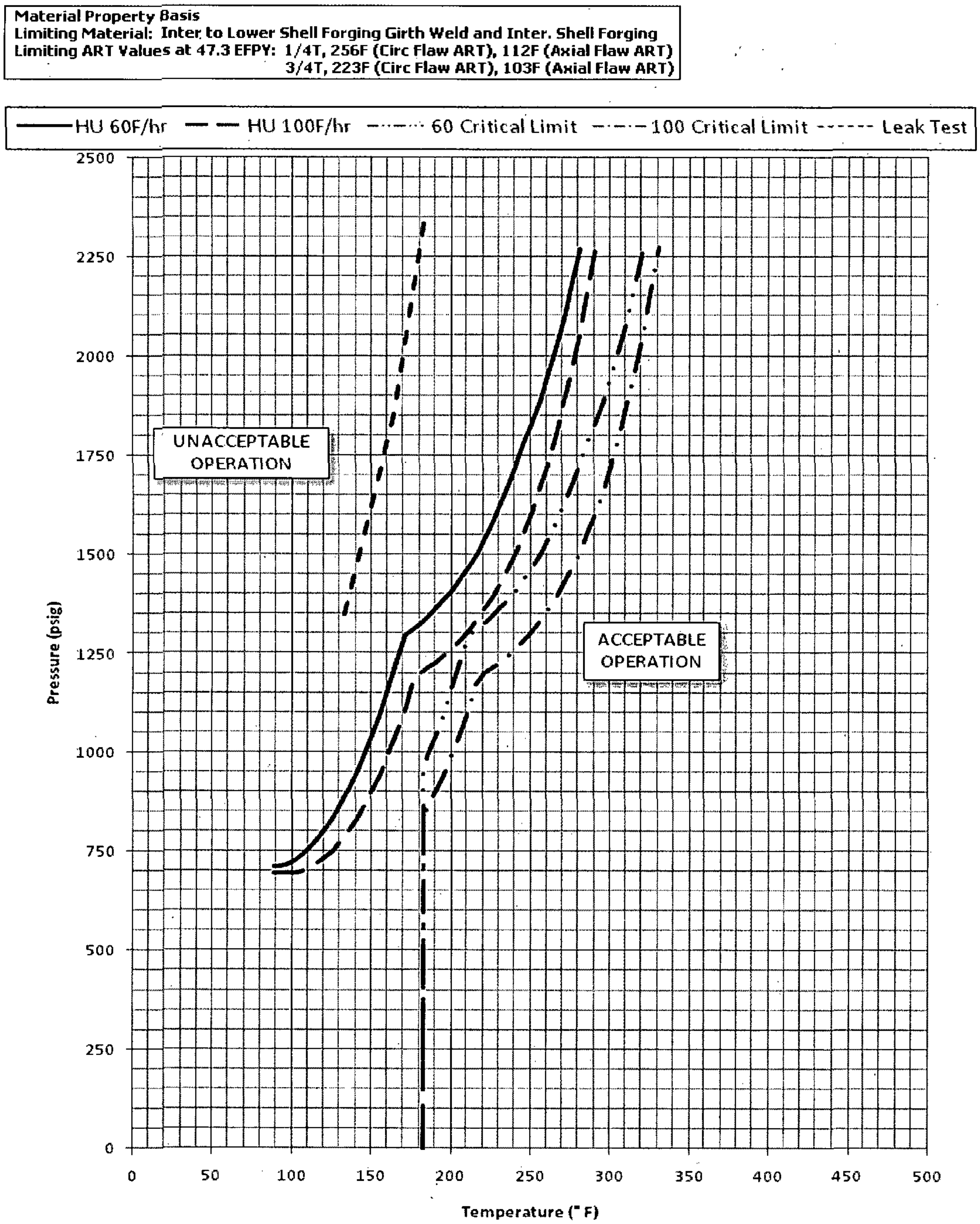

Figure PTLR - 1
R. E. Ginna Reactor Coolant System Heatup Limitations (Heatup Rates up to 100°F/hr) Applicable for the First 47.3 EFPY (Including Normal Instrument Errors) (Reference 12)

