Westinghouse

Westinghouse Electric Company
Nuclear Power Plants
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355
USA

U.S. Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, D.C. 20555

Direct tel: 412-374-6206
Direct fax: 412-374-5005
e-mail: sisk1rb@westinghouse.com

Your ref: Docket Number 52-006
Our ref: DCP/NRC2145

May 30, 2008

Subject: AP1000 Submittal of APP-RXS-Z0R-001, Revision 1

Westinghouse is submitting Revision 1 of APP-RXS-Z0R-001, "AP1000 Generic Pressure Temperature Limits Report." The purpose of this report is to provide the basis for the use, on a generic basis, of the pressure temperature limit curves found in DCD figures 5.3-2 and 5.3-3. Revision 0 of this report was a Westinghouse internal revision not submitted to the NRC.

This report is submitted in support of the AP1000 Design Certification Amendment Application (Docket No. 52-006). The information provided in this report is generic and is expected to apply to all Combined Operating License (COL) applicants referencing the AP1000 Design Certification and the AP1000 Design Certification Amendment Application.

Questions or requests for additional information related to the content and preparation of this report should be directed to Westinghouse. Please send copies of such questions or requests to the prospective applicants for combined licenses referencing the AP1000 Design Certification. A representative for each applicant is included on the cc: list of this letter.

Very truly yours,

Robert Sisk, Manager
Licensing and Customer Interface
Regulatory Affairs and Standardization

/Enclosure

1. APP-RXS-Z0R-001, Revision 1, "AP1000 Generic Pressure Temperature Limits Report"

00385psa.doc

D063
NRO

cc:	D. Jaffe	-	U.S. NRC	1E
	E. McKenna	-	U.S. NRC	1E
	P. Buckberg	-	U.S. NRC	1E
	P. Ray	-	TVA	1E
	P. Hastings	-	Duke Power	1E
	R. Kitchen	-	Progress Energy	1E
	A. Monroe	-	SCANA	1E
	J. Wilkinson	-	Florida Power & Light	1E
	C. Pierce	-	Southern Company	1E
	E. Schmiech	-	Westinghouse	1E
	G. Zinke	-	NuStart/Entergy	1E
	R. Grumbir	-	NuStart	1E
	D. Ekeroth	-	Westinghouse	1E

ENCLOSURE 1

APP-RXS-Z0R-001

Revision 1

"AP1000 Generic Pressure Temperature Limits Report"

AP1000 DOCUMENT COVER SHEET

TDC: ______ Permanent File: ______

AP1000 DOCUMENT NO.	REVISION	PAGE	ASSIGNED TO	OPEN ITEMS (Y/N)
APP-RXS-Z0R-001	1	1 of 16	W-NS-ES-POAM	N

ALTERNATE DOCUMENT NUMBER: WORK BREAKDOWN #:

ORIGINATING ORGANIZATION: Westinghouse Electric Company

TITLE: AP1000 GENERIC PRESSURE TEMPERATURE LIMITS REPORT

ATTACHMENTS: None	DCP #/REV. INCORPORATED IN THIS DOCUMENT REVISION: Class 3 changes only
CALCULATION/ANALYSIS REFERENCE: None	

ELECTRONIC FILENAME	ELECTRONIC FILE FORMAT	ELECTRONIC FILE DESCRIPTION
APP-RSX-Z0R-001.doc	Microsoft Word Document	

☒ © 2008 WESTINGHOUSE ELECTRIC COMPANY LLC – WESTINGHOUSE NON-PROPRIETARY CLASS 3

Class 3 Documents being transmitted to the NRC require the following two review signatures in lieu of a Form 36.

LEGAL REVIEW	SIGNATURE / DATE (If processing electronic approval select option)
T. J. WHITE	T. J. White 5-29-08
PATENT REVIEW Douglas E. Ekeroth	SIGNATURE / DATE Douglas E. Ekeroth 5/29/08

☐ © 2008 WESTINGHOUSE ELECTRIC COMPANY LLC – WESTINGHOUSE PROPRIETARY CLASS 2

This document is the property of and contains Proprietary Information owned by Westinghouse Electric Company LLC and/or its subcontractors and suppliers. It is transmitted to you in confidence and trust, and you agree to treat this document in strict accordance with the terms and conditions of the agreement under which it was provided to you.

