Serial Number 18-098
Docket Nos. 50-280/281

Attachment 4

<u>WCAP-18243-NP, REV. 0, SURRY UNITS 1 AND 2 HEATUP AND COOLDOWN LIMIT CURVES FOR NORMAL OPERATION</u>

<u>OCTOBER 2017</u>

VIRGINIA ELECTRIC AND POWER COMPANY
(DOMINION ENERGY VIRGINIA)
SURRY POWER STATION UNITS 1 AND 2

Westinghouse Non-Proprietary Class 3

WCAP-18243-NP
Revision 0

October 2017

Surry Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation

Westinghouse

WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-18243-NP
Revision 0

Surry Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation

Benjamin E. Mays*
Structural Design & Analysis III

Louis W. Turicik*
Aging Management & License Renewal

October 2017

Reviewers: D. Brett Lynch*
Anees Udyawar*
Alexandria M. Carolan*
Structural Design & Analysis III

Arzu Alpan*
Radiation Engineering & Analysis

Amy E. Freed*
Aging Management & License Renewal

Approved: Lynn A. Patterson*, Manager
Structural Design & Analysis III

Laurent P. Houssay*, Manager
Radiation Engineering & Analysis

*Electronically approved records are authenticated in the electronic document management system.

Westinghouse Electric Company LLC
1000 Westinghouse Dr.
Cranberry Township, PA 16066

RECORD OF REVISION

Revision 0: Original Issue

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

EXECUTIVE SUMMARY

This report provides the methodology and results of the generation of heatup and cooldown pressure-temperature (P-T) limit curves for normal operation of the Surry Units 1 and 2 reactor vessels. The heatup and cooldown P-T limit curves were generated using the limiting Adjusted Reference Temperature (ART) values for Surry Units 1 and 2. The limiting ART values which pertain to "axial flaw" materials were those of the Surry Unit 1 Lower Shell Longitudinal Weld L2 (Heat # 299L44, using Position 2.1) at both the 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations. The limiting ART values which pertain to "circumferential flaw" materials were those of the Surry Unit 1 Intermediate to Lower Shell Circumferential Weld (Heat # 72445, using Position 1.1 or Position 2.2) at the 1/4T location and the Surry Unit 2 Intermediate to Lower Shell Circumferential Weld (Heat # 0227, using Position 2.1) at the 3/4T location.

The P-T limit curves were generated for 68 effective full-power years (EFPY) using the K_{Ic} methodology detailed in the 1998 Edition through 2000 Addenda of the ASME Code, Section XI, Appendix G. The P-T limit curve generation methodology is consistent with the NRC-approved methodology documented in WCAP-14040-A, Revision 4. Heatup rates of 20, 40, and 60°F/hr, and cooldown rates of 0 (steady-state), 20, 40, 60, and 100°F/hr were used to generate the P-T limit curves, with the flange requirements and without margins for instrumentation errors. The Surry Units 1 and 2 Subsequent License Renewal (SLR) period of operation, also known as the Subsequent Period of Extended Operation (SPEO), corresponding to 80 years of operation is 68 EFPY. The SLR P-T limit curves can be found in Figures 6-1 and 6-2. As concluded in Section 7, the new 68 EFPY P-T limit curves are bounded by the current Surry Power Station P-T limit curves. Thus, continued use of the current Surry Power Station P-T limit curves is justified through 68 EFPY.

Appendix A contains the thermal stress intensity factors for the maximum heatup and cooldown rates at 68 EFPY based on the Section 6 P-T limit curves.

Appendix B contains a P-T limit evaluation of the reactor vessel inlet and outlet nozzles based on a 1/4T flaw postulated at the inside surface of the reactor vessel nozzle corner, where T is the thickness of the nozzle corner region. As discussed in Appendix B, the P-T limit curves generated based on the limiting cylindrical beltline materials bound the P-T limit curves for the reactor vessel inlet and outlet nozzles for Surry Units 1 and 2 at 68 EFPY.

Appendix C contains discussion of the other ferritic Reactor Coolant Pressure Boundary (RCPB) components relative to P-T limits. As discussed in Appendix C, all of the other ferritic RCPB components meet or are reconciled to the applicable requirements of Section III of the ASME Code.

Appendix D contains the determination of the Low Temperature Overpressure Protection (LTOP) system minimum enable temperature at 68 EFPY.

Appendix E contains an updated evaluation of weld Heat # 0227 initial material properties.

Appendix F contains a brief history of the Surry Units 1 and 2 P-T limit curves.

Appendix G contains an evaluation of the Surry Units 1 and 2 surveillance data credibility.

Appendix H contains a comparison of the "Axial Flaw" and "Circumferential Flaw" P-T limit curves.

Appendix I contains an evaluation of the Surry Units 1 and 2 Upper-Shelf Energy (USE) at 68 EFPY.

Appendix J contains a comparison of the material property input values used in this evaluation and those used in past evaluations as well as the Updated Final Safety Analysis Report (UFSAR).

1 INTRODUCTION

The purpose of this report is to present the calculations and the development of the Surry Units 1 and 2 heatup and cooldown P-T limit curves for 68 EFPY. This report documents the calculated Adjusted Reference Temperature (ART) values, the development of the P-T limit curves for normal operation, and comparison of these new P-T limit curves to the current P-T limit curves in the Surry Power Station Technical Specifications [Ref. 1]. The goal of this report is to demonstrate that the current P-T limit curves in the Surry Power Station Technical Specifications are bounding and remain valid through 80 years of operation. Note that the term "current" is utilized herein regarding P-T limit curves only in reference to the Surry Power Station Technical Specifications [Ref. 1] P-T limit curves.

Heatup and cooldown P-T limit curves are calculated using the adjusted RT_{NDT} (reference nil-ductility temperature) corresponding to the limiting beltline region material of the reactor vessel. The adjusted RT_{NDT} of the limiting material in the core region of the reactor vessel is determined by using the unirradiated reactor vessel material fracture toughness properties, estimating the radiation-induced ΔRT_{NDT}, and adding a margin. The unirradiated RT_{NDT} ($RT_{NDT(U)}$) is designated as the higher of either the drop weight nil-ductility transition temperature (NDTT) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (normal to the major working direction) minus 60°F. In instances where insufficient data is available to determine $RT_{NDT(U)}$ using ASME Code methods, alternate estimation methods such as Branch Technical Position (BTP) 5-3 are applied.

RT_{NDT} increases as the material is exposed to fast-neutron radiation. Therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the unirradiated RT_{NDT}. The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and nickel) present in reactor vessel steel. The U.S. Nuclear Regulatory Commission (NRC) has published a method for predicting radiation embrittlement in Regulatory Guide 1.99, Revision 2 [Ref. 2]. Regulatory Guide 1.99, Revision 2 is used for the calculation of ART values ($RT_{NDT(U)}$ + ΔRT_{NDT} + margins for uncertainties) at the 1/4T and 3/4T locations, where T is the thickness of the vessel at the beltline region measured from the clad/base metal interface. The calculated ART values for 68 EFPY are documented in Section 5 of this report. The fluence projections used in calculation of the ART values are provided in Section 2 of this report.

The heatup and cooldown P-T limit curves documented in this report were generated using the most limiting ART values and the NRC-approved methodology documented in WCAP-14040-A, Revision 4 [Ref. 3]. Specifically, the "Axial Flaw" and "Circumferential Flaw" methodologies of the 1998 Edition through 2000 Addenda of ASME Code, Section XI, Appendix G [Ref. 4] were used, which make use of the K_{Ic} methodology. The K_{Ic} curve is a lower bound static fracture toughness curve obtained from test data gathered from several different heats of pressure vessel steel. The limiting material is indexed to the K_{Ic} curve so that allowable stress intensity factors can be obtained for the material as a function of temperature. Allowable operating limits are then determined using the allowable stress intensity factors. The current P-T limit curves in the Surry Power Station Technical Specifications are based on the more conservative K_{Ir} fracture toughness curve. The methodology utilizing the K_{Ir} fracture toughness curve is equivalent to the K_{Ia} methodology, which is discussed further in this report.

The P-T limit curves presented herein were generated without instrumentation errors consistent with the Surry Power Station Technical Specification P-T limit curves. The reactor vessel flange requirements of

10 CFR 50, Appendix G [Ref. 5] have been incorporated in the P-T limit curves. The P-T limit curves generated in Section 6 were compared to the current Surry Units 1 and 2 P-T limit curves, contained in the Technical Specifications [Ref. 1], in Section 7 to determine if adequate margin exists to justify continued use of the Surry Units 1 and 2 current P-T limits through the Subsequent License Renewal (SLR) period of operation.

The P-T limit curves generated in Section 6 bound the P-T limit curves for the reactor vessel inlet and outlet nozzles generated in Appendix B for Surry Units 1 and 2 at 68 EFPY. Additionally, per Section 7, the current maximum allowable Low Temperature Overpressure Protection System (LTOPS) pressurizer Power Operated Relief Valve (PORV) setpoint Technical Specification value of ≤ 390.0 psig is bounding and will remain valid through the 80-year period of operation. Discussion of the other ferritic RCPB components relative to P-T limits is contained in Appendix C. Appendix D contains a calculation of the Low Temperature Overpressure Protection (LTOP) system enable temperature. Appendix E contains an evaluation of the initial material properties of weld Heat # 0227. Appendix F provides a summary of the Surry Units 1 and 2 P-T limit curves applicability. Appendix G provides a credibility evaluation of the Surry Units 1 and 2 surveillance data. Appendix H provides a comparison of the "Axial Flaw" and "Circumferential Flaw" P-T limit curves. Appendix I contains an evaluation of the Surry Units 1 and 2 Upper-Shelf Energy (USE) values at 68 EFPY. Appendix J contains a comparison of the material property input values used in this evaluation and those used in past evaluations as well as the Updated Final Safety Analysis Report (UFSAR).

2 CALCULATED NEUTRON FLUENCE

For the initial 60-year End of License Extension (EOLE) term, the Surry Units 1 and 2 fracture toughness properties provide adequate margins of safety against vessel failure. However, as the reactor operates, neutron irradiation (fluence) reduces material fracture toughness. Reactor Pressure Vessel (RPV) integrity is assured by demonstrating that RPV material fracture toughness will remain at levels that resist brittle fracture throughout the period of SLR operation. The first step in the analysis of vessel embrittlement is calculation of the neutron fluence that causes increased embrittlement.

Estimated RPV beltline and extended beltline fast neutron fluences (E > 1.0 MeV) at the end of 80 years of operation were calculated for Surry Units 1 and 2. The analyses methodologies used to calculate the Surry Units 1 and 2 RPV fluences satisfy the guidance set forth in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" [Ref. 6]. These methodologies have been approved by the U.S. NRC and are described in detail in Reference 3.

