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STP Units 3 & 4	SPECIFICATION	
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FOR APPROVAL

South Texas Project – Nuclear Operating Company
STP Units 3 & 4

Title : Pressure and Temperature Limits Report (PTLR) for 60 Effective Full-Power Years (EFPY)

Customer Name	STPNOC
Project Name	STP-34
Item Name	Reactor Pressure Vessel(RPV)
Item Number	U7-RPV-RPV-001
Job Number	B03974
Applicable Plant	STP units 3 & 4

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2	Jul. 20, 2010	Revised the PT curve, using the calculation result of ANSYS	K. Matsunaga Jul. 20, 2010	K. Matsunaga Jul. 20, 2010	A. Mori Jul. 20, 2010

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Jul.23, 2009	Vessels & Components Engineering Group	K. Matsunaga Jul.23, 2009	K. Matsunaga Jul.23, 2009	A. Mori Jul.23, 2009	RS-5134511

Record of Revisions

Rev.	Issue date	Changed Place and Content	Organization	Approved by	Reviewed by	Prepared by
0	See cover page	Initial issue	Vessels & Components Engineering Group	See cover page	See cover page	See cover page
1	Jul.24, 2009	Revise to use more conservative RTNDT value for the upper vessel region of Curves A and B (Page 11, 12, 14, 15)	Vessels & Components Engineering Group	K. Matsunaga Jul.24, 2009	K. Matsunaga Jul.24, 2009	A. Mori Jul.24, 2009
2	See cover page	Revised the PT curve, using the calculation result of ANSYS	Vessels & Components Engineering Group	See cover page	See cover page	See cover page

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1.0 Purpose

The purpose of the South Texas Project (STP) Units 3 and 4 Pressure and Temperature Limits Report (PTLR) is to present operating limits relating to:

1) Reactor Coolant System (RCS) Pressure versus Temperature limits during Heatup, Cooldown and Hydrostatic/Class 1 Leak Testing;
2) RCS Heatup and Cooldown rates;
3) Reactor Pressure Vessel (RPV) head flange boltup temperature limits.

This report has been prepared in accordance with the requirements of Reference [6.1], Licensing Topical Report SIR-05-044-A, Revision 0.

2.0 Applicability

This report is applicable to the STP Units 3 and 4 RPVs for 60 years of operation at 100% rated power, or 60 Effective Full-Power Years (EFPY). The following portion of the STP COLA, Part 2, Tier 2, Chapter 16, "Technical Specifications" is affected by the information contained in this report:

- 3.4.9 RCS Pressure and Temperature (P/T) Limits

The STP Units 3 and 4 Reactor Vessel Pressure and Temperature Limits for 60 EFPY have been developed per Reference [6.15]. Future revisions of the PTLR must be revised per the 10 CFR 50.59 review process, as applicable.

3.0 Methodology

The limits in this report were derived as follows:

1) The methodology used is in accordance with Reference [6.1], which has been approved for BWR use by the NRC.

2) The neutron fluence used for the P-T curves is the value specified in ABWR DCD Tier 2 5.3.3.2, which was confirmed to be conservative by a calculation performed in accordance with NRC Regulatory Guide 1.190 (Reference [6.5]), as documented in Reference [6.2].

3) The adjusted reference temperature (ART) values for the limiting beltline materials are calculated in accordance with NRC Regulatory Guide 1.99, Revision 2 (Reference [6.6]), as documented in Reference [6.3].

4) The pressure and temperature limits were calculated in accordance with Reference [6.1], as documented in Reference [6.15].

5) This revision of the pressure and temperature limits is to incorporate the following changes:
   - Initial issue of PTLR.

Changes to the curves, limits, or parameters within this PTLR, based upon new irradiation fluence data of the RPV, or other plant design assumptions in the Updated Final Safety Analysis Report (UFSAR), can be made pursuant to 10 CFR 50.59, provided the above methodologies are utilized. The revised PTLR shall be submitted to the NRC upon issuance.