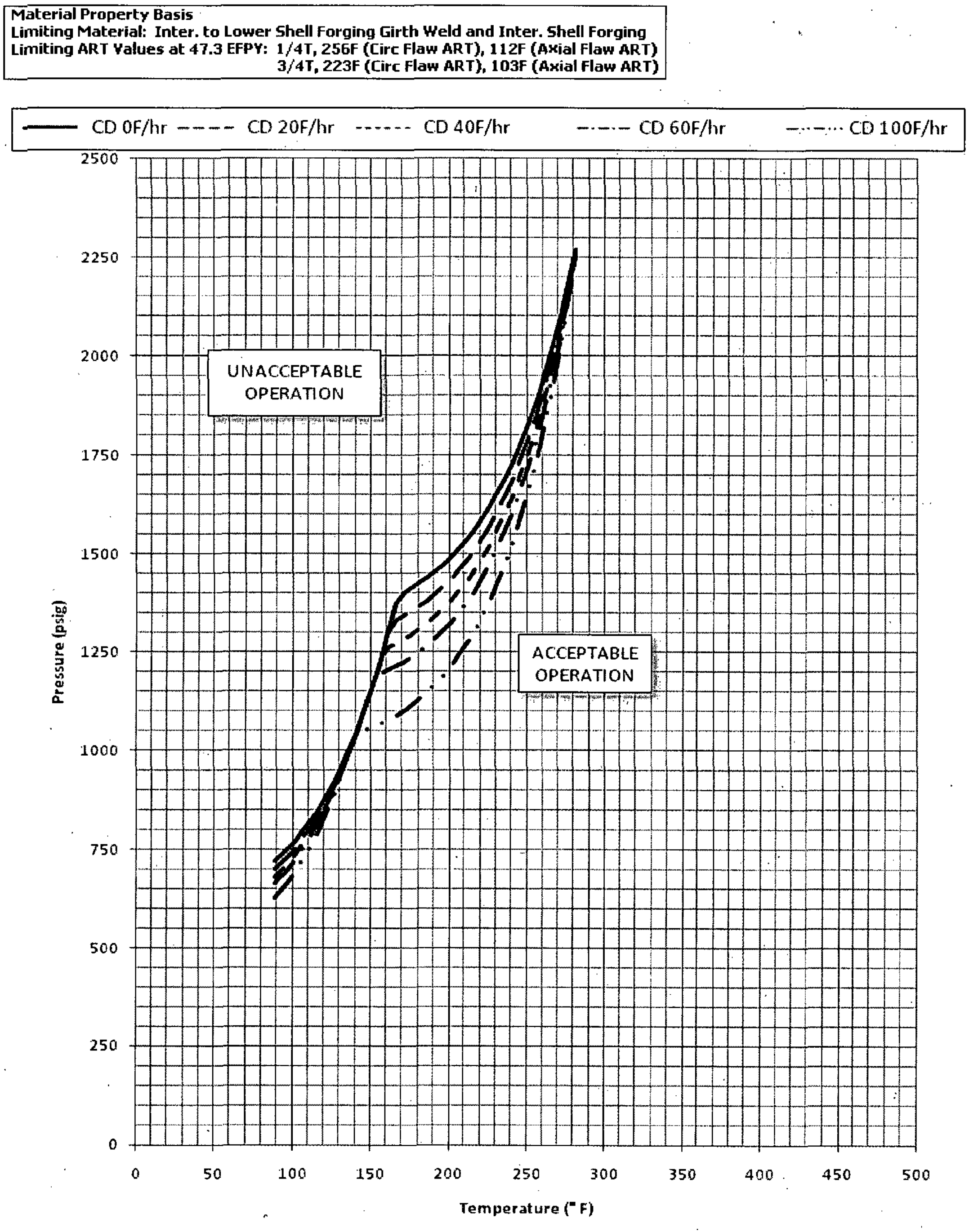

Figure PTLR - 2
R. E. Ginna Reactor Coolant System Cooldown Limitations (Cooldown Rates of up to 100°F/hr) Applicable for the First 47.3 EFPY (Including Normal Instrument Errors) (Reference 12)