In accordance with Sections 3.1 and 4.2 of NUREG-2192 [Ref. 7], materials exceeding a fast neutron fluence (E > 1.0 MeV) of 1.0×10^{17} n/cm^2 at the end of the SLR period are evaluated for changes in fracture toughness. This guidance is consistent with Regulatory Issue Summary (RIS) 2014-11 [Ref. 8]. RPV materials that are not traditionally plant-limiting because of low levels of neutron radiation must now be evaluated to determine the accumulated fluence at SLR. Therefore, fast neutron fluence (E > 1.0 MeV) calculations were performed for the Surry Units 1 and 2 RPV circumferential welds (lower shell to lower vessel head, intermediate shell to lower shell, and nozzle shell to intermediate shell), inlet and outlet nozzle forging to vessel shell welds at the lowest extent, 1/4T flaw location in the inlet and outlet nozzle [Refs. 9 and 10], longitudinal welds (lower shell and intermediate shell), and plates (lower shell and intermediate shell), to determine if they will exceed a fast neutron fluence (E > 1.0 MeV) of 1.0×10^{17} n/cm^2 at SLR. The materials that exceed the 1.0×10^{17} n/cm^2 fast neutron fluence (E > 1.0 MeV) threshold, and were not evaluated in past analyses of record as part of the traditional beltline, are referred to as extended beltline materials in this report and are evaluated to determine the effect of neutron irradiation embrittlement during the SLR period.

In performing the fast neutron exposure evaluations for the Surry Units 1 and 2 reactor vessels, a series of fuel-cycle-specific forward transport calculations were carried out using the following two-dimensional/one-dimensional fluence rate synthesis technique:

$$\varphi(r,\theta,z) = \varphi(r,\theta) \times \frac{\varphi(r,z)}{\varphi(r)}$$

where $\varphi(r,\theta,z)$ is the synthesized 3D neutron fluence rate distribution, $\varphi(r,\theta)$ is the transport solution in r,θ geometry, $\varphi(r,z)$ is the two-dimensional solution for a cylindrical reactor model using the actual axial core power distribution, and $\varphi(r)$ is the one-dimensional solution for a cylindrical reactor model using the same source per unit height as that used in the r,θ two-dimensional calculation. This synthesis procedure was carried out for each operating cycle at Surry Units 1 and 2.

All of the transport calculations were carried out using the DORT discrete ordinates code [Ref. 11] with the BUGLE-96 cross-section library [Ref. 12]. The BUGLE-96 library provides a coupled 47-neutron-, 20-gamma-ray-group cross-section data set produced specifically for light water reactor applications. In these analyses, anisotropic scattering was treated with a P_5 Legendre expansion and the angular discretization was modeled with an S_{16} order of angular quadrature. Energy- and space-dependent core power distributions, as well as system operating temperatures, were treated on a fuel-cycle-specific basis.

The calculations for fuel Cycles 1 through 26 for Surry Unit 1 and fuel Cycles 1 through 25 for Surry Unit 2 determine the neutron exposure of the pressure vessel and surveillance capsules based on completed fuel cycles. For Surry Unit 1, projections for Cycle 27 and beyond were based on Cycle 26. For Surry Unit 2, projections for Cycle 26 and beyond were based on Cycle 25. Projected results (Cycle 27 and beyond for Surry Unit 1 and Cycle 26 and beyond for Surry Unit 2) will remain valid as long as future plant operation is consistent with these assumptions.

Table 2-1 gives the Surry Unit 1 calculated fast neutron fluences (E > 1.0 MeV) for all withdrawn surveillance capsules (Capsules T, W, V, and X). Table 2-2 gives the Surry Unit 2 calculated fast neutron fluences (E > 1.0 MeV) for all withdrawn surveillance capsules (Capsules X, W, V, S, W1, and Y). The EFPY and fast neutron fluences (E > 1.0 MeV) in Tables 2-1 and 2-2 were obtained from calculations performed to support the Measurement Uncertainty Recapture (MUR) power uprate. These fast neutron fluences (E > 1.0 MeV) were calculated using methodologies that follow the guidance of Regulatory Guide 1.190.

Table 2-1 Calculated Fast Neutron Fluence (E > 1.0 MeV) at the Surveillance Capsule Center for Surry Unit 1

Capsule ID	Azimuthal Location from Core Cardinal Axis (°)	Irradiation Cycle(s)	Cumulative Irradiation Time (EFPY)	Fast Neutron Fluence (E > 1.0 MeV) (n/cm^2)
T	15	1	1.1	2.71E+18
W	35	1-4	3.4	3.68E+18
V	15	1-8	8.0	1.80E+19
X	25	1-12	16.1	2.11E+19
	15	13-14		

Table 2-2 Calculated Fast Neutron Fluence (E > 1.0 MeV) at the Surveillance Capsule Center for Surry Unit 2

<table>
<tr><th>Capsule ID</th><th>Azimuthal Location from Core Cardinal Axis (°)</th><th>Irradiation Cycle(s)</th><th>Cumulative Irradiation Time (EFPY)</th><th>Fast Neutron Fluence (E > 1.0 MeV) (n/cm^2)</th></tr>
<tr><td>X</td><td>15</td><td>1</td><td>1.2</td><td>2.97E+18</td></tr>
<tr><td>W</td><td>25</td><td>1-4</td><td>3.8</td><td>6.36E+18</td></tr>
<tr><td>V</td><td>15</td><td>1-8</td><td>8.4</td><td>1.89E+19</td></tr>
<tr><td>S</td><td>45</td><td>1-13</td><td>15.0</td><td>1.07E+19</td></tr>
<tr><td>W1</td><td>15</td><td>11-14</td><td>5.3</td><td>7.80E+18</td></tr>
<tr><td rowspan="2">Y</td><td>25</td><td>1-12</td><td rowspan="2">20.3</td><td rowspan="2">2.72E+19</td></tr>
<tr><td>15</td><td>13-17</td></tr>
</table>

Selected results for the pressure vessel from the neutron transport analyses are provided in Tables 2-3 and 2-4 for Surry Units 1 and 2, respectively. Calculated fast neutron fluences (E > 1.0 MeV) for reactor vessel materials, on the pressure vessel clad/base metal interface, is provided for the nominal end of Cycle (EOC) 26 for Surry Unit 1 (32.5 EFPY) and nominal EOC 25 for Surry Unit 2 (31.3 EFPY). Surry Units 1 and 2 80-year plant life corresponds to 68 EFPY.

From Table 2-3 it is observed that one outlet nozzle and two inlet nozzles have fast neutron fluence (E > 1.0 MeV) greater than 1.0 x 10^{17} n/cm^2 at the nozzle forging to vessel shell weld and one inlet nozzle has fast neutron fluence (E > 1.0 MeV) greater than 1.0 x 10^{17} n/cm^2 at the 1/4T nozzle flaw location at 68 EFPY for Surry Unit 1. From Table 2-4, it is observed that one outlet nozzle and two inlet nozzles have fast neutron fluence (E > 1.0 MeV) greater than 1.0 x 10^{17} n/cm^2 at the nozzle forging to vessel shell weld and one outlet and one inlet nozzle have fast neutron fluence (E > 1.0 MeV) greater than 1.0 x 10^{17} n/cm^2 at the 1/4T nozzle flaw location at 68 EFPY for Surry Unit 2. Tables 2-3 and 2-4 indicate that the lower shell to lower vessel head circumferential weld will remain below 1.0 x 10^{17} n/cm^2 through SLR for both Surry Units 1 and 2.

Table 2-3 Surry Unit 1 – Maximum Fast Neutron Fluence (E > 1.0 MeV) Experienced by the Pressure Vessel Materials in the Beltline and Extended Beltline Regions

Material	Fast Neutron Fluence (n/cm^2)			
	32.5 EFPY	54 EFPY	68 EFPY	72 EFPY
1/4T Flaw in Outlet Nozzle				
Nozzle 1	1.53E+16	2.69E+16	3.45E+16	3.67E+16
Nozzle 2	1.08E+16	1.93E+16	2.49E+16	2.65E+16
Nozzle 3[(a)]	4.48E+16	7.59E+16	9.62E+16	1.02E+17
1/4T Flaw in Inlet Nozzle				
Nozzle 1[(b)]	5.80E+16	9.82E+16	1.24E+17	1.32E+17
Nozzle 2	1.40E+16	2.50E+16	3.22E+16	3.42E+16
Nozzle 3	1.98E+16	3.48E+16	4.46E+16	4.74E+16
Outlet Nozzle Forging to Vessel Shell Welds – Lowest Extent				
Nozzle 1	3.62E+16	6.35E+16	8.13E+16	8.63E+16
Nozzle 2	2.55E+16	4.55E+16	5.86E+16	6.23E+16
Nozzle 3[(c)]	1.06E+17	1.79E+17	2.27E+17	2.40E+17
Inlet Nozzle Forging to Vessel Shell Welds – Lowest Extent				
Nozzle 1[(d)]	1.42E+17	2.40E+17	3.04E+17	3.22E+17
Nozzle 2	3.43E+16	6.10E+16	7.84E+16	8.34E+16
Nozzle 3[(e)]	4.85E+16	8.51E+16	1.09E+17	1.16E+17
Nozzle Shell	3.64E+18	6.00E+18	7.54E+18	7.98E+18
Nozzle Shell to Intermediate Shell Circumferential Weld	3.64E+18	6.00E+18	7.54E+18	7.98E+18
Intermediate Shell				
Plate 1	3.17E+19	5.06E+19	6.29E+19	6.65E+19
Plate 2	3.17E+19	5.06E+19	6.29E+19	6.65E+19
Intermediate Shell Longitudinal Welds				
Weld 1	5.75E+18	9.85E+18	1.25E+19	1.33E+19
Weld 2	5.75E+18	9.85E+18	1.25E+19	1.33E+19
Intermediate Shell to Lower Shell Circumferential Weld	3.18E+19	5.08E+19	6.31E+19	6.67E+19
Lower Shell				
Plate 1	3.20E+19	5.11E+19	6.35E+19	6.70E+19
Plate 2	3.20E+19	5.11E+19	6.35E+19	6.70E+19
Lower Shell Longitudinal Welds				
Weld 1	5.80E+18	9.94E+18	1.26E+19	1.34E+19
Weld 2	5.80E+18	9.94E+18	1.26E+19	1.34E+19
Lower Shell to Lower Vessel Head Circumferential Weld	<1E+17	<1E+17	<1E+17	<1E+17

Notes:

(a) 1/4T flaw in Outlet Nozzle 3 is projected to reach 1.0 x 10^{17} n/cm^2 at approximately 70.7 EFPY.
(b) 1/4T flaw in Inlet Nozzle 1 is projected to reach 1.0 x 10^{17} n/cm^2 at approximately 55.0 EFPY.
(c) Outlet Nozzle 3 forging to vessel shell weld reached 1.0 x 10^{17} n/cm^2 at approximately 30.8 EFPY.
(d) Inlet Nozzle 1 forging to vessel shell weld reached 1.0 x 10^{17} n/cm^2 at approximately 23.2 EFPY.
(e) Inlet Nozzle 3 forging to vessel shell weld is projected to reach 1.0 x 10^{17} n/cm^2 at approximately 62.8 EFPY.