4.0 Operating Limits

The pressure-temperature (P-T) curves included in this report represent steam dome pressure versus minimum vessel coolant temperature and incorporate the appropriate non-beltline limits and irradiation embrittlement effects in the beltline region.

The operating limits for pressure and temperature are required for three categories of operation: (a) hydrostatic pressure tests and leak tests, referred to as Curve A; (b) core not critical operation, referred to as Curve B; and (c) core critical operation, referred to as Curve C. Each of these curves contains three P-T limits based on a consolidation of all RPV regions into three bounding evaluation regions: (1) the beltline, (2) the bottom head, and (3) the feedwater nozzle / upper vessel. The feedwater nozzle / upper vessel curve also bounds the RPV instrument (N12) nozzles, which are exposed to a fluence greater than 1×10^{17} n/cm^2 for 60 EFPY, as the stresses for the feedwater nozzle are higher, and the limiting ART value used for the feedwater nozzle / upper vessel region bounds the ART value calculated for the N12 nozzle location (Reference [6.3]).

Complete P-T curves were developed for 60 EFPY for STP Units 3 and 4, as documented in Reference [6.15]. The STP Units 3 and 4 P-T curves for 60 EFPY are provided in Figures 1 through 3, and a tabulation of the curves is included in Tables 1 through 3. The adjusted reference temperature (ART) tables for the STP Units 3 and 4 vessel beltline materials are shown in Table 4 for 60 EFPY (Reference [6.3]). The resulting P-T curves are based on the geometry, design and materials information for the STP Units 3 and 4 vessels with the following conditions:

- Heatup and Cooldown rate limit during Hydrostatic and Class 1 Leak Testing (Figure 1: Curve A): ≤ **25°F/hour**[1].
- Normal Operating Heatup and Cooldown rate limit (Figure 2: Curve B - non-nuclear heating, and Figure 3: Curve C - nuclear heating): ≤ **101°F/hour**[2].
- RPV flange and adjacent shell temperature limit: ≥ **60°F**.

5.0 Discussion

The adjusted reference temperature (ART) of the limiting beltline material is used to adjust the beltline P-T curves to account for irradiation effects. Regulatory Guide 1.99, Revision 2 (RG 1.99) provides the methods for determining the ART. The RG 1.99 methods for determining the limiting material and adjusting the P-T curves using ART are discussed in this section.

The vessel beltline copper (Cu) and nickel (Ni) values were obtained from the design specification for the vessel plate and weld materials planned for use to fabricate the STP Units 3 and 4 RPVs (Reference [6.3]). The Cu and Ni values were used with Tables 1 and 2 of RG 1.99 to determine a chemistry factor (CF) per Paragraph 1.1 of RG 1.99 for welds and plates, respectively.

The peak RPV inside surface (at clad/base metal interface) fluence used in the P-T curve evaluation for 60 EFPY is 6.00×10^{17} n/cm^2 for STP Units 3 and 4 from Reference [6.2], which was calculated using methods that comply with the guidelines of RG 1.190 (Reference [6.5]). This fluence value applies to the limiting beltline Lower (Core Beltline) Shell and adjacent weld

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[1] Interpreted as the temperature change in any 1-hour period is less than or equal to 25°F.

[2] Interpreted as the temperature change in any 1-hour period is less than or equal to 101°F.

material. The fluence value was adjusted for the Lower (Core Beltline) Shell and adjacent weld material based upon an attenuation factor of 0.663 for a postulated one-quarter thickness (1/4t) flaw. As a result, the 1/4t 60 EFPY fluence for the Lower (Core Beltline) Shell and adjacent weld material is 3.98×10^{17} n/cm^2 for STP Units 3 and 4.

