Southern Nuclear

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U.S. Nuclear Regulatory Commission
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Southern Nuclear Operating Company
Vogtle Electric Generating Plant Units 3&4
Pressure and Temperature Limits Report (PTLR)

Ladies and Gentlemen:

In accordance with Technical Specifications (TS) 5.6.4.c for Vogtle Electric Generating Plant (VEGP) Units 3 and 4, Southern Nuclear Operating Company (SNC) submits the VEGP Unit 3 Pressure and Temperature Limits Report (PTLR), Revision 0, and the VEGP Unit 4 PTLR, Revision 0.

Enclosure 1 provides the VEGP Unit 3 PTLR, Revision 0.

Enclosure 2 provides the VEGP Unit 4 PTLR, Revision 0.

This letter contains no regulatory commitments. This letter has been reviewed and determined not to contain security-related information.

If you have any questions, please contact Amy Chamberlain at 205.992.6361.

Respectfully submitted,

Brian H. Whitley
Regulatory Affairs Director
Southern Nuclear Operating Company

Enclosures:

1) Vogtle Electric Generating Plant (VEGP) Unit 3 Pressure and Temperature Limits Report (PTLR), Revision 0

2) Vogtle Electric Generating Plant (VEGP) Unit 4 Pressure and Temperature Limits Report (PTLR), Revision 0

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Southern Nuclear Operating Company

ND-21-0366

Enclosure 1

Vogtle Electric Generating Plant (VEGP) Unit 3

Pressure and Temperature Limits Report (PTLR), Revision 0

(This Enclosure consists of 18 pages, including this cover page)

Vogtle Electric Generating Plant (VEGP) Unit 3

Pressure and Temperature Limits Report (PTLR)

Revision 0

April 2021

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1.0 RCS PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

This PTLR has been prepared in accordance with the requirements of Technical Specification (TS) 5.6.4. 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, are contained in this PTLR. Revisions to the PTLR shall be provided to the U.S. Nuclear Regulatory Commission (NRC) after issuance. Note that these PTLR limits are consistent with, and bounded by, the generic AP1000 PTLR limits [18]. Reference 17 utilized site-specific material properties as inputs to validate that the P/T limit curves in Reference 18 remained bounding for VEGP Units 3 and 4.

This PTLR follows the guidelines set forth in U.S. NRC Generic Letter 96-03 [1].

2.0 OPERATING LIMITS

2.1 RCS PRESSURE AND TEMPERATURE LIMITS

The RCS P/T limits for LCO 3.4.3, presented in Figures 1 and 2 and tabulated in Tables 1 and 2, were developed using the NRC-approved methodology in WCAP-14040-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 RTNDT) 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 APP-RXS-M3C-026 [4] and APP-MV01-Z0-101 [5], and are summarized in Tables 3 through 5. Calculated ART values, used in the original development of the P/T limits, are documented in APP-RXS-M3C-012 [6]. These ART values are conservative for use as summarized in Table 6 and APP-RXS-M3C-033 [16]. The inputs and calculations are discussed in Section 4.0.

2.2 LOW TEMPERATURE OVERPRESSURE PROTECTION (LTOP) SYSTEM

Low Temperature Overpressure Protection (LTOP) is required to provide overpressure protection when any RCS cold leg temperature is $\leq$275˚F with the reactor vessel head on as required by Technical Specifications and APP-RCS-M3-001 [13]. This temperature is above the minimum LTOP enable temperature calculated using ASME Code Case N-641 (See Reference [2]).

A method of LTOP of the reactor vessel is provided by two parallel relief valves in the Normal Residual Heat Removal System (RNS) suction line inside containment. The larger RNS pump suction line relief valve (RNS-PL-V021) has a lift setpoint of 500 psig with a full open pressure of 550 psig [14]. The smaller relief valve (RNS-PL-V020) has a lift setpoint of 470 psig with a full open pressure of 517 psig [15]. The set pressure of RNS-PL-V020 is lower than the set pressure for RNS-PL-V021 so that the RNS-PL-V020 valve lifts first, and it is intended to be the only relief valve to lift during pressure transients that can be relieved with its capacity.

The lift setpoints were determined using the methodology in WCAP-14040-A [2], and the validation of the adequate capacity and set pressures of the valves is in APP-RNS-M3C-002 [12].

The maximum allowable RNS pump suction line relief valve lift setpoints are derived by analysis of the LTOP design basis Mass Input (MI) and the Heat Input (HI) transients. Operation with RNS pump suction line relief valves lift setpoints of less than or equal to the maximum allowable setpoint, ensures that the allowable steady state pressure temperature limit shown in Table 2 [6] will not be violated with consideration for: (1) relief valves' set pressure tolerances and accumulations; and (2) effects of reactor coolant pump operation.