Table 2-4 Surry Unit 2 – Maximum Fast Neutron Fluence (E > 1.0 MeV) Experienced by the Pressure Vessel Materials in the Beltline and Extended Beltline Regions

Material	Neutron Fluence [n/cm²]			
	31.3 EFPY	54 EFPY	68 EFPY	72 EFPY
1/4T Flaw in Outlet Nozzle				
Nozzle 1	1.49E+16	2.66E+16	3.38E+16	3.58E+16
Nozzle 2	1.09E+16	1.95E+16	2.48E+16	2.63E+16
Nozzle 3[(a)]	4.29E+16	8.28E+16	1.07E+17	1.15E+17
1/4T Flaw in Inlet Nozzle				
Nozzle 1[(b)]	5.55E+16	1.07E+17	1.39E+17	1.48E+17
Nozzle 2	1.41E+16	2.52E+16	3.21E+16	3.40E+16
Nozzle 3	1.93E+16	3.44E+16	4.37E+16	4.63E+16
Outlet Nozzle Forging to Vessel Shell Welds – Lowest Extent				
Nozzle 1	3.52E+16	6.27E+16	7.96E+16	8.45E+16
Nozzle 2	2.57E+16	4.60E+16	5.85E+16	6.20E+16
Nozzle 3[(c)]	1.01E+17	1.95E+17	2.53E+17	2.70E+17
Inlet Nozzle Forging to Vessel Shell Welds – Lowest Extent				
Nozzle 1[(d)]	1.36E+17	2.62E+17	3.40E+17	3.62E+17
Nozzle 2	3.45E+16	6.17E+16	7.84E+16	8.32E+16
Nozzle 3[(e)]	4.73E+16	8.41E+16	1.07E+17	1.13E+17
Nozzle Shell	3.52E+18	6.70E+18	8.65E+18	9.21E+18
Nozzle Shell to Intermediate Shell Circumferential Weld	3.52E+18	6.70E+18	8.65E+18	9.21E+18
Intermediate Shell				
Plate 1	3.10E+19	5.64E+19	7.20E+19	7.65E+19
Plate 2	3.10E+19	5.64E+19	7.20E+19	7.65E+19
Intermediate Shell Longitudinal Welds				
Weld 1	5.98E+18	1.03E+19	1.29E+19	1.36E+19
Weld 2	5.98E+18	1.03E+19	1.29E+19	1.36E+19
Intermediate Shell to Lower Shell Circumferential Weld	3.11E+19	5.66E+19	7.22E+19	7.67E+19
Lower Shell				
Plate 1	3.12E+19	5.68E+19	7.26E+19	7.71E+19
Plate 2	3.12E+19	5.68E+19	7.26E+19	7.71E+19
Lower Shell Longitudinal Welds				
Weld 1	6.03E+18	1.03E+19	1.30E+19	1.37E+19
Weld 2	6.03E+18	1.03E+19	1.30E+19	1.37E+19
Lower Shell to Lower Vessel Head Circumferential Weld	<1E+17	<1E+17	<1E+17	<1E+17

Notes:

(a) 1/4T flaw in Outlet Nozzle 3 is projected to reach 1.0 x 10^{17} n/cm² at approximately 63.8 EFPY.

(b) 1/4T flaw in Inlet Nozzle 1 is projected to reach 1.0 x 10^{17} n/cm² at approximately 50.9 EFPY.

(c) Outlet Nozzle 3 forging to vessel shell weld reached 1.0 x 10^{17} n/cm² at approximately 31.0 EFPY.

(d) Inlet Nozzle 1 forging to vessel shell weld reached 1.0 x 10^{17} n/cm² at approximately 23.5 EFPY.

(e) Inlet Nozzle 3 forging to vessel shell weld is projected to reach 1.0 x 10^{17} n/cm² at approximately 63.9 EFPY.

3 MATERIAL PROPERTY INPUT

The requirements for P-T limit curve development are specified in 10 CFR 50, Appendix G [Ref. 5]. The beltline region of the reactor vessel is defined as the following in 10 CFR 50, Appendix G:

> *"the region of the reactor vessel (shell material including welds, heat affected zones and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage."*

Per RIS 2014-11 [Ref. 8] materials which are predicted to experience neutron fluence greater than 1 x 10^{17} n/cm^2 (E > 1 MeV) at the end of the licensed operating period must also be evaluated for neutron embrittlement effects. Materials which have not previously been considered in the beltline region, but are predicted to experience neutron fluence greater than 1 x 10^{17} n/cm^2 are termed "extended beltline" materials.

The Surry Unit 1 beltline materials consist of two (2) Intermediate Shell (IS) Plates, two (2) Lower Shell (LS) Plates, one (1) Upper Shell (US) Forging (also termed nozzle shell forging), two (2) IS Longitudinal Welds, two (2) LS Longitudinal Welds, and two (2) circumferential welds: the IS to LS Circumferential Weld and the US to IS Circumferential Weld. The Surry Unit 1 surveillance plate material was made from reactor vessel Lower Shell Plate C4415-1. Since Lower Shell Plate C4415-1 shares a heat number with Lower Shell Plate C4415-2, the surveillance plate results also apply to Lower Shell Plate C4415-2. The Surry Unit 1 reactor vessel beltline LS Longitudinal weld (L2) was fabricated using weld wire Heat # 299L44, Linde 80 Flux Type, Lot Number 8596. The weld material in the Surry Unit 1 surveillance program was fabricated with the same material heat, flux type, and lot number as reactor vessel beltline Longitudinal Weld L2. Weld material Heat # 299L44 was included in the surveillance programs of other plants, as summarized in Table 3-7. The US to IS Circumferential Weld (W06) was fabricated with weld wire Heat # 25017, SAF 89 Flux Type, Flux Lot Number 1197. The IS to LS Circumferential Weld (W05) was fabricated with weld wire Heat # 72445, Linde 80 Flux Type, Flux Lot Number 8597 (40%) and Flux Lot Number 8623 (60%). Surveillance data does not exist for Heat # 25017 or Heat # 72445 in the Surry Unit 1 reactor vessel surveillance program; however weld wire Heat # 72445 was included in the surveillance programs of other plants, as summarized in Table 3-8. The LS Longitudinal Weld (L1) and both IS Longitudinal Welds (L3 and L4) were fabricated using weld wire Heat # 8T1554, Linde 80 Flux Type, Flux Lot Number 8579. Surveillance data does not exist for Heat # 8T1554.

The Surry Unit 2 beltline materials consist of two (2) Intermediate Shell (IS) Plates, two (2) Lower Shell (LS) Plates, one (1) Upper Shell (US) Forging, two (2) IS Longitudinal Welds, two (2) LS Longitudinal Welds, and two (2) circumferential welds: the IS to LS Circumferential Weld and US to IS Circumferential Weld. The Surry Unit 2 surveillance plate material was made from reactor vessel Lower Shell Plate C4339-1. Since Lower Shell Plate C4339-1 shares a heat number with Intermediate Shell Plate C4339-2, the surveillance plate results also apply to Intermediate Shell Plate C4339-2. The Surry Unit 2 reactor vessel beltline IS to LS Circumferential Weld (W05) was fabricated using weld wire Heat # 0227, Grau Lo Flux Type, Flux Lot Number LW320. The weld material in the Surry Unit 2 surveillance program was fabricated with the same material heat, flux type, and lot number as the IS to LS Circumferential Weld. The US to IS circumferential weld (W06) was fabricated with weld wire Heat #

4275, SAF 89 Flux Type, Flux Lot Number 02275. Weld material Heat # 0227 and Heat # 4275 are not included in the surveillance programs of other plants. The IS Longitudinal Weld L3 and 50% of IS Longitudinal Weld L4 were fabricated with weld wire Heat # 72445, Linde 80 Flux Type, Flux Lot Number 8597. Data does not exist for Heat # 72445 in the Surry Unit 2 reactor vessel surveillance program; however, weld wire Heat # 72445 was included in the surveillance programs of other plants, as summarized in Table 3-8. The remaining 50% of IS Longitudinal Weld L4, LS Longitudinal Weld L1, and 63% of LS Longitudinal Weld L2 were fabricated from weld wire Heat # 8T1762, Linde 80 Flux Type, Flux Lot Number 8597. The remaining 37% of LS Longitudinal Weld L2 was fabricated from weld wire Heat # 8T1762, Linde 80 Flux Type, Flux Lot Number 8632. Surveillance data does not exist for Heat # 8T1762.

Based on the results of Section 2 of this report, the materials that exceeded the 1 x 10^{17} n/cm^2 (E > 1.0 MeV) threshold at 68 EFPY are considered to be the Surry Units 1 and 2 extended beltline materials and are evaluated to determine their impact on the SLR period of operation. The forgings and welds corresponding to the Surry Units 1 and 2 Inlet Nozzles 1, Inlet Nozzles 3, and Outlet Nozzles 3 are predicted to experience neutron fluence greater than 1.0 x 10^{17} n/cm^2 at SLR. However, for conservatism all of the Surry Units 1 and 2 inlet and outlet nozzle materials are considered part of the extended beltline. Thus, the Surry Units 1 and 2 extended beltline materials consist of three (3) Inlet Nozzles, three (3) Outlet Nozzles, three (3) Inlet Nozzle to US Welds, and three (3) Outlet Nozzle to US Welds per Unit. The Surry Unit 1 Inlet Nozzle to Upper Shell Welds were fabricated using Heat #s 299L44 and 8T1762, Linde 80 Flux Type, Lot Number 8596. The Surry Unit 1 Outlet Nozzle to Upper Shell Welds were fabricated using Heat # 8T1762, Linde 80 Flux Type, Lot Number 8578 and Heat # 8T1554B, Linde 80 Flux Type, Flux Lot Number 8579. The Surry Unit 2 Inlet Nozzle to Upper Shell Welds were fabricated using Heat # 8T1762, Linde 80 Flux Type, and Lot Numbers 8597 and 8632. The materials constituting the Surry Unit 2 Outlet Nozzle to Upper Shell Welds could not be determined; however, these welds were completed at Rotterdam per BAW-2313, Revision 7, Supplement 1, Revision 1 [Ref. 13]. Surveillance data from Surry Unit 1 and additional plant surveillance programs exists, as previously described, for Heat # 299L44. No additional surveillance data exists for any of the materials in the Surry Units 1 and 2 extended beltline. The data supporting this materials summary was gathered primarily from PWROG-16045-NP, Revision 0 [Ref. 14].

The identification of the RPV beltline and extended beltline plate and weld materials are included in Figures 3-1 and 3-2 for Surry Unit 1 and Figures 3-3 and 3-4 Surry Unit 2. The material property inputs used for the subsequent P-T limits evaluations contained in this report are described in this section. Note that some of the beltline material initial properties were updated from previous RV integrity evaluations per PWROG-16045-NP, Revision 0 and Appendix E herein, and the fluence values were updated per WCAP-18028-NP, Revision 0 [Ref. 15] and Section 2 herein. Additionally, initial USE values are supplied in Table 3-1 and Table 3-3 for certain welds, which had an initial USE value designated as "EMA" in PWROG-16045-NP, Revision 0. The sources and methods used in the determination of the chemistry factors and the fracture toughness properties are summarized below.

Chemical Compositions

The best-estimate copper (Cu) and nickel (Ni) chemical compositions for the Surry Units 1 and 2 beltline and extended beltline materials are presented in Tables 3-1 through 3-4. The best-estimate weight percent copper and nickel values for the beltline and extended beltline materials were previously reported in PWROG-16045-NP, Revision 0.