The P-T curves for the core not critical and core critical operating conditions at a given EFPY apply for both the 1/4t and 3/4t locations. When combining pressure and thermal stresses, it is usually necessary to evaluate stresses at the 1/4t location (inside surface flaw) and the 3/4t location (outside surface flaw). This is because the thermal gradient tensile stress of interest is in the inner wall during cooldown and is in the outer wall during heatup. However, as a conservative simplification, the thermal gradient stress at the 1/4t location is assumed to be tensile for both heatup and cooldown. This results in the approach of applying the maximum tensile stress at the 1/4t location. This approach is conservative because irradiation effects cause the allowable toughness at 1/4t to be less than that at 3/4t for a given metal temperature. This approach causes no operational difficulties, since the BWR is at steam saturation conditions during normal operation, which is well within the P-T curve limits.

For the core not critical curve (Curve B) and the core critical curve (Curve C), the P-T curves specify a coolant heatup and cooldown temperature rate of ≤ **101°F/hr** for which the curves are applicable. However, the core not critical and the core critical curves were also developed to bound transients defined on the RPV thermal cycle diagram and the nozzle thermal cycle diagrams. For the hydrostatic pressure and leak test curve (Curve A), a coolant heatup and cooldown temperature rate of ≤ **25°F/hr** must be maintained. The P-T limits and corresponding limits of either Curve A or B may be applied, if necessary, while achieving or recovering from test conditions. So, although Curve A applies during pressure testing, the limits of Curve B may be conservatively used during pressure testing if the pressure test heatup/cooldown rate limits cannot be maintained.

The initial RT_{NDT}, the chemistry (weight-percent Cu and Ni), and adjusted reference temperature at the 1/4 thickness location for all RPV beltline materials significantly affected by fluence (i.e., fluence > 10^{17} n/cm^2 for E > 1 MeV) are shown in Table 4 for 60 EFPY. The initial RT_{NDT} values shown in Table 4 (from Reference [6.3]) were obtained from the design specification for the vessel plate and weld materials planned for use to fabricate the STP Units 3 and 4 RPVs (Reference [6.3]).

Appendix A of Reference [6.1] does not apply since STP Units 3 and 4 are not participants in the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP).

The only computer code used in the determination of the STP Units 3 and 4 pressure/temperature curves was the ANSYS (Release 11.0) finite element computer program for the feedwater nozzle (non-beltline) stresses. This analysis was performed to determine through-wall thermal and pressure stress distributions for the STP Units 3 and 4 RPV nozzles due to a thermal transient. The ANSYS program is controlled under the vendor's 10 CFR 50 Appendix B Quality Assurance Program for nuclear quality-related work. Benchmarking consistent with NRC GL 83-11, Supplement 1 (Reference 6.8) was performed. The verification and validation process compared ANSYS verification problem results provided by ANSYS Inc. with the results of the same problems run by the performer of the STP 3&4 stress calculation. No significant differences were found between the results provided by ANSYS and the results obtained by the STP 3&4 stress calculation performer. The documentation for the verification and validation process for Version 11.0 of ANSYS can be found in RPV Vendor Document Program Test (Validation) Report ANSYS 11.0, document No. 32820500-C80L500, Rev. 0.

The following inputs were used for the finite element analysis.

- With respect to operating conditions, stress distributions were developed based on the stress analysis results, considering the thermal transient conditions of service level A and B (normal and upset conditions) for the feedwater nozzle, which is determined to exhibit the most limiting stresses among the RPV nozzles under these conditions. Based on the stress analysis results, the event where the maximum stress occurs for the feedwater nozzle is after the reactor coolant temperature drops from 552°F to 376°F in 10 minutes during the turbine bypass transient.
- Heat transfer coefficients were calculated in the design basis stress report for the STP Units 3 & 4 feedwater nozzle and from a model of the heat transfer coefficient as a function of flow rate. The heat transfer coefficients were evaluated at flow rates that bound the actual operating conditions in the feedwater nozzle.
- A two-dimensional, axisymmetric finite element model of the feedwater nozzle was constructed using the same modeling techniques that were employed to evaluate the nozzles in the design basis stress report. The analysis was performed as a penetration in a sphere and not in a cylinder. To make up for this difference in geometry, a conversion factor of 2.0 times the cylinder radius was used to model the sphere. Material properties were evaluated during the transient event condition.