☐ © 2008 WESTINGHOUSE ELECTRIC COMPANY LLC and/or STONE & WEBSTER, INC.
WESTINGHOUSE PROPRIETARY CLASS 2 and/or STONE & WEBSTER CONFIDENTIAL AND PROPRIETARY

This document is the property of and contains Proprietary Information owned by Westinghouse Electric Company LLC and/or is the property of and contains Confidential and Proprietary Information owned by Stone & Webster, Inc. and/or their affiliates, subcontractors and suppliers. It is transmitted to you in confidence and trust, and you agree to treat this document in strict accordance with the terms and conditions of the agreement under which it was provided to you.

ORIGINATOR Frank C. Gift Jr.	SIGNATURE / DATE (If processing electronic approval select option) Frank C Gift 05/28/2008
REVIEWER Thomas M. Malota	SIGNATURE / DATE Thomas M. Malota 05/29/2008
REVIEWER Dale A. Wiseman	SIGNATURE / DATE Dale A. Wiseman 5/29/08
	SIGNATURE / DATE
VERIFIER Natalie R. Jurcevich	SIGNATURE / DATE Natalie Jurcevich 5/29/08 — Verification Method: Independent Review

**Plant Applicability: ☒ All AP1000 plants except: No exceptions
☐ Only the following plants:

APPLICABILITY REVIEWER** James A. Speer	SIGNATURE / DATE James A. Speer 5/29/08
RESPONSIBLE MANAGER* Douglas E. Ekeroth	SIGNATURE / DATE Douglas E. Ekeroth 5/29/08

* Approval of the responsible manager signifies that the document and all required reviews are complete, the appropriate proprietary class has been assigned, electronic file has been provided to the EDMS, and the document is released for use.

Table of Contents

List of Tables

List of Figures

RECORD OF REVISION

Revision 0:	**Original Issue**
Revision 1:	**This revision was issued to change the document's proprietary classification from Proprietary Class 2 to Non-Proprietary Class 3. This change is a Class 3 DCP change.**

1.0 RCS PRESSURE TEMPERATURE LIMITS REPORT (PTLR)

This PTLR has been prepared in accordance with the requirements of Technical Specification (TS) 5.6.6. The limiting condition for operation (LCO) 3.4.3, reactor coolant system (RCS) pressure and temperature (P/T) limits, also referred to as heatup and cooldown limit curves, for the AP1000 design are contained in this PTLR. Revisions to the PTLR shall be provided to the U.S. Nuclear Regulatory Commission (NRC) after issuance. Note that this is a generic PTLR, and accordingly utilizes generic inputs to determine the P/T limit curves. The inputs used herein will be substantiated for use as a plant-specific PTLR by the Licensee. This PTLR follows the guidelines set forth in U.S. NRC Generic Letter 96-03[1]. Revision 1 of this document was created to establish a non-proprietary version for issuance to the NRC for review.

2.0 RCS PRESSURE AND TEMPERATURE LIMITS

The RCS P/T limits for LCO 3.4.3, are presented in Figures 1 and 2 and tabulated in Tables 1 and 2, were developed using the NRC-approved methodology in the latest approved revision of WCAP-14040-NP-A[2].

The boltup temperature shall be ≥ 60°F. The minimum (allowable) boltup temperature is established as the higher of 60°F or the highest material reference temperature (initial RT_{NDT}) in the highly-stressed reactor pressure vessel (RPV) flange region. The heatup and cooldown curves utilize 60°F, since this value is limiting for the respective vessel materials. This allows the plant to perform the boltup operations at 60°F or higher.

The RCS temperature rate-of-change limits are:

a. A maximum heatup rate of 100°F in any one hour period.

b. A maximum cooldown rate of 100°F in any one hour period.

c. A maximum temperature change of 10°F in any one-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

The RCS P/T limits for heatup, cooldown, inservice hydrostatic and leak testing, and criticality are specified by Figures 1 and 2. Data points for these figures are tabulated in Tables 1 and 2, respectively. The 54 effective full power years' (EFPY) term of applicability identified for these limit curves is based on adjusted reference temperature (ART) calculations that utilize Regulatory Guide 1.99, Revision 2[3] Position 1.1 chemistry factors and peak neutron fluence projections on the vessel forgings and beltline welds. The inputs for the ART calculations are contained in References 4 and 5, and are summarized in Tables 3 through 5. Calculated ART values, used in the development of the P/T limits, are documented in Reference 6, as summarized in Table 6. The inputs and calculations are discussed in Section 4.0.