Fracture Toughness Properties

The fracture toughness properties (initial RT_{NDT} and initial Upper-Shelf Energy [USE]) of most of beltline plate materials were originally determined using NUREG-0800, BTP 5-3 Position 1.1 [Ref. 16] methodology, with three exceptions. Surry Unit 1 IS Plate C4326-1, Surry Unit 1 LS Plate C4415-1, and Surry Unit 2 LS Plate C4339-1 were determined using the ASME Code, Section III [Ref. 17]. Many of the beltline and extended beltline fracture toughness properties were updated per ASME Section III, the General Electric (GE) Method [Ref. 18], and NUREG-0800, BTP 5-3 Position 1.1 methodologies, as described in PWROG-16045-NP, Revision 0 [Ref. 14]. The initial RT_{NDT} values for Surry Unit 1 Longitudinal Welds L1, L2, L3, and L4 and Intermediate to Lower Shell Circumferential Weld Heat # 72445 were determined using the "Master Curve" method ($RT_{NDT} = T_0 + 35°F$). The initial RT_{NDT} values for Surry Unit 2 Longitudinal Welds L1, L2, L3, and L4 were also determined using this method. Chemistry factor (CF) values and margin terms require evaluation when using "Master Curve"-generated initial RT_{NDT} values to calculate adjusted reference temperature (ART) values. When using these "Master Curve"-generated initial RT_{NDT} values, the CF and margin terms will be adjusted to a minimum of 167°F and 28°F, respectively. However, if the material-specific CF value or margin term is greater than 167°F or 28°F, respectively, the material-specific value(s) will be used. The most up-to-date initial RT_{NDT} and initial USE values are documented in PWROG-16045-NP, Revision 0 for Surry Units 1 and 2 with the exception of the Surry Unit 2 Intermediate to Lower Shell Circumferential Weld (Heat # 0227), which was updated in Appendix E herein. Table 8 of PWROG-16045-NP contains the Surry Unit 1 initial properties, and Table 9 of PWROG-16045-NP contains the Surry Unit 2 initial properties. The Surry Unit 2 IS to LS Circumferential Weld initial material properties are updated in Appendix E herein. The beltline and extended beltline material properties of the Surry Units 1 and 2 reactor vessels are presented in Tables 3-1 through 3-4 herein. A comparison of the material property input values utilized herein and those utilized previously is documented in Appendix J.

The initial RT_{NDT} values of the reactor vessel flange and closure head serve as input to the P-T limit curves "flange-notch" per 10 CFR 50, Appendix G [Ref. 5] and were confirmed to be acceptable. Since Surry Units 1 and 2 share P-T Limit curves for operation, materials for both plants must be considered. The closure heads at both Surry Units 1 and 2 have been replaced, and the initial RT_{NDT} values of the Surry Units 1 and 2 flange materials were updated in PWROG-16045-NP, Revision 0 [Ref. 14]. The Surry Unit 1 replacement closure head has an initial RT_{NDT} value of -67°F, determined per ASME Code Section III, NB-2300. The Surry Unit 1 reactor vessel flange has an initial RT_{NDT} of -114.6°F, calculated using the GE methodology. The Surry Unit 2 replacement head has an initial RT_{NDT} value of -60°F, determined per ASME Code Section III, NB-2300. The Surry Unit 2 reactor vessel flange has an initial RT_{NDT} of -156.3°F, calculated using the GE methodology. See Tables 3-5 and 3-6 for a summary of the initial RT_{NDT} values for these two components at each plant.

Chemistry Factor Values

The chemistry factor (CF) values were calculated using Positions 1.1 and 2.1 of Regulatory Guide 1.99, Revision 2 [Ref. 2]. Position 1.1 uses Tables 1 and 2 from the Regulatory Guide along with the best-estimate copper and nickel weight percent values (contained in Tables 3-1 through 3-4, and Tables 3-7 and 3-8). Position 2.1 uses the surveillance capsule data from all capsules tested to date and surveillance data from other plants, as applicable. A credibility evaluation of the surveillance data is provided in Appendix G. The calculated capsule fluence values are provided in Tables 2-1 and 2-2 and are used to determine the Position 2.1 CFs as shown in Tables 3-9 and 3-11 for Surry Units 1 and 2, respectively. Tables 3-10 and 3-12 summarize the Positions 1.1 and 2.1 CF values determined for the Surry Units 1 and 2 RPV beltline and extended beltline materials, respectively.

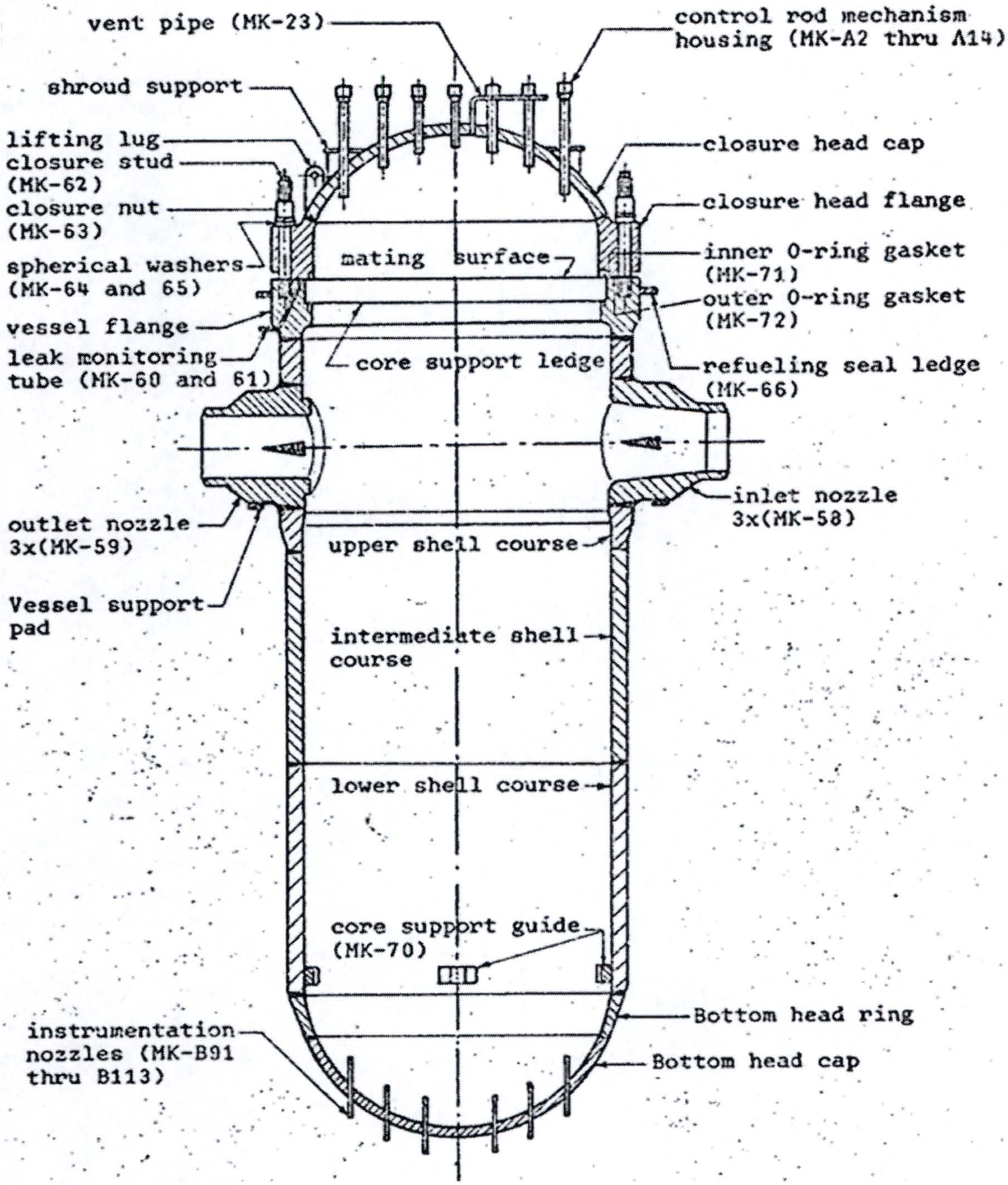


Figure 3-1 RPV Base Metal Material Identifications for Surry Unit 1

*Note: Figure may not be representative of the replacement RPV closure head at Surry Unit 1.