6.0 References

6.1 Structural Integrity Associates Report No. SIR-05-044-A, Revision 0, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," April 2007.

6.2 Westinghouse Letter No. LTR-REA-09-111, "Transmittal of Neutron Fluence Information for the Initial PTLR Submittal for South Texas 3 and 4," July 15, 2009.

6.3 Structural Integrity Associates, Inc. Calculation No. 0900495.301, Revision 0, "Task 1: ΔRT_{NDT} and ART Evaluation."

6.4 Structural Integrity Associates, Inc. Calculation No. 0900495.302, Revision 1, "Task 2: Develop P-T Curves."

6.5 U. S. Nuclear Regulatory Commission Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001.

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

6.7 Toshiba Project Document No. 7B11-D003-1101-03, Revision 2, "Input Data for Pressure-Temperature Limits of RPV."

6.8 U. S. Nuclear Regulatory Commission, Generic Letter 83-11, Supplement 1, "Licensee Qualification for Performing Safety Analyses," June 24, 1999.

6.9 Not used

6.10 Part 50 of Title 10 of the Code of Federal Regulations, Appendix G, "Fracture Toughness Requirements," January 2005.

6.11 Part 50 of Title 10 of the Code of Federal Regulations, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," January 2005.

6.12 Toshiba Corporation Topical Report No. UTLR-0003, Revision 0, "Reactor Pressure Vessel Material Surveillance Program," April 2009.

6.13 ASTM E185-82^{e2}, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels, E 706 (IF)."

6.14 ASTM E23-07a^{e1}, "Standard Test Methods for Notched Bar Impact Testing of Metallic Materials."

6.15 Toshiba Project Document No. 7B11-D003-3809-17, Revision 0, "Develop P-T Curves."

Figure 1: STP Units 3 and 4 Pressure Test (Curve A) P-T Curve (60 EFPY) (Reference [6.15])