To ensure mass and heat input transients more severe than those assumed cannot occur, Technical Specifications and operating procedures place limitations on coolant input capabilities and RCP operation during the appropriate LTOP modes. The magnitude of the maximum mass injection rate is controlled by isolating accumulators

and closing the Chemical and Volume Control System (CVS) makeup line containment isolation valve, CVS-PL-V091, to limit the mass injection rate to the relieving capacity of RNS-PL-V020. The magnitude of the maximum heat injection rate is also administratively controlled. To limit the heat injection rate, the reactor coolant pumps are started at less than 25% speed and the steam generator temperatures must be less than or equal to 50°F higher than each of the cold leg temperatures prior to pump start [12, 13]. The required relieving rate for this transient exceeds the capacity of RNS-PL-V020, and causes lifting of the higher capacity relief valve RNS-PL-V021. Both relief valves will lift and maintain the peak pressure below the limit in Table 2.

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 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

Reference 17, Appendix A, provides less restrictive VEGP Units 3 and 4 specific updated tables that reflect the plant-specific material properties; however, the generic AP1000 tables remain bounding and are utilized as the approved plant-specific data tables.

Table 3 contains the elemental chemistry limits for the 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 56 EFPY, along with the calculated fluence values at the 1/4T position (vessel wall quarter thickness from the inside surface) and 3/4T position (vessel wall quarter thickness from the outside surface).

Table 6 contains the calculations of the ARTs at the 1/4T position and 3/4T position, which are bounded by the ART values used in the determination of the P/T limit curves for normal operation [9, 10]. Note that the 56 EFPY ART values in Table 6 are lower than those utilized in the original development of the P/T limit curves, which are listed on Figures 1 and 2. The Figure 1 and 2 54 EFPY ARTs remain acceptable for use, since they bound the values in Table 6 for 56 EFPY.

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 ASTM E185 [7]. The maximum ΔRT_{NDT} value is calculated to be 109.0°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 78.4°F for the beltline forgings and 145.0°F or less for the beltline circumferential welds, and are in compliance with 10CFR50.61 [11].

Table 8 contains the conservative projections of USE for the reactor vessel beltline materials to show compliance with the requirements of 10CFR50, Appendix G [9].