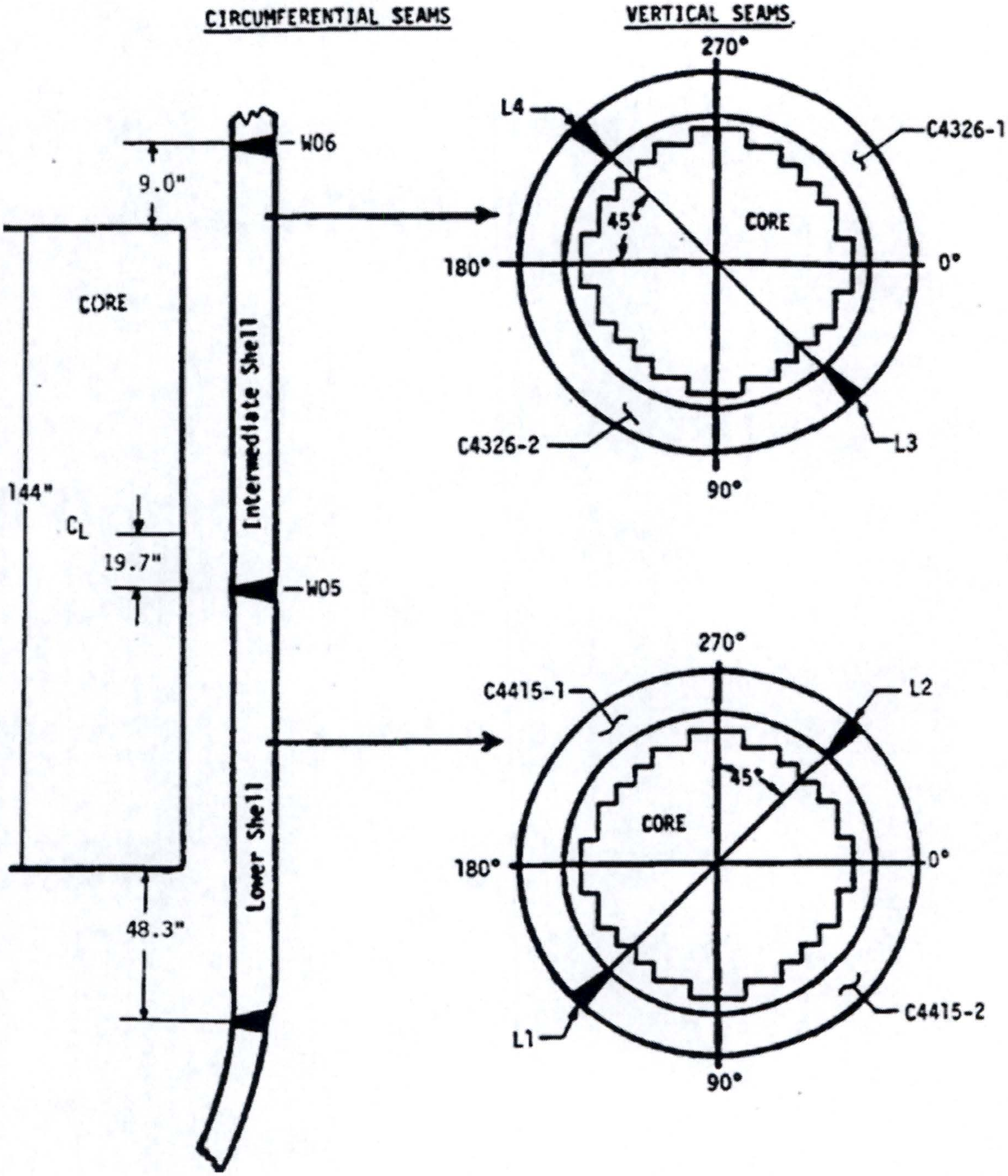


Figure 3-2 RPV Weld Identifications for Surry Unit 1

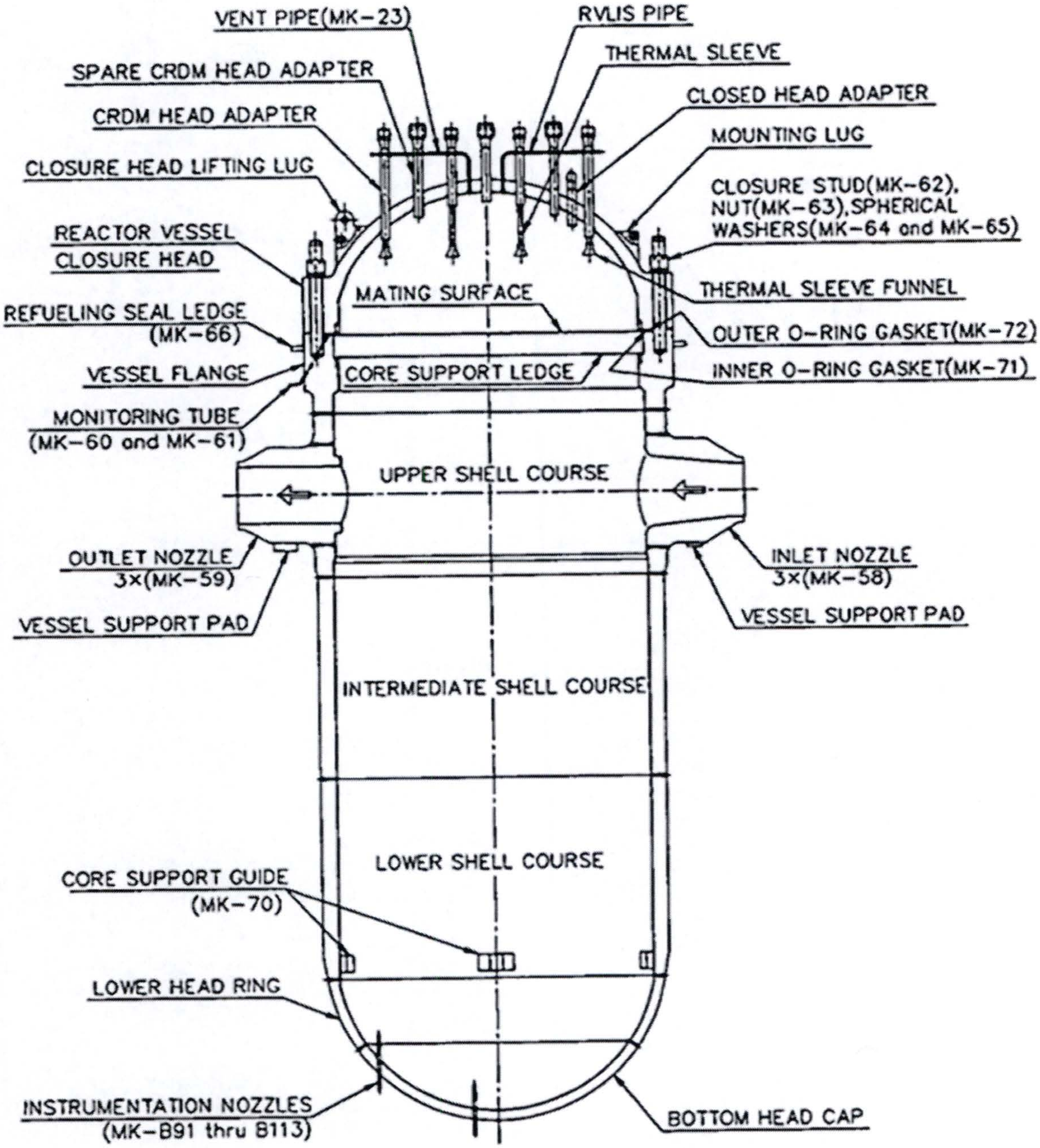


Figure 3-3 RPV Base Metal Material Identifications for Surry Unit 2

*Note: Figure may not be representative of the replacement RPV closure head at Surry Unit 2.

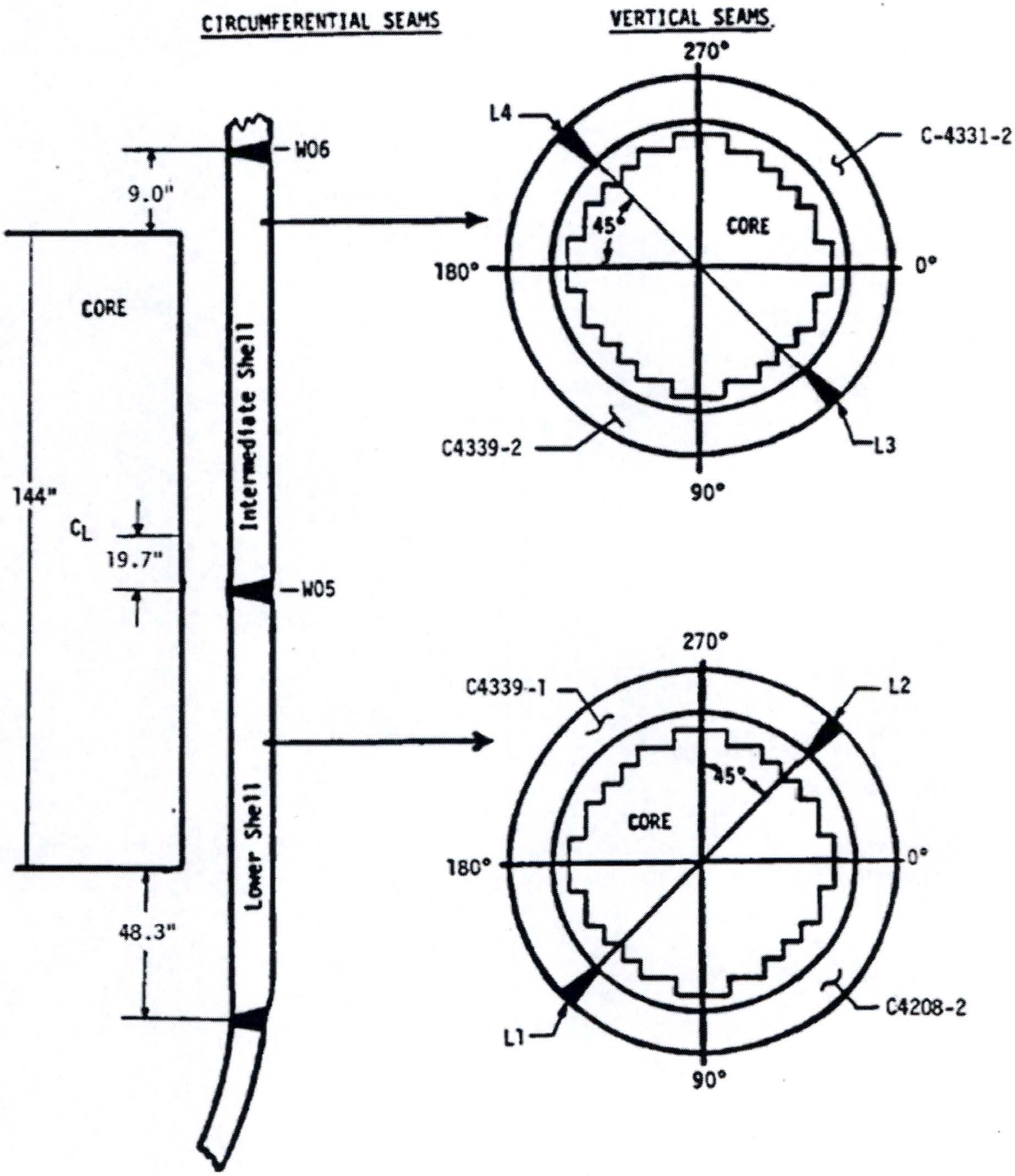


Figure 3-4 RPV Weld Identifications for Surry Unit 2

Table 3-1 Best-Estimate Cu and Ni Weight Percent Values, Initial RT_{NDT} Values, and Initial USE Values for the Surry Unit 1 RPV Beltline and Surveillance Materials

RPV Material	Wt. % Cu	Wt. % Ni	$RT_{NDT(U)}$ (°F)	σ_I (°F)	Initial USE (ft-lb)
Reactor Vessel Beltline Materials[(a)]					
Upper Shell Forging 122V109VA1	0.11	0.74	40	0	114
Intermediate Shell Plate C4326-1	0.11	0.55	10	0	115
Intermediate Shell Plate C4326-2	0.11	0.55	11.4	0	94
Lower Shell Plate C4415-1	0.102	0.493	20	0	103
Lower Shell Plate C4415-2	0.11	0.50	4.6	0	82
Upper to Intermediate Shell Circumferential Weld (Heat # 25017)	0.33	0.10	0	20.0	≥64[(b)]
Intermediate Shell Longitudinal Welds L3 and L4 (Heat # 8T1554)	0.16	0.57	-48.6	18.0	64[(b)]
Intermediate to Lower Shell Circumferential Weld (Heat # 72445)	0.22	0.54	-72.5	12.0	64[(b)]
Lower Shell Longitudinal Weld L1 (Heat # 8T1554)	0.16	0.57	-48.6	18.0	64[(b)]
Lower Shell Longitudinal Weld L2 (Heat # 299L44)	0.34	0.68	-74.3	12.8	64[(b)]
Reactor Vessel Surveillance Materials[(c)]					
Lower Shell Plate C4415-1	0.102	0.493	20	0	103
Surveillance Weld (Heat # 299L44)	0.23	0.64	---	---	70

Notes:

(a) All values were taken from Table 8 of PWROG-16045-NP, Revision 0 [Ref. 14], unless otherwise noted.