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. Table 1 of ASTM E-185[7] identifies the requirement for four capsules to be withdrawn for a maximum projected transition temperature shift (ΔRT_{NDT}) of the beltline materials exceeding 100°F. The AP1000 surveillance program withdrawal schedule for removal of the capsules for post-irradiation testing exceeds this requirement and identifies five capsules to be withdrawn.

The surveillance capsule withdrawal schedule and pressure vessel steel surveillance program is in compliance with Appendix H to 10CFR50, "Reactor Vessel Material Surveillance Program Requirements"[8]. Accordingly, the surveillance capsule withdrawal schedule meets the requirements of ASTM E185[7], supplemented as needed for a 60-year design life. The results of these examinations shall be used to update the RCS P/T limits. The recommended surveillance capsule withdrawal schedule, shown below, is consistent with the AP1000 certified design.

Capsule	Withdrawal Time
1st	When the accumulated neutron fluence of the capsule is 5 x 10^{18} n/cm^2.
2nd	When the accumulated neutron fluence of the capsule corresponds to the approximate end of life fluence at the reactor vessel 1/4T location.
3rd	When the accumulated neutron fluence of the capsule corresponds to the approximate end of life fluence at the reactor vessel inner wall location.
4th	When the accumulated neutron fluence of the capsule corresponds to a fluence not less than once or greater than twice the peak end of vessel life fluence.
5th	End of plant design objective of 60 years.
6th	Standby
7th	Standby
8th	Standby

4.0 SUPPLEMENTAL DATA TABLES

Table 3 contains the elemental chemistry limits for the manufacture of the AP1000 reactor vessel beltline materials, along with the Regulatory Guide 1.99, Revision 2[3] Position 1.1 chemistry factors based on these limits. The chemistry values listed are the maximum allowed values and are therefore conservative for use in determining the effects of irradiation embrittlement for the beltline materials.

Table 4 contains the mechanical properties of the reactor vessel materials utilized in calculating the ARTs for the P/T limit curves, the preliminary pressurized thermal shock (PTS) reference temperature (RT_{PTS}) values, and projection of upper shelf energy (USE) for the reactor vessel materials.

Table 5 contains the maximum projected neutron fluence values for the beltline materials of the reactor vessel at end-of-life (EOL), which was assumed to be 54 EFPY, along with the calculated fluence values at the 1/4 T position (vessel wall quarter thickness from the inside surface) and 3/4 T position (vessel wall quarter thickness from the outside surface).

Table 6 contains the calculations of the ARTs at the 1/4 T position and 3/4 T position, which are used in the determination of the P/T limit curves for normal operation[9,10].

Table 7 contains the calculations of ΔRT_{NDT} and RT_{PTS}. As noted in Section 3, values of ΔRT_{NDT} exceeding 100°F require four surveillance capsules to be withdrawn in accordance with Reference 7. The maximum ΔRT_{NDT} value is calculated to be 102.1°F (based on Position 1.1 Chemistry Factors). The screening criteria for PTS are provided in 10CFR50.61[11], which states that the values of RT_{PTS} (using EOL neutron fluence projections) must be less than 270°F for plates, forgings, and axial weld materials, and less than 300°F for circumferential weld materials. The preliminary RT_{PTS} values are calculated to be 98°F for the beltline forgings and 148°F for the beltline circumferential welds, and are in compliance with Reference 11.

Table 8 contains the conservative projections of USE for the reactor vessel beltline materials to show compliance with the requirements of Reference 9.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL:	SHELL FORGING
LIMITING ART VALUES AT 54 EFPY:	1/4T, 90°F
	3/4T, 82°F