<u>MATERIAL PROPERTY BASIS</u>

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

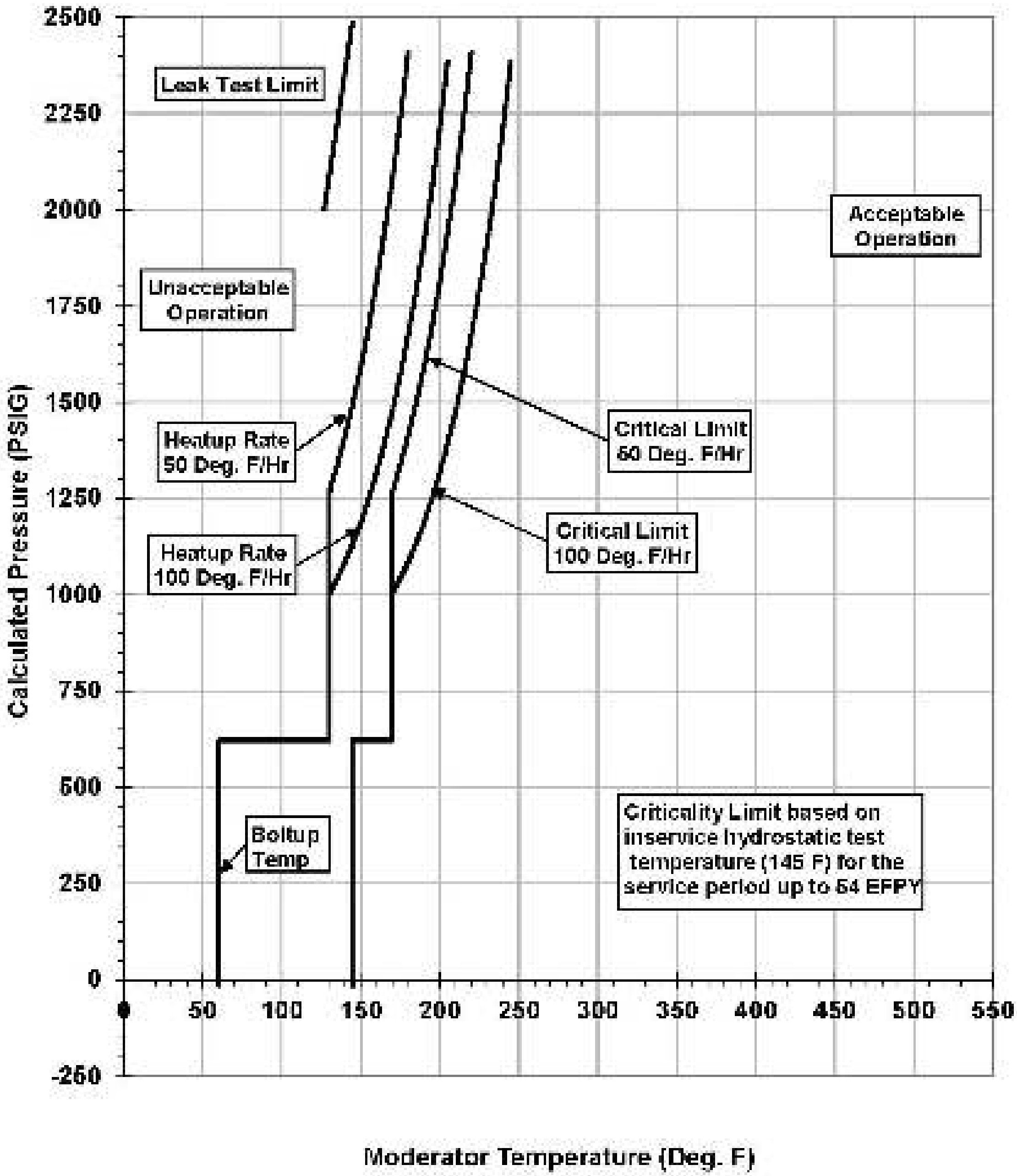

Figure 1: Reactor Coolant System Heatup Limitations (Maximum Heatup Rate of 100°F/hr) Applicable to 54 EFPY (without Margins for Instrumentation Errors)

(Plotted Data provided on Table 1)

MATERIAL PROPERTY BASIS

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

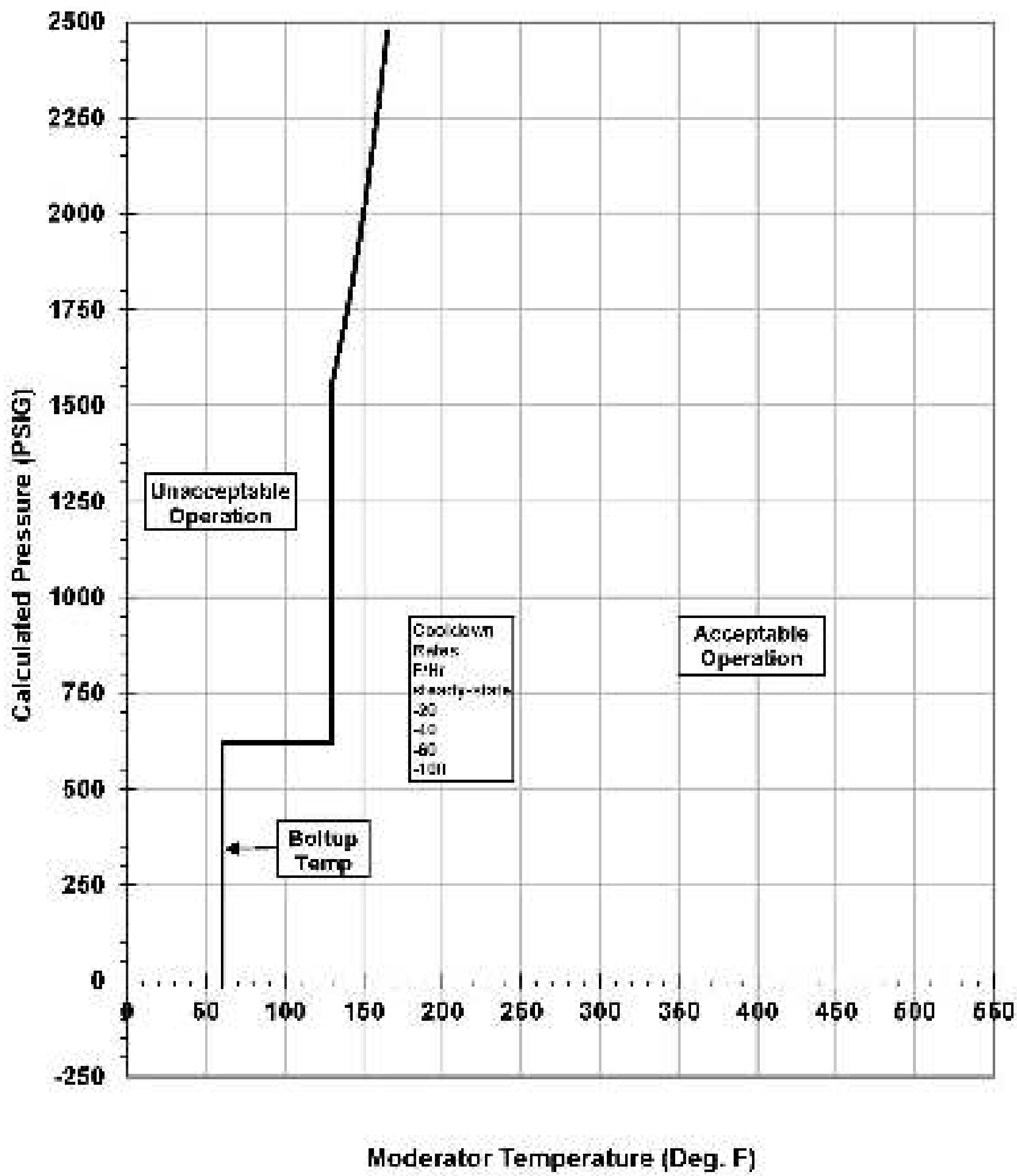

Figure 2: Reactor Coolant System Cooldown Limitations (Maximum Cooldown Rate of 100°F/hr) Applicable to 54 EFPY (without Margins for Instrumentation Error)