(b) Per Surry Power Station UFSAR [Ref. 19], reactor vessel Equivalent Margins Analysis (EMA) report BAW-2494, Revision 1 [Ref. 20] has been approved for these welds. The EMA is updated for SLR under Pressurized Water Reactor Owners Group (PWROG) PA-MSC-1481. Linde 80 initial USE values are set to the generic value of 64 ft-lbs per BAW-2313, Revision 7, Supplement 1, Revision 1 [Ref. 13]. Only limited Charpy test information is available for Heat # 25017. Based on the average Charpy energy value of the weld qualification tests completed at 10°F, the USE for Heat # 25017 is at least 64 ft-lbs.

(c) The surveillance plate data was taken to be the same as the vessel plate data. The surveillance weld data was obtained from BAW-2324, Revision 0 [Ref. 21].

Table 3-2 Best-Estimate Cu and Ni Weight Percent Values, Initial RT_{NDT} Values, and Initial USE Values for the Surry Unit 1 RPV Extended Beltline Materials

RPV Material		Wt. % Cu	Wt. % Ni	$RT_{NDT(U)}$ (°F)	σ_I (°F)	Initial USE (ft-lb)
Reactor Vessel Extended Beltline Materials[(a)]						
Inlet Nozzle 1 (Heat # 9-4787)		0.159	0.85	10.3	0	63
Inlet Nozzle 2 (Heat # 9-5078)		0.159	0.87	11.6	0	64
Inlet Nozzle 3 (Heat # 9-4819)		0.159	0.84	-47.2	0	68
Outlet Nozzle 1 (Heat # 9-4825-1)		0.159	0.85	-44.9	0	68
Outlet Nozzle 2 (Heat # 9-4762)		0.159	0.83	-87.5	0	82
Outlet Nozzle 3 (Heat # 9-4788)		0.159	0.84	-50.2	0	71
Inlet Nozzle to Upper Shell Welds	Heat # 299L44	0.34	0.68	-7.0	20.6	64
	Heat # 8T1762	0.19	0.57	-4.9	19.7	64
Outlet Nozzle to Upper Shell Welds	Heat # 8T1762	0.19	0.57	-4.9	19.7	64
	Heat # 8T1554B	0.16	0.57	-4.9	19.7	64

Note:

(a) All values were taken from Table 8 of PWROG-16045-NP, Revision 0 [Ref. 14].

Table 3-3 Best-Estimate Cu and Ni Weight Percent Values, Initial RT_{NDT} Values, and Initial USE Values for the Surry Unit 2 RPV Beltline and Surveillance Materials

RPV Material	Wt. % Cu	Wt. % Ni	$RT_{NDT(U)}$ (°F)	σ_I (°F)	Initial USE (ft-lb)
Reactor Vessel Beltline Materials[(a)]					
Upper Shell Forging 123V303VA1	0.11	0.72	30	0	104
Intermediate Shell Plate C4331-2	0.12	0.60	15.0	0	84
Intermediate Shell Plate C4339-2	0.11	0.54	7.8	0	83
Lower Shell Plate C4208-2	0.15	0.55	-30	0	94
Lower Shell Plate C4339-1	0.107	0.53	-4.4	0	101
Upper to Intermediate Shell Circumferential Weld (Heat # 4275)	0.35	0.10	0	20.0	≥68[(b)]
Intermediate Shell Longitudinal Welds L3 and L4 (OD 50%) (Heat # 72445)	0.22	0.54	-72.5	12.0	64[(b)]
Intermediate Shell Longitudinal Weld L4 (ID 50%) (Heat # 8T1762)	0.19	0.57	-48.6	18.0	64[(b)]
Intermediate to Lower Shell Circumferential Weld (Heat # 0227)	0.187	0.545	0[(c)]	0[(c)]	82[(c)]
Lower Shell Longitudinal Welds L1 and L2 (Heat # 8T1762)	0.19	0.57	-48.6	18.0	64[(b)]
Reactor Vessel Surveillance Materials[(d)]					
Lower Shell Plate C4339-1	0.107	0.53	-4.4	0	101
Surveillance Weld (Heat # 0227)	0.19	0.56	---	---	91

Notes:

(a) All values were taken from Table 9 of PWROG-16045-NP, Revision 0 [Ref. 14], unless otherwise noted.

(b) Per Surry Power Station UFSAR [Ref. 19], reactor vessel EMA report BAW-2494, Revision 1 [Ref. 20] has been approved for these welds. The EMA is updated for SLR under PWROG PA-MSC-1481. Linde 80 initial USE values are set to the generic value of 64 ft-lbs per BAW-2313, Revision 7, Supplement 1, Revision 1 [Ref. 13]. Only limited Charpy test information is available for Heat # 4275. Based on the average Charpy energy value of the weld qualification tests completed at 10°F, the USE for Heat # 4275 is at least 68 ft-lbs.

(c) Initial properties are established in Appendix E. Since the initial RT_{NDT} is based on measured data, σ_I is equal to 0°F. Per Surry Power Station UFSAR [Ref. 19], reactor vessel EMA report BAW-2494, Revision 1 [Ref. 20] has been approved for this weld. The EMA is updated for SLR under PWROG PA-MSC-1481.

(d) The surveillance plate data was taken to be the same as the vessel plate data. The surveillance weld data was obtained from WCAP-16001, Revision 0 [Ref. 22].

Table 3-4 Best-Estimate Cu and Ni Weight Percent Values, Initial RT_{NDT} Values, and Initial USE Values for the Surry Unit 2 RPV Extended Beltline Materials

RPV Material		Wt. % Cu	Wt. % Ni	$RT_{NDT(U)}$ (°F)	σ_I (°F)	Initial USE (ft-lb)
Reactor Vessel Extended Beltline Materials[(a)]						
Inlet Nozzle 1 (Heat # 9-5104)		0.159	0.84	-29.7	0	73
Inlet Nozzle 2 (Heat # 9-4815)		0.159	0.87	4.5	0	66
Inlet Nozzle 3 (Heat # 9-5205)		0.159	0.86	6.5	0	67
Outlet Nozzle 1 (Heat # 9-4825-2)		0.159	0.85	-58.1	0	73
Outlet Nozzle 2 (Heat # 9-5086-1)		0.159	0.86	-26.6	0	77
Outlet Nozzle 3 (Heat # 9-5086-2)		0.159	0.87	-33.8	0	71
Inlet Nozzle to Upper Shell Welds	Heat # 8T1762	0.19	0.57	-4.9	19.7	64
Outlet Nozzle to Upper Shell Welds	Rotterdam	0.35	1.0	30	0	71[(b)]

Notes:

(a) All values were taken from Table 9 of PWROG-16045-NP, Revision 0 [Ref. 14].

(b) Per PWROG-16045-NP, Revision 0 [Ref. 14], this initial USE value is set equal to the USE value of the first tested capsule from WCAP-16001 [Ref. 22]. This methodology utilizes BTP 5-3 [Ref. 16], Position 1.2 guidance, as no USE data is available from the supplier.

Table 3-5 Initial RT_{NDT} Values for the Surry Unit 1 Replacement Reactor Vessel Closure Head and Vessel Flange Materials

RPV Material	Initial RT_{NDT} (°F)
Replacement Closure Head E4381/E4382	-67[(a)]
Vessel Flange FV-1870	-144.6[(b)]

Notes:

(a) Value taken from Table 8 of PWROG-16045-NP, Revision 0 [Ref. 14]. This value is based on ASME Code Section III, NB-2300 criteria. Note that the original Closure Head Flange initial RT_{NDT} was 10°F per WCAP-14177 [Ref. 23].

(b) Value taken from Table 8 of PWROG-16045-NP, Revision 0 [Ref. 14]. This value is based on the GE Methodology. Note that the Vessel Flange Initial RT_{NDT} used in previous reactor vessel integrity calculations was 10°F as documented in WCAP-14177 [Ref. 23].

Table 3-6 Initial RT_{NDT} Values for the Surry Unit 2 Replacement Reactor Vessel Closure Head and Vessel Flange Materials

RPV Material	Initial RT_{NDT} (°F)
Replacement Closure Head 02W1-1-1-1	-60[(a)]
Vessel Flange FV-2542	-156.3[(b)]

Notes:

(a) Value taken from Table 9 of PWROG-16045-NP, Revision 0 [Ref. 14]. This value is based on ASME Code Section III, NB-2300 criteria. Note that the original Closure Head Flange initial RT_{NDT} was 10°F per WCAP-14177 [Ref. 23].

(b) Value taken from Table 9 of PWROG-16045-NP, Revision 0 [Ref. 14]. This value is based on the GE Methodology. Note that the Vessel Flange Initial RT_{NDT} used in previous reactor vessel integrity calculations was -65°F as documented in WCAP-14177 [Ref. 23].

Table 3-7 Surveillance Data for Weld Wire Heat # 299L44

Capsule Designation[a]	Cu wt. %	Ni wt. %	CF (°F)	Capsule Fluence ($x\ 10^{19}\ n/cm^2$, E > 1.0 MeV)	ΔRT_{NDT} (°F)	Irradiation Temperature (°F)
TMI2-LG1 (CR-3)[b]	0.37	0.70	234.0	0.830	216	556
W1(CR-3)[c]	0.37	0.70	234.0	0.780	262	545
TMI1-E	0.33	0.67	215.2	0.107	74	556
TMI1-C	0.33	0.67	215.2	0.882	166	556
TMI2-LG1(TMI-2)[b]	0.33	0.67	215.2	0.968	226	556
CR3-LG1(ONS-3)	0.36	0.70	230.5	0.779	202	556
A5[d]	0.23	0.64	175.8	2.75	246.6	556
Surry Unit 1: Capsule T	0.23	0.64	175.8	0.271	171	537
Surry Unit 1: Capsule V	0.23	0.64	175.8	1.80	250	539
Surry Unit 1: Capsule X	0.23	0.64	175.8	2.11	234	542

Notes:

(a) Data was obtained from ANP-2650 [Ref. 24], unless otherwise noted. Material source is indicated in parentheses. CR-3 = Crystal River Unit 3, TMI1 = Three Mile Island Unit 1, ONS = Oconee Nuclear Station Unit 3.

(b) Material is from different sources, irradiated in the same capsule.

(c) Capsule W1 was irradiated in Surry Unit 2. The fluence value is updated from ANP-2650 [Ref. 24] per Section 2. The irradiation temperature value is the time-weighted average T_{cold} considering the cycles that W1 was inside the Surry Unit 2 reactor vessel.

(d) Data taken from AREVA-17-01417 [Ref. 25].

Table 3-8 Surveillance Data for Weld Wire Heat # 72445

Capsule Designation[a]	Cu wt. %	Ni wt. %	CF (°F)	Capsule Fluence (x 10^{19} n/cm^2, E > 1.0 MeV)	ΔRT_{NDT} (°F)	Irradiation Temperature (°F)
CR3-LG1 (ANO-1)	0.22	0.59	165.5	0.510	139	556
CR3-LG2 (ANO-1)	0.22	0.59	165.5	1.670	164	556
W1 (ANO-1)[b]	0.22	0.59	165.5	0.780	138	545
Point Beach Unit 1: Capsule V	0.23	0.62	172.4	0.634	107	542
Point Beach Unit 1: Capsule S	0.23	0.62	172.4	0.829	165	542
Point Beach Unit 1: Capsule R	0.23	0.62	172.4	2.190	155	541.6
Point Beach Unit 1: Capsule T	0.23	0.62	172.4	2.230	181	533.4

Notes:

(a) Data was obtained from ANP-2650 [Ref. 24], unless otherwise noted. Material source is indicated in parentheses. ANO-1 = Arkansas Nuclear One Unit 1

(b) Capsule W1 was irradiated in Surry Unit 2. The fluence value is updated from ANP-2650 [Ref. 24] per Section 2. The irradiation temperature value is the time-weighted average T_{cold} considering the cycles that W1 was inside the Surry Unit 2 reactor vessel.