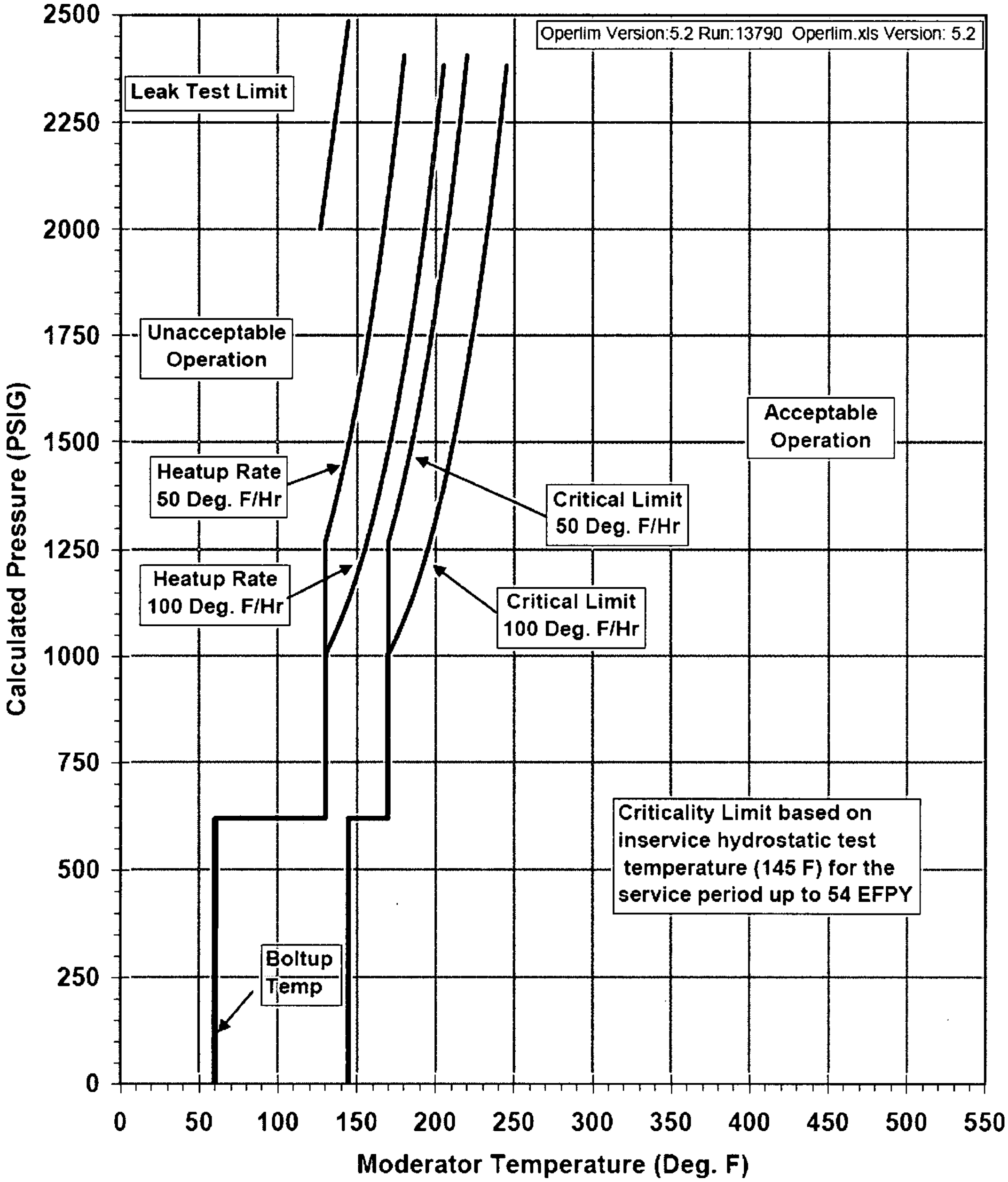

Figure 1: Reactor Coolant System Heatup Limitations (Maximum Heatup Rate of 100°F/hr) Applicable to 54 EFPY (without Margins for Instrumentation Errors)

(Plotted Data provided on Table 1)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL:	SHELL FORGING
LIMITING ART VALUES AT 54 EFPY:	1/4T, 90°F
	3/4T, 82°F