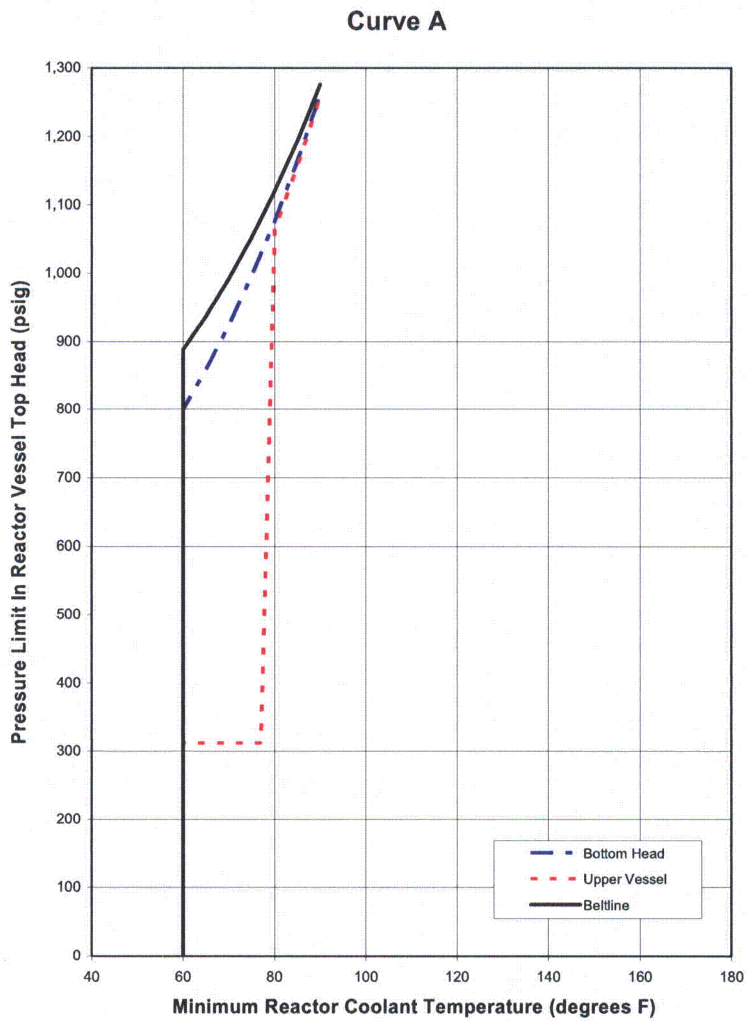

Figure 2: STP Units 3 and 4 Core Not Critical (Curve B) P-T Curve (60 EFPY) (Reference [6.15])

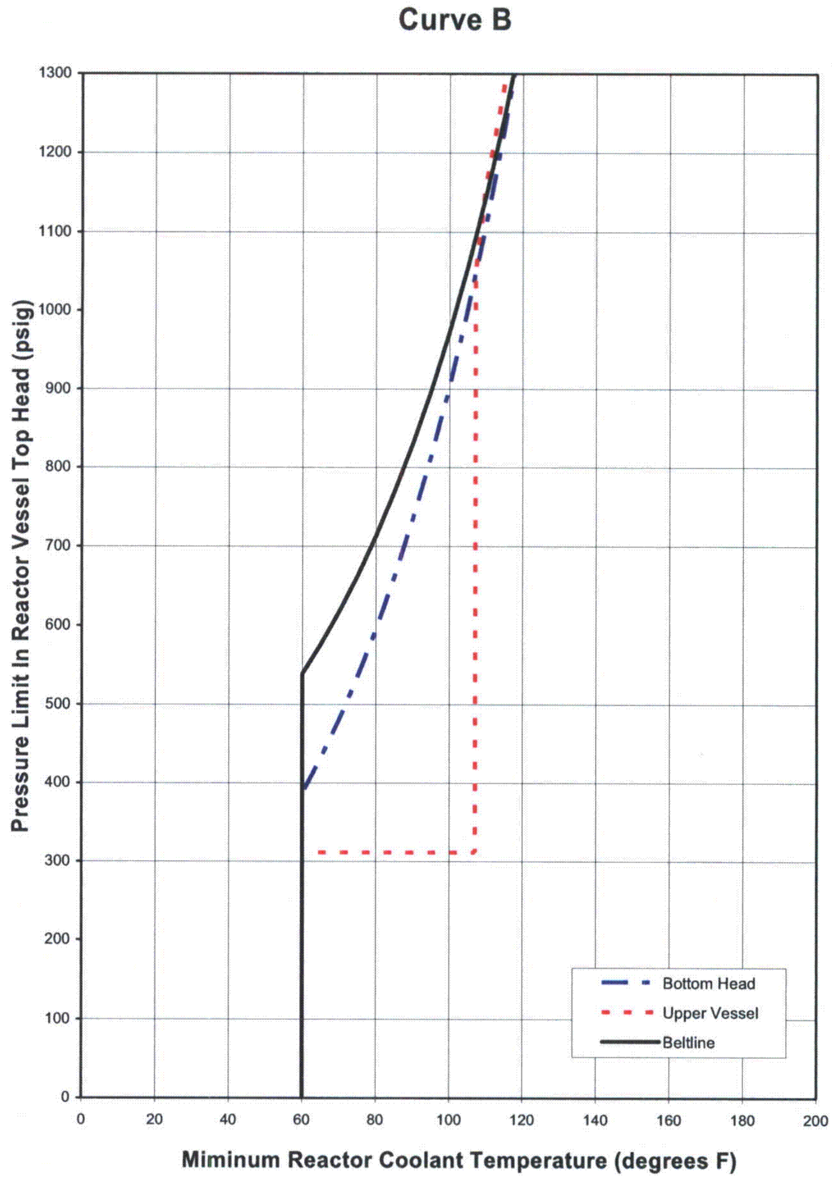

Figure 3: STP Units 3 and 4 Core Critical (Curve C) P-T Curve (60 EFPY) (Reference [6.15])