Table 3-9 Calculation of Position 2.1 CF Values for Surry Unit 1[(a)]

RPV Material	Capsule	Capsule Fluence (x 10^{19} n/cm^2, E > 1.0 MeV)	FF[(c)]	ΔRT_{NDT} (°F)	Adjusted ΔRT_{NDT} [(d)] (°F)	FF*Adjusted ΔRT_{NDT} (°F)	FF^2
Lower Shell Plate C4415-1[(b)] (Longitudinal)	T	0.271	0.644	50	50	32.21	0.415
	V	1.80	1.161	113	113	131.23	1.349
	X	2.11	1.203	86	86	103.46	1.447
					SUM:	266.91	3.211
	$CF_{C4415-1} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (266.91) \div (3.211) =$ **83.1°F**						
Surveillance Weld Material (Heat # 299L44)	T	0.271	0.644	171	208	133.69	0.415
	V	1.80	1.161	250	309	358.56	1.349
	X	2.11	1.203	234	293	351.89	1.447
	TMI2-LG1	0.830	0.948	216	230	217.98	0.898
	W1	0.780	0.930	262	265	246.53	0.865
	TMI1-E	0.107	0.431	74	91	39.02	0.185
	TMI1-C	0.882	0.965	166	185	178.87	0.931
	TMI2-LG1	0.968	0.991	226	247	244.95	0.982
	CR3-LG1	0.779	0.930	202	216	200.87	0.865
	A5	2.75	1.270	246.6	326	413.61	1.612
					SUM:	2385.98	9.550
	$CF_{Heat\,\#\,299L44} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (2385.98) \div (9.550) =$ **249.8°F**						
Surveillance Weld Material (Heat # 72445)	CR3-LG1	0.510	0.812	139	153	124.24	0.659
	CR3-LG2	1.67	1.141	164	178	203.15	1.303
	W1	0.780	0.930	138	141	131.17	0.865
	PB-1: V	0.634	0.872	107	107	93.34	0.761
	PB-1: S	0.829	0.947	165	165	156.32	0.898
	PB-1: R	2.19	1.213	155	155	187.48	1.471
	PB-1: T	2.23	1.217	181	172	209.86	1.482
					SUM:	1105.56	7.438
	$CF_{Heat\,\#\,72445} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (1105.56) \div (7.438) =$ **148.6°F**						

Notes:

(a) Fluence and ΔRT_{NDT} data taken from Tables 3-7 and 3-8, unless otherwise noted.

(b) Surry Unit 1 Lower Shell Plate C4415-1 capsule fluence values obtained from Section 2. ΔRT_{NDT} values obtained from BAW-2324, Revision 0 [Ref. 21].

(c) FF = fluence factor = $f^{(0.28-0.10*\log(f))}$.

(d) The surveillance weld ΔRT_{NDT} values have been adjusted, as applicable, first by adding the temperature adjustment, then by multiplying by a ratio determined using the ratio procedure to account for differences in the surveillance weld chemistry and the reactor vessel weld chemistry. Pre-adjusted values are listed in the ΔRT_{NDT} column. Temperature adjustment = $1.0*(T_{capsule} - T_{plant})$, where T_{plant} = 542°F for Surry Unit 1 and $T_{capsule}$ is the irradiation temperature in Table 3-7 or 3-8. The temperature adjustment procedure is not utilized when plant-specific capsules are analyzed alone. The ratio procedure is applicable only to surveillance welds and the ratio applied = $CF_{Vessel\ Weld} / CF_{Surv.\ Weld}$. If the ratio procedure yields a ratio less than 1, a ratio of 1.00 is utilized; this approach is conservative.

Table 3-10 Summary of the Surry Unit 1 RPV Beltline, Extended Beltline, and Surveillance Material CF Values based on Regulatory Guide 1.99, Revision 2, Position 1.1 and Position 2.1

RPV Material	Chemistry Factor Position 1.1 (°F)	Chemistry Factor Position 2.1 (°F)
Reactor Vessel Beltline Materials		
Upper Shell Forging 122V109VA1	76.1	---
Intermediate Shell Plate C4326-1	73.5	---
Intermediate Shell Plate C4326-2	73.5	---
Lower Shell Plate C4415-1	66.6	83.1[a]
Lower Shell Plate C4415-2	73.0	83.1[a]
Upper to Intermediate Shell Circumferential Weld (Heat # 25017)	152.0	---
Intermediate Shell Longitudinal Welds L3 and L4 (Heat # 8T1554)	143.9[c]	---
Intermediate to Lower Shell Circumferential Weld (Heat # 72445)	158.0[c]	148.6[c]
Lower Shell Longitudinal Weld L1 (Heat # 8T1554)	143.9[c]	---
Lower Shell Longitudinal Weld L2 (Heat # 299L44)	220.6[c]	249.8[c]
Reactor Vessel Extended Beltline Materials[b]		
Inlet Nozzle 1 (Heat # 9-4787)	123.5	---
Inlet Nozzle 2 (Heat # 9-5078)	123.7	---
Inlet Nozzle 3 (Heat # 9-4819)	123.4	---
Outlet Nozzle 1 (Heat # 9-4825-1)	123.5	---
Outlet Nozzle 2 (Heat # 9-4762)	123.3	---
Outlet Nozzle 3 (Heat # 9-4788)	123.4	---
Inlet Nozzle to Upper Shell Welds – Heat # 299L44	220.6	249.8
Inlet Nozzle to Upper Shell Welds – Heat # 8T1762	152.4	---
Outlet Nozzle to Upper Shell Welds – Heat # 8T1762	152.4	---
Outlet Nozzle to Upper Shell Welds – Heat # 8T1554B	143.9	---
Reactor Vessel Surveillance Materials		
Lower Shell Plate C4415-1	66.6	---
Surveillance Weld (Heat # 299L44)	175.8	---

Notes:

(a) Since Lower Shell Plate C4415-1 shares a heat number with Lower Shell Plate C4415-2, the surveillance plate results also apply to Lower Shell Plate C4415-2.

(b) The nozzle forging Cu wt. % values were conservatively rounded up to 0.16 for the purposes of CF determination.

(c) Linde 80 weld wire initial RT_{NDT} values were established using master curve data. Per BAW-2308 Revision 1-A SE and Revision 2-A SE [Refs. 26 and 27]. Chemistry Factors must be adjusted to a minimum of 167°F when used in ART calculations. If the Position 1.1 CF is greater than 167°F, it is used in calculations.

Table 3-11 Calculation of Position 2.1 CF Values for Surry Unit 2[(a)]

RPV Material	Capsule	Capsule Fluence (x 10^{19} n/cm^2, E > 1.0 MeV)	FF[(c)]	ΔRT_{NDT} (°F)	Adjusted ΔRT_{NDT}[(d)] (°F)	FF*Adjusted ΔRT_{NDT} (°F)	FF^2
Lower Shell Plate C4339-1 (Longitudinal)	X	0.297	0.668	59.08	59.08	39.45	0.446
	V	1.89	1.174	79.12	79.12	92.91	1.379
	Y	2.72	1.267	114.22	114.22	144.72	1.605
Lower Shell Plate C4339-1 (Transverse)	X	0.297	0.668	48.67	48.67	32.50	0.446
	V	1.89	1.174	63.60	63.60	74.68	1.379
	Y	2.72	1.267	106.81	106.81	135.33	1.605
					SUM:	519.59	6.860
	$CF_{C4339\text{-}1} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (519.59) \div (6.860) = \mathbf{75.7°F}$						
Surveillance Weld Material (Heat # 0227)	X	0.297	0.668	95.65	95.65	63.86	0.446
	V	1.89	1.174	140.21	140.21	164.64	1.379
	Y	2.72	1.267	178.32	178.32	225.94	1.605
					SUM:	454.45	3.430
	$CF_{Heat\ \#\ 0227} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (454.45) \div (3.430) = \mathbf{132.5°F}$						
Surveillance Weld Material[(b)] (Heat # 72445)	CR3-LG1	0.510	0.812	139	152	123.43	0.659
	CR3-LG2	1.67	1.141	164	177	202.01	1.303
	W1	0.780	0.930	138	140	130.24	0.865
	PB-1: V	0.634	0.872	107	106	92.46	0.761
	PB-1: S	0.829	0.947	165	164	155.37	0.898
	PB-1: R	2.19	1.213	155	154	186.26	1.471
	PB-1: T	2.23	1.217	181	171	208.64	1.482
					SUM:	1098.42	7.438
	$CF_{Heat\ \#\ 72445} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (1098.42) \div (7.438) = \mathbf{147.7°F}$						

Notes:

(a) Fluence and ΔRT_{NDT} data are from WCAP-16001, Revision 0 [Ref. 22], unless otherwise noted.

(b) Fluence and ΔRT_{NDT} data are from Table 3-8.

(c) FF = fluence factor = $f^{(0.28-0.10*\log(f))}$.

(d) The surveillance weld ΔRT_{NDT} values have been adjusted, as applicable, first by adding the temperature adjustment, then by multiplying by a ratio determined using the ratio procedure to account for differences in the surveillance weld chemistry and the reactor vessel weld chemistry. Pre-adjusted values are listed in the ΔRT_{NDT} column. Temperature adjustment = $1.0*(T_{capsule} - T_{plant})$, where T_{plant} = 543°F for Surry Unit 2 and $T_{capsule}$ is the irradiation temperature in Table 3-8. The temperature adjustment procedure is not utilized when plant-specific capsules are analyzed alone. The ratio procedure is applicable only to surveillance welds and the ratio applied = $CF_{Vessel\ Weld}$ / $CF_{Surv.\ Weld}$. If the ratio procedure yields a ratio less than 1, a ratio of 1.00 is utilized; this approach is conservative.