(Plotted Data provided on Table 2)

Table 1: RCS Heatup Limits at 54 EFPY

(without uncertainties for instrumentation errors)

50°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	-14.7	145	-14.7	60	-14.7	145	-14.7	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	-14.7	60	-14.7	60	-14.7	60	-14.7	60	-14.7
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	0.85	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 [5, 6, and 16]. 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: 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 values for initial RT_{NDT} are the basis for the ART calculations used to determine the P/T limit curves. These 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 56 EFPY

Reactor Vessel Materials	Surface Fluence[(a)]	1/4T Fluence[(b)]	3/4T Fluence[(c)]
Beltline Forgings	7.32 x 10^{19} n/cm^2	4.42 x 10^{19} n/cm^2	1.61 x 10^{19} n/cm^2
Upper Circumferential Weld	2.24 x 10^{19} n/cm^2	1.35 x 10^{19} n/cm^2	0.494 x 10^{19} n/cm^2
Lower Circumferential Weld	3.54 x 10^{19} n/cm^2	2.14 x 10^{19} n/cm^2	0.780 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 1/4T 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 3/4T 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 56 EFPY

1/4T Position ART Calculations						
Reactor Vessel Materials	**CF (°F)**	**1/4T FF[(a)]**	**IRT_{NDT}[(b)] (°F)**	**ΔRT_{NDT}[(c)] (°F)**	**Margin[(d)] (°F)**	**ART[(e)] (°F)**
Beltline Forgings	37	1.378	-10	51.0	34.0	75.0
Upper Circumferential Weld	82	1.084	-20	88.9	56.0	124.9
Lower Circumferential Weld	82	1.207	-20	98.9	56.0	134.9
3/4T Position ART Calculations						
Reactor Vessel Materials	**CF (°F)**	**3/4T FF[(a)]**	**IRT_{NDT}[(b)] (°F)**	**ΔRT_{NDT}[(c)] (°F)**	**Margin[(d)] (°F)**	**ART[(e)] (°F)**
Beltline Forgings	37	1.132	-10	41.9	34.0	65.9
Upper Circumferential Weld	82	0.803	-20	65.9	56.0	101.9
Lower Circumferential Weld	82	0.930	-20	76.3	56.0	112.3

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). 1/4T and 3/4T fluence values are provided in Table 5.

(b) Initial RT_{NDT} values are the 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} σ_I is set equal to 0°F for the calculations since the initial RT_{NDT} values are to be measured values per [5]. 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]. Note that the limiting ART values used in the development of the P/T limit curves conservatively bound the values shown in this table. The ART values utilized for P/T limit curve development are conservative values initially calculated for the forgings (90°F for 1/4T and 82°F for 3/4T) and consider 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 56 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	1.470	-10	54.4	34.0	78.4
Upper Circumferential Weld	82	1.218	-20	99.9	56.0	135.9
Lower Circumferential Weld	82	1.329	-20	109.0	56.0	145.0

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 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}. σ_I is set equal to 0°F for the calculations since the initial RT_{NDT} values are to be measured values per [5]. 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 56 EFPY

Reactor Vessel Materials	Initial USE [(a)]	1/4T Fluence[(b)]	Position 1.2 USE Decrease[(c)]	Projected EOL USE[(d)]
Beltline Forgings	> 75 ft-lbs	4.42×10^{19} n/cm^2	27%	54.8 ft-lbs
Upper Circumferential Weld	> 75 ft-lbs	1.35×10^{19} n/cm^2	26%	55.5 ft-lbs
Lower Circumferential Weld	> 75 ft-lbs	2.14×10^{19} n/cm^2	29%	53.3 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 1×10^{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 1/4T neutron fluence values in Table 5.
(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 (56 EFPY) USE values are based on the limiting copper chemistry allowed (see Table 3) and an assumed 75 ft-lb USE for the unirradiated material (the minimum allowed). The predicted decrease in USE for the reactor vessel materials conservatively uses the 0.10 wt% Cu content line for base and weld 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

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2. Westinghouse Report, WCAP-14040-A, Rev. 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves WOG Programs: MUHP-5073 MUHP-3073," May 30, 2004.

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-RXS-M3C-026, Revision 1, "**AP1000** Neutron Fast Fluence, DPA and Heating Rate Evaluation Long Term Power Distribution," April 13, 2012.

5. Westinghouse Design Specification, APP-MV01-Z0-101, Revision 15, "Design Specification for AP1000 Reactor Vessel for System: Reactor Coolant System (RCS)," May 12, 2017.