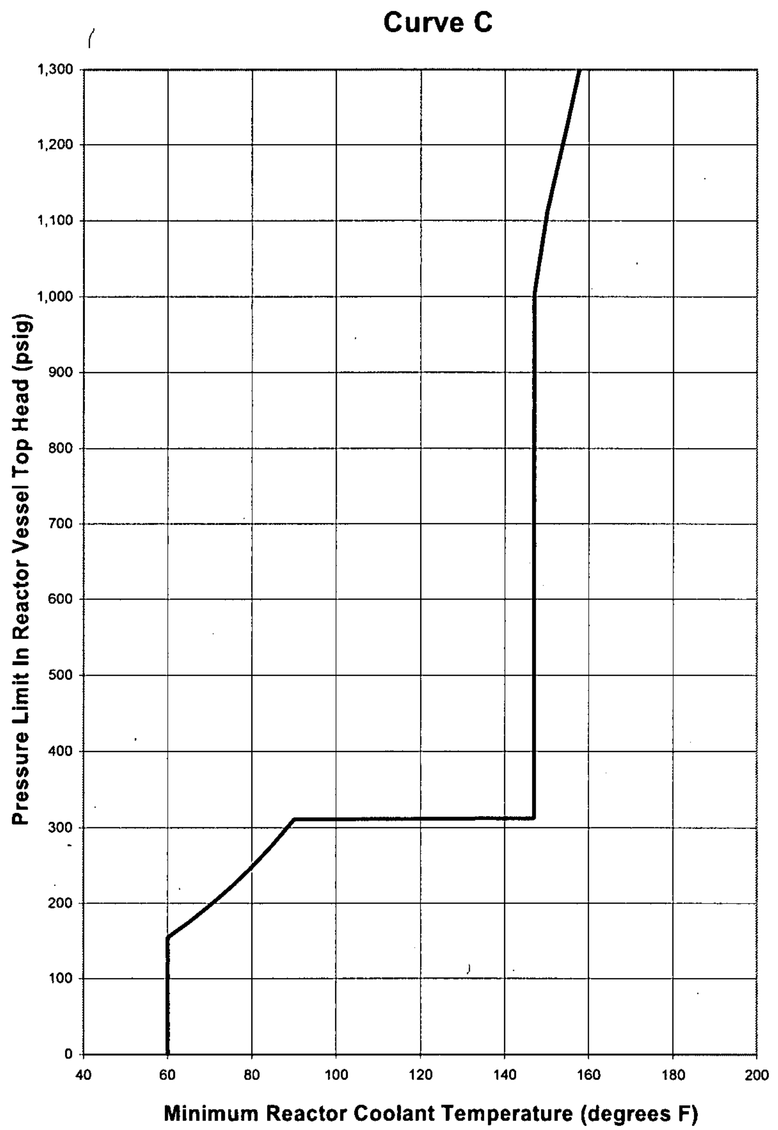

Table 1: STP Units 3 and 4 Pressure Test (Curve A) P-T Curve (60 EFPY) (Reference [6.15])

BOTTOM HEAD		UPPER VESSEL		BELTLINE	
Temperature (°F)	Pressure (psig)	Temperature (°F)	Pressure (psig)	Temperature (°F)	Pressure (psig)
60	0	60	0	60	0
60	800	60	313	60	888
65	859	77	313	65	938
70	924	77	313	70	993
75	997	77	313	75	1053
80	1076	80	1065	80	1120
85	1164	85	1157	85	1194
90	1262	90	1259	90	1276