Table PTLR - 1
Surveillance Capsule Removal Schedule[a]

Capsule	Vessel Location (deg.)	Capsule Lead Factor	Removal Schedule EFPY	Capsule Fluence E19(n/cm^2)
V	77°	2.96	1.4 (removed)	0.587
R	257°	2.97	2.6 (removed)	1.02
T	67°	1.82	6.9 (removed)	1.69
S	57°	1.79	17 (removed)	3.64
N	237°	1.81	TBD[b]	TBD[b]
P	247°	1.91	TBD[c]	N/A

(a) Reference 12.

(b) Capsule N was removed shortly after receiving a fast neutron fluence equivalent to operation to 2029 (60 year license). The fluence on Capsule N will be between 1 and 2 times the peak end of life fluence. Removal was in the Spring Outage of 2008. Analysis is in-process.

(c) Capsule P will be removed shortly following receiving a fast neutron fluence equivalent to operations to 2049. The specific withdrawal EFPY and fluence will be determined following the analysis of Capsule N.

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 (ΔRT_{NDT}) Predicted[(b)] (°F)	Measured[(c)] (°F)
	V	0.587	60	25
	R	1.02	65	25
	T	1.69	69	30
Lower Shell	S	3.64	75	42
	V	0.587	71	0
	R	1.02	78	0
	T	1.69	84	0
Intermediate Shell	S	3.64	93	60
	V	0.587	191	140
	R	1.02	216	165
	T	1.69	238	150
Weld Metal	S	3.64	268	205
	V	0.587	--	0
	R	1.02	--	90
	T	1.69	--	100
HAZ Metal	S	3.64	--	95

(a) Reference 1

(b) Using Equations of RG 1.99 Revision 2, with material chemistry of Table PTLR-4, plus 2 standard deviations of ΔRT_{NDT} (17F for forges, 28F welds) per Generic Letter 96-03 Reviewer Note 7.

(c) Table 5.10 of Reference 10.

Table PTLR - 3
Calculation of Chemistry Factors using R. E. Ginna, Turkey Point & Davis Besse Surveillance Capsule Data

Material	Capsule	Capsule f[a]	FF[b]	ΔRT_{NDT}[c]	FF*ΔRT_{NDT}	FF^2
Lower Shell Forging 125P666	V	0.587	0.851	25	21.275	0.724
	R	1.02	1.006	25	25.150	1.012
	T	1.69	1.144	30	34.320	1.309
	S	3.64	1.335	42	56.070	1.782
				Sum:	136.815	4.827
	$CF_{LSF\ 125P666} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (136.815) \div (4.827) = 28.3°F$					
Intermediate Shell Forging 125S255	V	0.587	0.851	0	0	0.724
	R	1.02	1.006	0	0	1.012
	T	1.69	1.144	0	0	1.309
	S	3.64	1.335	60	80.1	1.782
				Sum:	80.1	4.827
	$CF_{ISF\ 125S255} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (80.1) \div (4.827) = 16.6°F$					
Ginna Surveillance Weld Metal (Heat # 61782)	V	0.587	0.851	149.8 (140)	127.480	0.724
	R	1.02	1.006	176.6 (165)	177.660	1.012
	T	1.69	1.144	160.5 (150)	183.612	1.309
	S	3.64	1.335	219.4 (205)	292.899	1.782
				Sum:	781.651	4.827
	$CF_{Ht.\ \#61782} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (781.651) \div (4.827) = 161.9°F$					

Table PTLR - 3
Calculation of Chemistry Factors using R. E. Ginna, Turkey Point & Davis Besse Surveillance Capsule Data