6. Westinghouse Calculation Note, APP-RXS-M3C-012, Revision 1, "**AP1000** Pressure and Temperature Limit Curve Development," October 31, 2016.

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., Federal Register, Volume 60, No. 243, dated December 19, 1995.

9. Code of Federal Regulations, 10CFR50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Washington, D.C. Federal Register, Volume 60, No. 243, dated December 19, 1995.

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

11. Code of Federal Regulations, 10CFR50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock," U.S. Nuclear Regulatory Commission, Washington, D.C., Federal Register, Volume 60, No. 243, dated December 19, 1995.

12. Westinghouse Calculation Note, APP-RNS-M3C-002, Revision 5 "**AP1000** LTOPS Analyses / Normal RNR Relief Valve Sizing Evaluation," June 13, 2017.

13. Westinghouse Specification, APP-RCS-M3-001, Revision 11, “Reactor Coolant System, System Specification Document,” September 5, 2017.

14. Westinghouse Datasheet, APP-PV16-Z0D-105, Revision 4, “PV16 Datasheet 105,” October 24, 2011.

15. Westinghouse Datasheet, APP-PV16-Z0D-211, Revision 1, “PV16 Datasheet 211,” October 6, 2016.

16. Westinghouse Calculation Note, APP-RXS-M3C-033, Revision 0, “**AP1000**® Reactor Vessel Integrity Evaluations,” April, 2018

17. Westinghouse Document SV0-RXS-GLR-001, Revision 0, “Validation of the AP1000 Pressure-Temperature Limits Report (PTLR) for Use at Vogtle Units 3 and 4,” May 2020.

18. Westinghouse Document APP-RXS-Z0R-001, Revision 3, “**AP1000** Generic Pressure Temperature Limits Report,” December 2017.

Southern Nuclear Operating Company

ND-21-0366

Enclosure 2

Vogtle Electric Generating Plant (VEGP) Unit 4

Pressure and Temperature Limits Report (PTLR), Revision 0

(This Enclosure consists of 18 pages, including this cover page)

Vogtle Electric Generating Plant (VEGP) Unit 4

Pressure and Temperature Limits Report (PTLR)

Revision 0

April 2021

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1.0 RCS PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

This PTLR has been prepared in accordance with the requirements of Technical Specification (TS) 5.6.4. 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, are contained in this PTLR. Revisions to the PTLR shall be provided to the U.S. Nuclear Regulatory Commission (NRC) after issuance. Note that these PTLR limits are consistent with, and bounded by, the generic AP1000 PTLR limits [18]. Reference 17 utilized site-specific material properties as inputs to validate that the P/T limit curves in Reference 18 remained bounding for VEGP Units 3 and 4.

This PTLR follows the guidelines set forth in U.S. NRC Generic Letter 96-03 [1].

2.0 OPERATING LIMITS

2.1 RCS PRESSURE AND TEMPERATURE LIMITS

The RCS P/T limits for LCO 3.4.3, presented in Figures 1 and 2 and tabulated in Tables 1 and 2, were developed using the NRC-approved methodology in WCAP-14040-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 RTNDT) 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 APP-RXS-M3C-026 [4] and APP-MV01-Z0-101 [5], and are summarized in Tables 3 through 5. Calculated ART values, used in the original development of the P/T limits, are documented in APP-RXS-M3C-012 [6]. These ART values are conservative for use as summarized in Table 6 and APP-RXS-M3C-033 [16]. The inputs and calculations are discussed in Section 4.0.

2.2 LOW TEMPERATURE OVERPRESSURE PROTECTION (LTOP) SYSTEM

Low Temperature Overpressure Protection (LTOP) is required to provide overpressure protection when any RCS cold leg temperature is $\leq$275˚F with the reactor vessel head on as required by Technical Specifications and APP-RCS-M3-001 [13]. This temperature is above the minimum LTOP enable temperature calculated using ASME Code Case N-641 (See Reference [2]).

A method of LTOP of the reactor vessel is provided by two parallel relief valves in the Normal Residual Heat Removal System (RNS) suction line inside containment. The larger RNS pump suction line relief valve (RNS-PL-V021) has a lift setpoint of 500 psig with a full open pressure of 550 psig [14]. The smaller relief valve (RNS-PL-V020) has a lift setpoint of 470 psig with a full open pressure of 517 psig [15]. The set pressure of RNS-PL-V020 is lower than the set pressure for RNS-PL-V021 so that the RNS-PL-V020 valve lifts first, and it is intended to be the only relief valve to lift during pressure transients that can be relieved with its capacity.

The lift setpoints were determined using the methodology in WCAP-14040-A [2], and the validation of the adequate capacity and set pressures of the valves is in APP-RNS-M3C-002 [12].