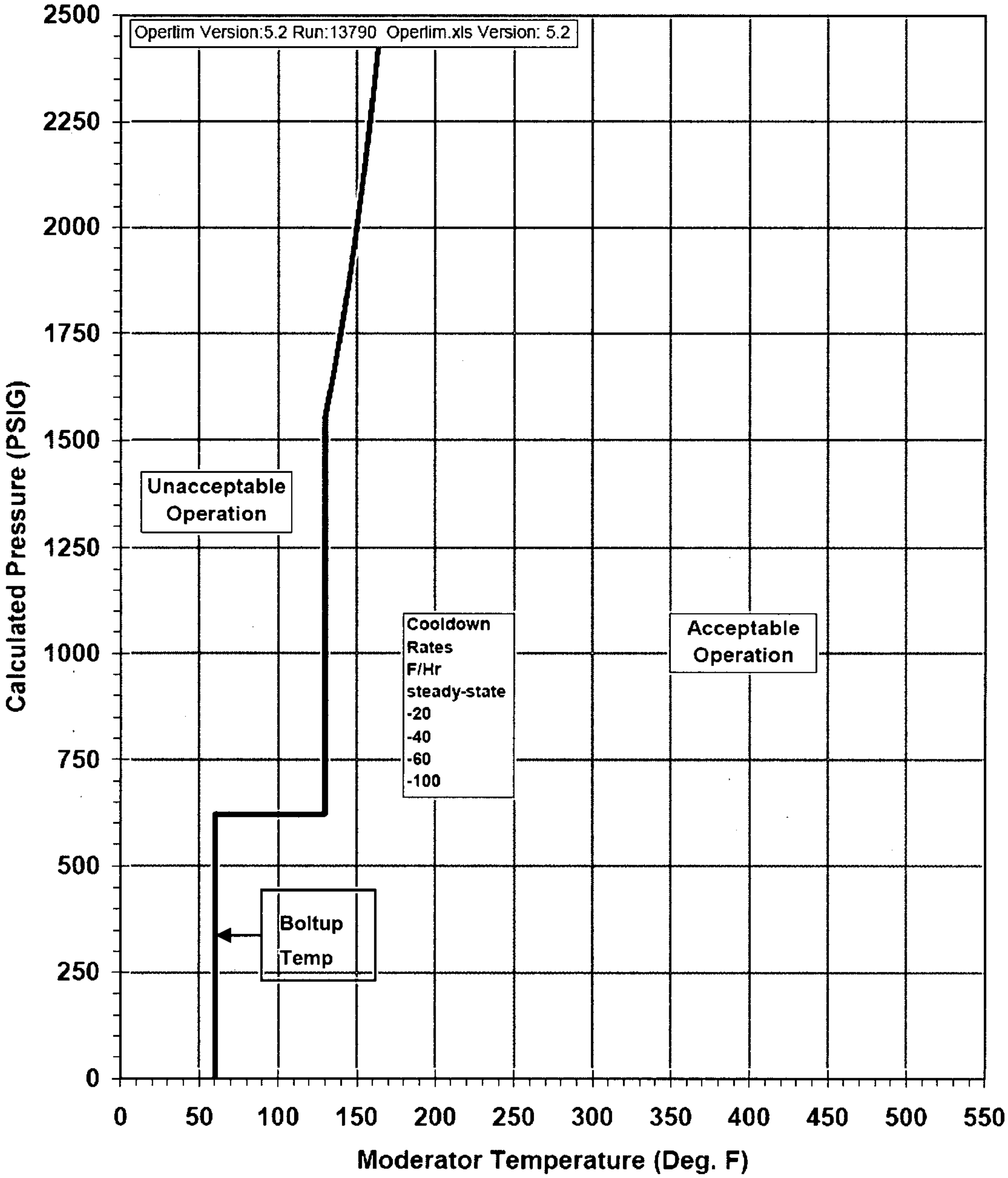

Figure 2: Reactor Coolant System Cooldown Limitations (Maximum Cooldown Rate of 100°F/hr) Applicable to 54 EFPY (without Margins for Instrumentation Error)

(Plotted Data provided on Table 2)

Table 1: RCS Heatup Limits at 54 EFPY

(without uncertainties for instrumentation errors)

60°F/hr Heatup		Critical Limit		100°F/hr Heatup		Critical Limit		Leak Test Limit	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
60	0	145	0	60	0	145	0	127	2,000
60	621	145	621	60	621	145	621	127	2,000
65	621	145	621	65	621	145	621	145	2,485
70	621	145	621	70	621	145	621	145	2,485
75	621	145	621	75	621	145	621		
80	621	145	621	80	621	145	621		
85	621	145	621	85	621	145	621		
90	621	145	621	90	621	145	621		
95	621	145	621	95	621	145	621		
100	621	145	621	100	621	145	621		
105	621	150	621	105	621	150	621		
110	621	155	621	110	621	155	621		
115	621	160	621	115	621	160	621		
120	621	165	621	120	621	165	621		
125	621	170	621	125	621	170	621		
130	621	170	1,268	130	621	170	1,004		
130	621	175	1,337	130	621	175	1,042		
130	1,268	180	1,413	130	1,004	180	1,085		
135	1,337	185	1,499	135	1,042	185	1,134		
140	1,413	190	1,593	140	1,085	190	1,188		
145	1,499	195	1,697	145	1,134	195	1,249		
150	1,593	200	1,812	150	1,188	200	1,318		
155	1,697	205	1,940	155	1,249	205	1,394		
160	1,812	210	2,081	160	1,318	210	1,478		
165	1,940	215	2,236	165	1,394	215	1,572		
170	2,081	220	2,408	170	1,478	220	1,675		
175	2,236			175	1,572	225	1,790		
180	2,408			180	1,675	230	1,917		
				185	1,790	235	2,058		
				190	1,917	240	2,213		
				195	2,058	245	2,384		
				200	2,213				
				205	2,384				