Material	Capsule	Capsule f(a)	FF(b)	ΔRT_{NDT}(c)	FF*ΔRT_{NDT}	FF^2
Turkey Point Surveillance Weld Material(d) (Heat # 71249)	Davis	2.956	1.287	221 (215)	284.427	1.656
	T (TP3)	0.699	0.900	163 (166)	146.700	0.810
	V (TP3)	1.484	1.109	176 (179)	195.184	1.230
	T (TP4)	0.673	0.889	208 (211)	184.912	0.790
				Sum:	811.223	4.486
	$CF_{Ht.\ \#71249} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (811.223°F) \div (4.486) = 180.8°F$					

(a) f = fluence. See Table 3 of Reference 1 , (x 10^{19} n/cm^2, E > 1.0 MeV)

(b) FF= fluence factor = $f^{(0.28 - 0.1 * \log f)}$

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values taken from the following documents:
- Ginna Plate and Weld...WCAP-14684
- Turkey Point & Davis Bessie...WCAP-15092 R.3

(d) Ginna operates with an average of the inlet temperature for each capsule that was removed of approximately 549°F, Turkey Point 3&4 operate with an average inlet temperature of approximately 546°F, and Davis Besse operates with an average inlet temperature of approximately 555°F. The measured ΔRT_{NDT} values from the Turkey Point 3&4 surveillance program were adjusted by subtracting 3°F to each measured ΔRT_{NDT} and the Davis Besse surveillance program data was adjusted by adding 6°F to the measured ΔRT_{NDT} value before applying the ratio procedure. The surveillance weld metal ΔRT_{NDT} values have been adjusted by a ratio factor of:
Ratio Ginna = 1.07, Ratio Turkey Point = 1.0 (conservative), Ratio Davis Besse = 1.0 (conservative). The pre-adjusted values are in parenthesis. Since Turkey Point and Davis Besse material is similiar to Ginna's, this is acceptable.

Table PTLR - 4
Reactor Vessel Toughness Table (Unirradiated) [a]

Material Description	Cu (%)	Ni (%)	Initial RT_{NDT} (°F)
Reactor Upper Closure Head Flange	n/a	n/a	0
Intermediate Shell	.07	.69	20
Lower Shell	.05	.69	40
Circumferential Weld	.25[b]	.56[b]	-4.8[c]

(a) Per Reference 1.

(b) For use in Table PTLR-2, material for the Circumferential Weld is based on Table 1 of Reference 1: Cu 0.23% and Ni 0.53%

(c) Per Reference 11.

Table PTLR - 5
Reactor Vessel Surface Fluence Values at 32 and 47.3 EFPY[a] x 10^{19}(n/cm^2, E > 1.0 MeV)

EFPY	0°	15°	30°	45°
32	3.26	2.05	1.48	1.33
47.3	4.85	3.05	2.21	2.00

(a) Reference 1.

Table PTLR - 6
Calculation of Adjusted Reference Temperatures at 47.3 EFPY for the Limiting Reactor Vessel Material

Parameter	Values			
Operating Time	47.3 EFPY			
Material	Inter. to Lower Shell Circ. Weld	Inter. Shell	Inter. to Lower Shell Circ. Weld	Inter. Shell
Location	1/4-T	1/4-T	3/4-T	3/4-T
Chemistry Factor (CF), °F[(a)]	161.9	44	170.4	44
Fluence (f), 10^{19} n/cm^2 (E > 1.0 MeV)[(b)]	3.28	3.28	1.51	1.51
Fluence Factor (FF)	1.31	1.31	1.11	1.11
ΔRT_{NDT} = CF x FF, °F	212.1	57.6	189.1	48.8
Initial RT_{NDT} (I), °F	-4.8[(c)]	20	-4.8[(c)]	20
Margin (M), °F[(b)]	48.3[(c)]	34	48.3[(c)]	34
ART = I + (CFxFF) + M, °F[(b)(c)]	256	112	223	103

(a) Values from Tables 21 and 22 of Reference 1.

(b) Per Reference 1

(c) Per Reference 11.