The maximum allowable RNS pump suction line relief valve lift setpoints are derived by analysis of the LTOP design basis Mass Input (MI) and the Heat Input (HI) transients. Operation with RNS pump suction line relief valves lift setpoints of less than or equal to the maximum allowable setpoint, ensures that the allowable steady state pressure temperature limit shown in Table 2 [6] will not be violated with consideration for: (1) relief valves' set pressure tolerances and accumulations; and (2) effects of reactor coolant pump operation.

To ensure mass and heat input transients more severe than those assumed cannot occur, Technical Specifications and operating procedures place limitations on coolant input capabilities and RCP operation during the appropriate LTOP modes. The magnitude of the maximum mass injection rate is controlled by isolating accumulators

and closing the Chemical and Volume Control System (CVS) makeup line containment isolation valve, CVS-PL-V091, to limit the mass injection rate to the relieving capacity of RNS-PL-V020. The magnitude of the maximum heat injection rate is also administratively controlled. To limit the heat injection rate, the reactor coolant pumps are started at less than 25% speed and the steam generator temperatures must be less than or equal to 50°F higher than each of the cold leg temperatures prior to pump start [12, 13]. The required relieving rate for this transient exceeds the capacity of RNS-PL-V020, and causes lifting of the higher capacity relief valve RNS-PL-V021. Both relief valves will lift and maintain the peak pressure below the limit in Table 2.

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 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 5×10^{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

Reference 17, Appendix A, provides less restrictive VEGP Units 3 and 4 specific updated tables that reflect the plant-specific material properties; however, the generic AP1000 tables remain bounding and are utilized as the approved plant-specific data tables.

Table 3 contains the elemental chemistry limits for the 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 56 EFPY, along with the calculated fluence values at the 1/4T position (vessel wall quarter thickness from the inside surface) and 3/4T position (vessel wall quarter thickness from the outside surface).

Table 6 contains the calculations of the ARTs at the 1/4T position and 3/4T position, which are bounded by the ART values used in the determination of the P/T limit curves for normal operation [9, 10]. Note that the 56 EFPY ART values in Table 6 are lower than those utilized in the original development of the P/T limit curves, which are listed on Figures 1 and 2. The Figure 1 and 2 54 EFPY ARTs remain acceptable for use, since they bound the values in Table 6 for 56 EFPY.

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 ASTM E185 [7]. The maximum ΔRT_{NDT} value is calculated to be 109.0°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 78.4°F for the beltline forgings and 145.0°F or less for the beltline circumferential welds, and are in compliance with 10CFR50.61 [11].

Table 8 contains the conservative projections of USE for the reactor vessel beltline materials to show compliance with the requirements of 10CFR50, Appendix G [9].