Table 2: RCS Cooldown Limits at 54 EFPY

(without uncertainties for instrumentation errors)

Steady State		20°F/hr		40°F/hr		60°F/hr		100°F/hr	
T (°F)	**P (psig)**	**T (°F)**	**P (psig)**	**T (°F)**	**P (psig)**	**T (°F)**	**P (psig)**	**T (°F)**	**P (psig)**
60	0	60	0	60	0	60	0	60	0
60	621	60	621	60	621	60	621	60	621
65	621	65	621	65	621	65	621	65	621
70	621	70	621	70	621	70	621	70	621
75	621	75	621	75	621	75	621	75	621
80	621	80	621	80	621	80	621	80	621
85	621	85	621	85	621	85	621	85	621
90	621	90	621	90	621	90	621	90	621
95	621	95	621	95	621	95	621	95	621
100	621	100	621	100	621	100	621	100	621
105	621	105	621	105	621	105	621	105	621
110	621	110	621	110	621	110	621	110	621
115	621	115	621	115	621	115	621	115	621
120	621	120	621	120	621	120	621	120	621
125	621	125	621	125	621	125	621	125	621
130	621	130	621	130	621	130	621	130	621
130	621	130	621	130	621	130	621	130	621
130	1,557	130	1,557	130	1,557	130	1,557	130	1,557
135	1,653	135	1,653	135	1,653	135	1,653	135	1,653
140	1,758	140	1,758	140	1,758	140	1,758	140	1,758
145	1,874	145	1,874	145	1,874	145	1,874	145	1,874
150	2,003	150	2,003	150	2,003	150	2,003	150	2,003
155	2,145	155	2,145	155	2,145	155	2,145	155	2,145
160	2,302	160	2,302	160	2,302	160	2,302	160	2,302
165	2,476	165	2,476	165	2,476	165	2,476	165	2,476

Table 3: Maximum Composition Limits for Reactor Vessel Beltline Materials

Reactor Vessel Beltline Materials	Element's Maximum Weight %					$CF^{(a)}$
	Cu	Ni	P	V	S	
Beltline Forgings	0.06	0.85	0.01	0.05	0.01	37
Beltline Circumferential Welds	0.06	0.95	0.01	0.05	0.01	82

Notes:
(a) Chemistry factors (CFs) are based on the maximum allowed Cu and Ni weight percentage values for manufacture. These chemistry factors were determined in accordance with Position 1.1 of Regulatory Guide 1.99, Revision 2[3] and have units of °F.

Table 4: Generic Mechanical Properties for Reactor Vessel Materials

Reactor Vessel Materials	Initial RT_{NDT} (a)	Initial USE (b)
Beltline Forgings	-10°F	> 75 ft-lbs
Beltline Circumferential Welds	-20°F	> 75 ft-lbs
Reactor Vessel Closure Head	10°F	N/A
Reactor Vessel Flange	10°F	N/A

Notes:
(a) These generic values for initial RT_{NDT} are the basis for the ART calculations used to determine the P/T limit curves. These generic values are also used in the RT_{PTS} calculations for EOL.
(b) The USE for the unirradiated materials that comprise regions of the reactor vessel that will be exposed to neutron fluences estimated to be over 1 x 10^{19} n/cm^2 (E > 1.0 MeV) must be at least 75 ft-lbs [11].

Table 5: Peak Reactor Vessel Neutron Fluence Projections at 54 EFPY

Reactor Vessel Materials	Surface Fluence(a)	1/4 T Fluence(b)	3/4 T Fluence(c)
Beltline Forgings	8.9 x 10^{19} n/cm^2	5.38 x 10^{19} n/cm^2	1.96 x 10^{19} n/cm^2
Upper Circumferential Weld	1.25 x 10^{19} n/cm^2	0.755 x 10^{19} n/cm^2	0.276 x 10^{19} n/cm^2
Lower Circumferential Weld	2.5 x 10^{19} n/cm^2	1.51 x 10^{19} n/cm^2	0.551 x 10^{19} n/cm^2

Notes:
(a) The surface fluence values represent the maximum projected neutron fluence (E > 1.0 MeV) at the clad/base metal interface for each of these separate reactor vessel beltline materials.
(b) The 1/4 T fluence values represent the attenuated neutron fluence values (E > 1.0 MeV) at the quarter-thickness position in the vessel wall from the inside surface, based on a wall thickness of 8.4 inches, calculated using Equation 3 in Regulatory Guide 1.99, Revision 2[3].
(c) The 3/4 T fluence values represent the attenuated neutron fluence values (E > 1.0 MeV) at the quarter-thickness position in the vessel wall from the outside surface, based on a wall thickness of 8.4 inches, calculated using Equation 3 in Regulatory Guide 1.99, Revision 2[3].