Table 2: STP Units 3 and 4 Core Not Critical (Curve B) P-T Curve (60 EFPY) (Reference [6.15])

BOTTOM HEAD		UPPER VESSEL		BELTLINE	
Temperature (°F)	Pressure (psig)	Temperature (°F)	Pressure (psig)	Temperature (°F)	Pressure (psig)
60	0	60	0	60	0
60	387	60	313	60	539
65	432	107	313	65	577
70	481	107	503	70	618
75	535	107	559	75	663
80	594	107	622	80	713
85	660	107	691	85	769
90	733	107	768	90	830
95	814	107	852	95	898
100	903	107	945	100	973
105	1002	107	1048	105	1055
110	1111	110	1162	110	1147
115	1231	115	1288	115	1248
120	1364	120	1427	120	1359

Table 3: STP Units 3 and 4 Core Critical (Curve C) P-T Curve (60 EFPY) (Reference [6.15])

Temperature (°F)	Pressure (psig)
60	0
60	93
60	106
60	121
60	137
60	155
65	175
70	197
75	222
80	248
85	278
90	311
147	313
147	387
147	432
147	481
147	535
147	594
147	660
147	733
147	814
147	903
147	1002
150	1111
155	1231
160	1359

Table 4: STP Units 3 and 4 ART Calculations (60 EFPY) (Reference [6.3])

(NOTE: This table covers all RPV materials with an exposed fluence, E > 1 MeV, of greater than 1.0×10^{17} n/cm^2.)

Location	Initial RT_{NDT} * (°F)	Chemistry * Cu (wt %)	Chemistry * Ni (wt %)	Chemistry Factor (°F)	Adjustments for 1/4T ΔRT_{NDT} (°F)	Margin Terms σ_Δ (°F)	Margin Terms σ_i (°F)	ART (°F)
Lower (Core Beltline) Shell	-22	0.05	1.03	31.0	8.0	4.0	0.0	-6.0
RPV Weld Material	-22	0.08	1.00	108.0	27.9	13.9	0.0	33.8
N12 Instrument Nozzle	-22	0.05	1.03	31.0	4.0	2.0	0.0	-14.0

Fluence Information: Location	Base Metal Thickness (inches) T	Base Metal Thickness (inches) 1/4T	Fluence at ID f_{surf} (n/cm^2)	Attenuation, 1/4t $e^{-0.24x}$	Fluence @ 1/4t (n/cm^2)	Fluence Factor, FF $f^{(0.28-0.10\log f)}$
Lower (Core Beltline) Shell	6.850	1.713	6.00E+17	0.663	3.98E+17	0.258
RPV Welding Material	6.850	1.713	6.00E+17	0.663	3.98E+17	0.258
N12 Instrument Nozzle	6.850	1.713	1.90E+17	0.663	1.26E+17	0.128

* Initial RT_{NDT} and Chemistry values are based on the bounding material specifications for STP Units 3 and 4.

Appendix A

STP Units 3 and 4 Reactor Vessel Material Surveillance Program

The STP Units 3 and 4 Reactor Vessel Material Surveillance Programs is described in Reference [6.12]. The program is in accordance with 10 CFR 50, Appendix H, Reactor Vessel Material Surveillance Program Requirements (Reference [6.11]). There are four surveillance capsules in each of the STP Units 3 and 4 RPVs. The first surveillance capsule is scheduled for withdrawal from each of the STP Units 3 and 4 reactor vessels after six (6) EFPY of operation. The surveillance capsules contain Charpy V-Notch (CVN) impact test specimens and uniaxial tensile test specimens fabricated in accordance with ASTM E185-82 (Reference [6.13]) using materials from the vessel materials within the core beltline region. When the test specimens are removed from the capsules, they will be tested according to ASTM E23-07a (Reference [6.14]). The methods and results of testing will be documented in test reports as required by 10 CFR 50, Appendices G and H (References [6.10] and [6.11]).