<u>MATERIAL PROPERTY BASIS</u>

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

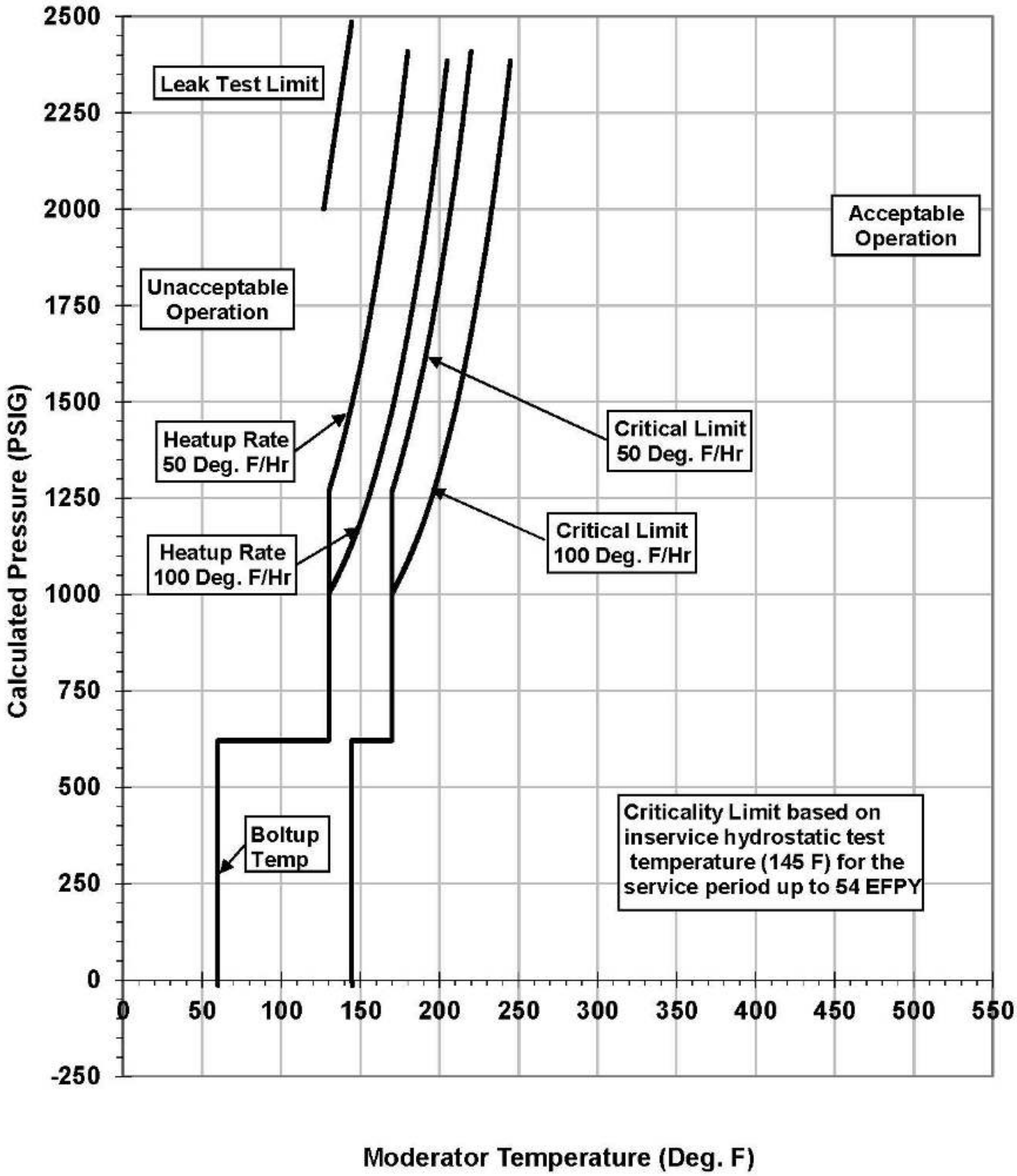

Figure 1: Reactor Coolant System Heatup Limitations (Maximum Heatup Rate of 100°F/hr) Applicable to 54 EFPY (without Margins for Instrumentation Errors)

(Plotted Data provided on Table 1)