Table 6: Adjusted Reference Temperature Calculations at 54 EFPY

1/4 T Position ART Calculations						
Reactor Vessel Materials	**CF (°F)**	**1/4 T FF[(a)]**	**IRT_{NDT}[(b)] (°F)**	**ΔRT_{NDT}[(c)] (°F)**	**Margin[(d)] (°F)**	**ART[(e)] (°F)**
Beltline Forgings	37	1.416	-10	52.4	48.1	90[(f)]
Lower Circumferential Weld	82	1.114	-20	91.3	65.5	137
Upper Circumferential Weld	82	0.921	-20	75.5	65.5	121
3/4 T Position ART Calculations						
Reactor Vessel Materials	**CF (°F)**	**3/4 T FF[(a)]**	**IRT_{NDT}[(b)] (°F)**	**ΔRT_{NDT}[(c)] (°F)**	**Margin[(d)] (°F)**	**ART[(e)] (°F)**
Beltline Forgings	37	1.184	-10	43.8	48.1	82[(f)]
Lower Circumferential Weld	82	0.833	-20	68.3	65.5	114
Upper Circumferential Weld	82	0.649	-20	53.2	65.5	99

Notes:

(a) The fluence factor (FF) is determined from the corresponding neutron fluence value by the equation provided in Regulatory Guide 1.99, Revision 2[3], which states that $FF = f^{0.28 - 0.10 \log f}$, where f is the high energy neutron fluence value (10^{19} n/cm^2, E > 1.0 MeV). 1/4 T and 3/4 T fluence values are provided in Table 5.

(b) Initial RT_{NDT} values are the generic values taken from Table 4.

(c) ΔRT_{NDT} = CF * FF, as stated by Equation 2 in Regulatory Guide 1.99, Revision 2[3].

(d) Margin (M) = $2 * (\sigma_i^2 + \sigma_\Delta^2)^{1/2}$, as stated by Equation 4 in Regulatory Guide 1.99, Revision 2[3], where σ_i is the standard deviation for the initial RT_{NDT} values and σ_Δ is the standard deviation for ΔRT_{NDT}. Conservatively, σ_I was chosen to be 17°F for the calculations, even though the initial RT_{NDT} values used represent bounding conditions for measured RT_{NDT} values (of the vessel materials), which would normally be assigned a σ_I value of 0°F. The σ_Δ values were 28°F for the welds and 17°F for the base metal (forging), as specified in the last paragraph of Section 1.1 of Regulatory Guide 1.99, Revision 2[3].

(e) ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin, as stated by Equation 2 in Regulatory Guide 1.99, Revision 2[3].

(f) Note that the limiting ART values used in the development of the P/T limit curves are those calculated for the forgings, which require consideration of the more restrictive axial flaw orientation in comparison to the relaxed "Circ-Flaw" methodology allowed for the circumferential girth welds[9,10].

Table 7: ΔRT_{NDT} and RT_{PTS} Calculations at 54 EFPY

Reactor Vessel Materials	CF (°F)	FF[(a)]	IRT_{NDT}[(b)] (°F)	ΔRT_{NDT}[(c)] (°F)	Margin[(d)] (°F)	RT_{PTS}[(e)] (°F)
Beltline Forgings	37	1.499	-10	55.4	48.1	94
Lower Circumferential Weld	82	1.246	-20	**102.1**	65.5	**148**
Upper Circumferential Weld	82	1.062	-20	87.1	65.5	133

Notes:

(a) The fluence factor (FF) is determined from the corresponding neutron fluence value by the equation provided in Regulatory Guide 1.99, Revision 2[3], which states that FF = $f^{0.28 - 0.10 \log f}$, where f is the high energy neutron fluence value (10^{19} n/cm^2, E > 1.0 MeV).

(b) Initial RT_{NDT} values are the generic values taken from Table 4.

(c) ΔRT_{NDT} = CF * FF, as stated by Equation 2 in Regulatory Guide 1.99, Revision 2[3].