MATERIAL PROPERTY BASIS

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

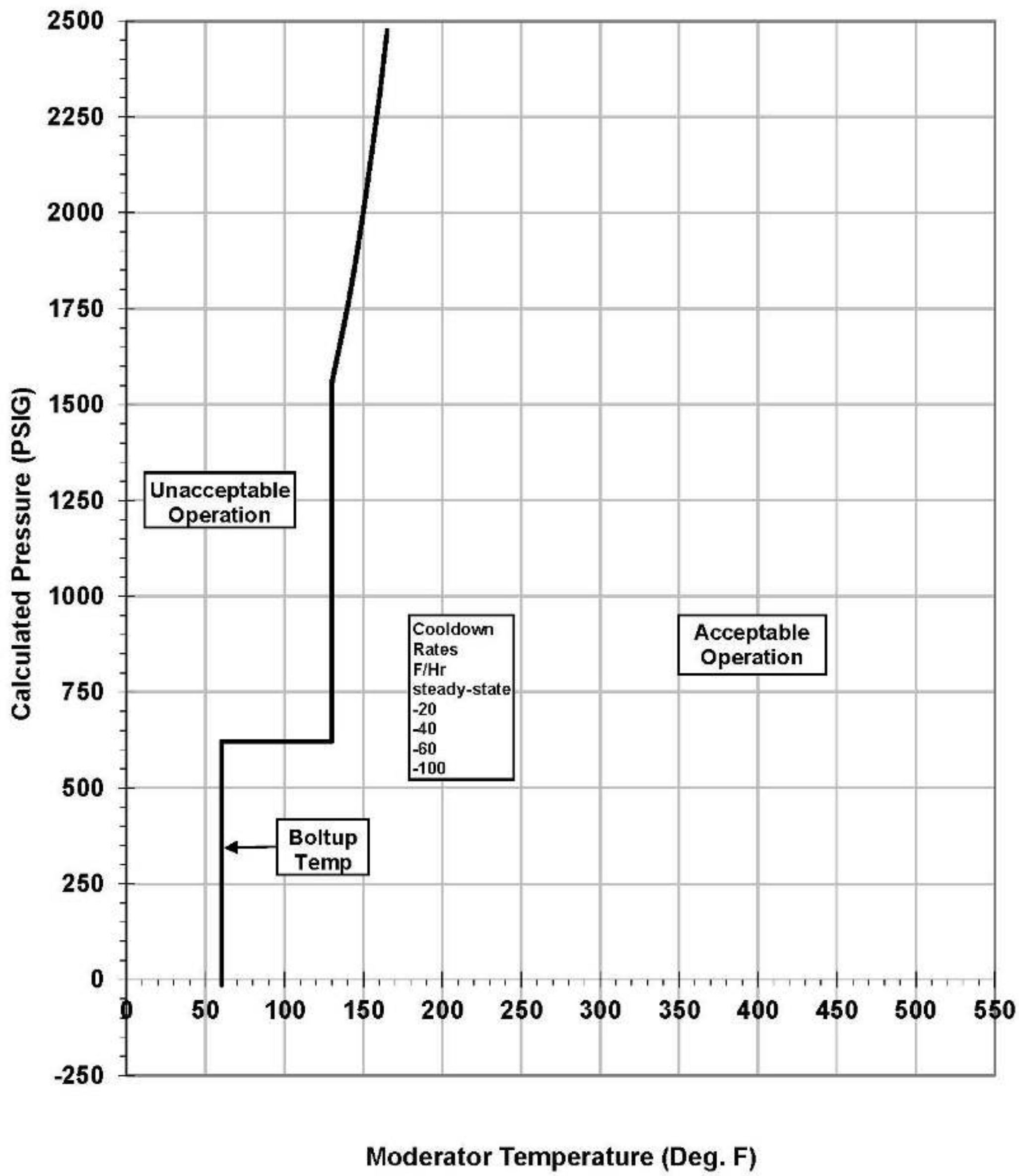

Figure 2: Reactor Coolant System Cooldown Limitations (Maximum Cooldown Rate of 100°F/hr) Applicable to 54 EFPY (without Margins for Instrumentation Error)

(Plotted Data provided on Table 2)

Table 1: RCS Heatup Limits at 54 EFPY

(without uncertainties for instrumentation errors)

50°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	-14.7	145	-14.7	60	-14.7	145	-14.7	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	-14.7	60	-14.7	60	-14.7	60	-14.7	60	-14.7
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	0.85	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 [5, 6, and 16]. 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: 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 values for initial RT_{NDT} are the basis for the ART calculations used to determine the P/T limit curves. These 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 1×10^{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 56 EFPY

Reactor Vessel Materials	Surface Fluence[(a)]	1/4T Fluence[(b)]	3/4T Fluence[(c)]
Beltline Forgings	7.32×10^{19} n/cm^2	4.42×10^{19} n/cm^2	1.61×10^{19} n/cm^2
Upper Circumferential Weld	2.24×10^{19} n/cm^2	1.35×10^{19} n/cm^2	0.494×10^{19} n/cm^2
Lower Circumferential Weld	3.54×10^{19} n/cm^2	2.14×10^{19} n/cm^2	0.780×10^{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 1/4T 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 3/4T 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 56 EFPY

1/4T Position ART Calculations						
Reactor Vessel Materials	**CF (°F)**	**1/4T FF(a)**	**IRT_{NDT}(b) (°F)**	**ΔRT_{NDT}(c) (°F)**	**Margin(d) (°F)**	**ART(e) (°F)**
Beltline Forgings	37	1.378	-10	51.0	34.0	75.0
Upper Circumferential Weld	82	1.084	-20	88.9	56.0	124.9
Lower Circumferential Weld	82	1.207	-20	98.9	56.0	134.9
3/4T Position ART Calculations						
Reactor Vessel Materials	**CF (°F)**	**3/4T FF(a)**	**IRT_{NDT}(b) (°F)**	**ΔRT_{NDT}(c) (°F)**	**Margin(d) (°F)**	**ART(e) (°F)**
Beltline Forgings	37	1.132	-10	41.9	34.0	65.9
Upper Circumferential Weld	82	0.803	-20	65.9	56.0	101.9
Lower Circumferential Weld	82	0.930	-20	76.3	56.0	112.3

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). 1/4T and 3/4T fluence values are provided in Table 5.

(b) Initial RT_{NDT} values are the 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} σ_I is set equal to 0°F for the calculations since the initial RT_{NDT} values are to be measured values per [5]. 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]. Note that the limiting ART values used in the development of the P/T limit curves conservatively bound the values shown in this table. The ART values utilized for P/T limit curve development are conservative values initially calculated for the forgings (90°F for 1/4T and 82°F for 3/4T) and consider 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 56 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	1.470	-10	54.4	34.0	78.4
Upper Circumferential Weld	82	1.218	-20	99.9	56.0	135.9
Lower Circumferential Weld	82	1.329	-20	109.0	56.0	145.0

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 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}. σ_I is set equal to 0°F for the calculations since the initial RT_{NDT} values are to be measured values per [5]. 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 56 EFPY

Reactor Vessel Materials	Initial USE (a)	1/4T Fluence(b)	Position 1.2 USE Decrease(c)	Projected EOL USE(d)
Beltline Forgings	> 75 ft-lbs	4.42×10^{19} n/cm^2	27%	54.8 ft-lbs
Upper Circumferential Weld	> 75 ft-lbs	1.35×10^{19} n/cm^2	26%	55.5 ft-lbs
Lower Circumferential Weld	> 75 ft-lbs	2.14×10^{19} n/cm^2	29%	53.3 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 1×10^{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 1/4T neutron fluence values in Table 5.

(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 (56 EFPY) USE values are based on the limiting copper chemistry allowed (see Table 3) and an assumed 75 ft-lb USE for the unirradiated material (the minimum allowed). The predicted decrease in USE for the reactor vessel materials conservatively uses the 0.10 wt% Cu content line for base and weld 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.

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