(d) Margin (M) = 2 $*(\sigma_i^2 + \sigma_\Delta^2)^{1/2}$, as stated by Equation 4 in Regulatory Guide 1.99, Revision 2[3], where σ_i is the standard deviation for the initial RT_{NDT} values and σ_Δ is the standard deviation for ΔRT_{NDT}. Conservatively, σ_I was chosen to be 17°F for the calculations, even though the initial RT_{NDT} values used represent bounding conditions for measured RT_{NDT} values (of the vessel materials), which would normally be assigned a σ_I value of 0°F. The σ_Δ values were 28°F for the welds and 17°F for the base metal (forging), as specified in the last paragraph of Section 1.1 of Regulatory Guide 1.99, Revision 2[3].

(e) RT_{PTS} = Initial RT_{NDT} + ΔRT_{PTS} + Margin, as stated by Equation 4 in 10CFR50.61[11], where $\Delta RT_{PTS} \equiv \Delta RT_{NDT}$.

Table 8: Beltline Reactor Vessel Materials USE Projection at 54 EFPY

Reactor Vessel Materials	Initial USE[(a)]	1/4 T Fluence[(b)]	Position 1.2 USE Decrease[(c)]	Projected EOL USE[(d)]
Beltline Forgings	> 75 ft-lbs	5.38 x 10^{19} n/cm^2	28%	54 ft-lbs[(e)]
Beltline Circumferential Weld	> 75 ft-lbs	1.51 x 10^{19} n/cm^2	23%	58 ft-lbs

Notes:

(a) The upper shelf energy for the unirradiated materials that comprise regions of the reactor vessel that will be exposed to neutron fluences estimated to be over 1 x 10^{19} n/cm^2 (E > 1.0 MeV) must be at least 75 ft-lbs[9]. Modern advances in manufacturing and steel production have generally provided reactor vessel plates, forgings, and welds with USE values that exceed 100 ft-lbs.

(b) USE projections are based on the 1/4 T neutron fluence value in from Table 5; the higher value for the lower circumferential weld was used for the EOL USE projection of the girth welds.

(c) The upper shelf energy is predicted to decrease (in accordance with Position 1.2 of Regulatory Guide 1.99, Revision 2[3]) as a function of neutron fluence and Cu content as illustrated in Figure 2 of Regulatory Guide 1.99, Revision 2[3].

(d) The projected EOL (54 EFPY) USE values are based on the limiting copper chemistry allowed (see Table 4) and an assumed 75 ft-lb USE for the unirradiated material (the minimum allowed).

(e) The predicted decrease in USE for the beltline forging conservatively uses the 0.10 wt% Cu content line for base metal (from Figure 2 of Regulatory Guide 1.99, Revision 2[3]) in the projection, even though the maximum Cu content permitted is 0.06 weight percentage.

5.0 REFERENCES

1. U.S. Nuclear Regulatory Commission Generic Letter, 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," January 31, 1996.

2. Westinghouse Report, WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," (Latest Approved Revision).

3. U.S. Nuclear Regulatory Commission Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.

4. Westinghouse Calculation Note, APP-RSX-M3C-026, Revision 0, "AP1000 Neutron Fast Fluence, DPA and Heating Rate Evaluation Long Term Power Distribution," March 2006.

5. Westinghouse Design Specification, APP-MV01-Z0-101, Revision 0, "AP1000 Reactor Pressure Vessel Design Specification," May 1, 2007.

6. Westinghouse Calculation Note, CN-EMT-02-13, Revision 1, "AP1000 Pressure Temperature Limit Curve Development," September 25, 2006.

7. ASTM E185-82, Annual Book of ASTM Standards, Section 12, Volume 12.02, *Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels*.

8. Code of Federal Regulations, 10CFR50, Appendix H, *Reactor Vessel Material Surveillance Program Requirements*, U.S. Nuclear Regulatory Commission, Washington, D.C. (Latest Approved Edition).

9. Code of Federal Regulations, 10CFR50, Appendix G, *Fracture Toughness Requirements*, U.S. Nuclear Regulatory Commission, Washington, D.C. (Latest Approved Edition).

10. Section XI of the 1998 Edition (through 2000 Addenda) ASME Boiler and Pressure Vessel Code, Appendix G, "Fracture Toughness Criteria for Protection Against Failure".

11. Code of Federal Regulations, 10CFR50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock," U.S. Nuclear Regulatory Commission, Washington, D.C. (Latest Approved Edition).