Table 3-12 Summary of the Surry Unit 2 RPV Beltline, Extended Beltline, and Surveillance Material CF Values based on Regulatory Guide 1.99, Revision 2, Position 1.1 and Position 2.1

<table>
<tr><th colspan="2" rowspan="2">RPV Material</th><th colspan="2">Chemistry Factor</th></tr>
<tr><th>Position 1.1 (°F)</th><th>Position 2.1 (°F)</th></tr>
<tr><td colspan="4">Reactor Vessel Beltline Materials</td></tr>
<tr><td colspan="2">Upper Shell Forging 123V303VA1</td><td>75.8</td><td>---</td></tr>
<tr><td colspan="2">Intermediate Shell Plate C4331-2</td><td>83.0</td><td>---</td></tr>
<tr><td colspan="2">Intermediate Shell Plate C4339-2</td><td>73.4</td><td>75.7(a)</td></tr>
<tr><td colspan="2">Lower Shell Plate C4208-2</td><td>107.3</td><td>---</td></tr>
<tr><td colspan="2">Lower Shell Plate C4339-1</td><td>70.8</td><td>75.7(a)</td></tr>
<tr><td colspan="2">Upper to Intermediate Shell Circumferential Weld (Heat # 4275)</td><td>160.5</td><td>---</td></tr>
<tr><td colspan="2">Intermediate Shell Longitudinal Welds L3 and L4 (OD 50%) (Heat # 72445)</td><td>158.0(c)</td><td>147.7(c)</td></tr>
<tr><td colspan="2">Intermediate Shell Longitudinal Weld L4 (ID 50%) (Heat # 8T1762)</td><td>152.4(c)</td><td>---</td></tr>
<tr><td colspan="2">Intermediate to Lower Shell Circumferential Weld (Heat # 0227)</td><td>147.5</td><td>132.5</td></tr>
<tr><td colspan="2">Lower Shell Longitudinal Welds L1 and L2 (Heat # 8T1762)</td><td>152.4(c)</td><td>---</td></tr>
<tr><td colspan="4">Reactor Vessel Extended Beltline Materials(b)</td></tr>
<tr><td colspan="2">Inlet Nozzle 1 (Heat # 9-5104)</td><td>123.4</td><td>---</td></tr>
<tr><td colspan="2">Inlet Nozzle 2 (Heat # 9-4815)</td><td>123.7</td><td>---</td></tr>
<tr><td colspan="2">Inlet Nozzle 3 (Heat # 9-5205)</td><td>123.6</td><td>---</td></tr>
<tr><td colspan="2">Outlet Nozzle 1 (Heat # 9-4825-2)</td><td>123.5</td><td>---</td></tr>
<tr><td colspan="2">Outlet Nozzle 2 (Heat # 9-5086-1)</td><td>123.6</td><td>---</td></tr>
<tr><td colspan="2">Outlet Nozzle 3 (Heat # 9-5086-2)</td><td>123.7</td><td>---</td></tr>
<tr><td>Inlet Nozzle to Upper Shell Welds</td><td>Heat # 8T1762</td><td>152.4</td><td>---</td></tr>
<tr><td>Outlet Nozzle to Upper Shell Welds</td><td>Rotterdam</td><td>272.0</td><td>---</td></tr>
<tr><td colspan="4">Reactor Vessel Surveillance Materials</td></tr>
<tr><td colspan="2">Lower Shell Plate C4339-1</td><td>70.8</td><td>---</td></tr>
<tr><td colspan="2">Surveillance Weld (Heat # 0227)</td><td>150.8</td><td>---</td></tr>
</table>

Note:

(a) Since Lower Shell Plate C4339-1 shares a heat number with Intermediate Shell Plate C4339-2, the surveillance plate results also apply to Intermediate Shell Plate C4339-2.

(b) The nozzle forging Cu wt. % values were conservatively rounded up to 0.16 for the purposes of CF determination.

(c) Linde 80 weld wire initial RT_{NDT} values were established using master curve data. Per BAW-2308 Revision 1-A SE and Revision 2-A SE [Refs. 26 and 27] Chemistry Factors must be adjusted to a minimum of 167°F when used in ART calculations. If the Position 1.1 CF is greater than 167°F, it will be used in calculations.

4 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

4.1 OVERALL APPROACH

The ASME (American Society of Mechanical Engineers) approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{Ic}, for the metal temperature at that time. K_{Ic} is obtained from the reference fracture toughness curve, defined in the 1998 Edition through 2000 Addenda of Section XI, Appendix G of the ASME Code [Ref. 4]. The K_{Ic} curve is given by the following equation:

$$K_{Ic} = 33.2 + 20.734 * e^{[0.02(T - RT_{NDT})]} \quad (1)$$

where,

K_{Ic} (ksi√in.) = reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature RT_{NDT}

This K_{Ic} curve is based on the lower bound of static critical K_I values measured as a function of temperature on specimens of SA-533 Grade B Class 1, SA-508-1, SA-508-2, and SA-508-3 steel.

4.2 METHODOLOGY FOR PRESSURE-TEMPERATURE LIMIT CURVE DEVELOPMENT

The governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C^* K_{Im} + K_{It} < K_{Ic} \quad (2)$$

where,

K_{Im} = stress intensity factor caused by membrane (pressure) stress

K_{It} = stress intensity factor caused by the thermal gradients

K_{Ic} = reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature RT_{NDT}

C = 2.0 for Level A and Level B service limits

C = 1.5 for hydrostatic and leak test conditions during which the reactor core is not critical

For membrane tension, the corresponding K_I for the postulated defect is:

$$K_{Im} = M_m \times (pR_i / t) \qquad (3)$$

where, M_m for an inside axial surface flaw is given by:

M_m = 1.85 for $\sqrt{t} < 2$,
M_m = $0.926\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$,
M_m = 3.21 for $\sqrt{t} > 3.464$

and, M_m for an outside axial surface flaw is given by:

M_m = 1.77 for $\sqrt{t} < 2$,
M_m = $0.893\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$,
M_m = 3.09 for $\sqrt{t} > 3.464$

Similarly, M_m for an inside or an outside circumferential surface flaw is given by:

M_m = 0.89 for $\sqrt{t} < 2$,
M_m = $0.443\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$,
M_m = 1.53 for $\sqrt{t} > 3.464$

where,

p = internal pressure (ksi), R_i = vessel inner radius (in), and t = vessel wall thickness (in).

For bending stress, the corresponding K_I for the postulated axial or circumferential defect is:

K_{Ib} = M_b * Maximum Bending Stress, where M_b is two-thirds of M_m (4)

The maximum K_I produced by radial thermal gradient for the postulated axial or circumferential inside surface defect of G-2120 is:

$$K_{It} = 0.953 \times 10^{-3} \times CR \times t^{2.5} \qquad (5)$$

where CR is the cooldown rate in °F/hr., or for a postulated axial or circumferential outside surface defect

$$K_{It} = 0.753 \times 10^{-3} \times HU \times t^{2.5} \qquad (6)$$

where HU is the heatup rate in °F/hr.

The through-wall temperature difference associated with the maximum thermal K_I can be determined from ASME Code, Section XI, Appendix G, Fig. G-2214-1. The temperature at any radial distance from the vessel surface can be determined from ASME Code, Section XI, Appendix G, Fig. G-2214-2 for the maximum thermal K_I.

(a) The maximum thermal K_I relationship and the temperature relationship in Fig. G-2214-1 are applicable only for the conditions given in G-2214.3(a)(1) and (2).

(b) Alternatively, the K_I for radial thermal gradient can be calculated for any thermal stress distribution and at any specified time during cooldown for a ¼-thickness axial or circumferential inside surface defect using the relationship:

$$K_{It} = (1.0359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3) * \sqrt{\pi a} \tag{7}$$

or similarly, K_{It} during heatup for a ¼-thickness outside axial or circumferential surface defect using the relationship:

$$K_{It} = (1.043C_0 + 0.630C_1 + 0.481C_2 + 0.401C_3) * \sqrt{\pi a} \tag{8}$$

where the coefficients C_0, C_1, C_2, and C_3 are determined from the thermal stress distribution at any specified time during the heatup or cooldown using the form:

$$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3 \tag{9}$$

and x is a variable that represents the radial distance (in) from the appropriate (i.e., inside or outside) surface to any point on the crack front, and a is the maximum crack depth (in).

Note that Equations 3, 7, and 8 were implemented in the OPERLIM computer code, which is the program used to generate the pressure-temperature (P-T) limit curves. The P-T curve methodology is the same as that described in WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" [Ref. 3] Section 2.6 (equations 2.6.2-4 and 2.6.3-1). Finally, the reactor vessel metal temperature at the crack tip of a postulated flaw is determined based on the methodology contained in Section 2.6.1 of WCAP-14040-A, Revision 4 (equation 2.6.1-1). This equation is solved utilizing values for thermal diffusivity of 0.518 ft^2/hr at 70°F and 0.379 ft^2/hr at 550°F and a constant convective heat-transfer coefficient value of 7000 Btu/hr-ft^2-°F.

At any time during the heatup or cooldown transient, K_{Ic} is determined by the metal temperature at the tip of a postulated flaw (the postulated flaw has a depth of 1/4 of the section thickness and a length of 1.5 times the section thickness per ASME Code, Section XI, Paragraph G-2120), the appropriate value for RT_{NDT}, and the reference fracture toughness curve (Equation 1). The thermal stresses resulting from the temperature gradients through the vessel wall are calculated, and then the corresponding (thermal) stress intensity factors, K_{It}, for the reference flaw are computed. From Equation 2, the pressure stress intensity factors are obtained, and from these the allowable pressures are calculated.

For the calculation of the allowable pressure versus coolant temperature during cooldown, the reference 1/4T flaw of Appendix G to Section XI of the ASME Code is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the vessel wall because the thermal gradients, which increase with increasing cooldown rates, produce tensile stresses at the inside surface that would tend to open (propagate) the existing flaw. Allowable pressure-temperature curves are generated for steady-state (zero-rate) and each finite cooldown rate specified. From these curves, composite limit curves are constructed as the minimum of the steady-state or finite rate curve for each cooldown rate specified.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on the measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel inner diameter. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the ΔT (temperature) across the vessel wall developed during cooldown results in a higher value of K_{Ic} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist so that the increase in K_{Ic} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location, and therefore allowable pressures could be lower if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and ensures conservative operation of the system for the entire cooldown period.

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the wall. The heatup results in compressive stresses at the inside surface that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{Ic} for the inside 1/4T flaw during heatup is lower than the K_{Ic} for the flaw during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist so that the effects of compressive thermal stresses and lower K_{Ic} values do not offset each other, and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to ensure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The third portion of the heatup analysis concerns the calculation of the pressure-temperature limitations for the case in which a 1/4T flaw located at the 1/4T location from the outside surface is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and therefore tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for the steady-state and finite heatup rate situations, the final limit curves are produced by constructing a composite curve based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the least of the three values taken from the curves under consideration. The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist wherein, over the course of the heatup ramp, the controlling condition switches from the inside to the outside, and the pressure limit must at all times be based on analysis of the most critical criterion.

4.3 PRESSURE CORRECTION

The current Surry Units 1 and 2 heatup and cooldown limit curves in the Surry Power Station Technical Specifications [Ref. 1] include a pressure correction value of 21.5 psi. This pressure correction later was applied to the curves developed in WCAP-14177 [Ref. 23] to account for the pressure difference between the location of pressure measurement and the reactor vessel. See Appendix F for details. This pressure correction has not been incorporated into the heatup and cooldown limit curves developed in Section 6 of this report. The pressure correction value has been removed from the current Technical Specification heatup and cooldown limit curves in Section 7 to appropriately compare the current Technical Specification P-T limit curves to the P-T limit curves developed in this report.

4.4 LOWEST SERVICE TEMPERATURE REQUIREMENTS

Surry Units 1 and 2 are Westinghouse-designed plants; thus, the primary Reactor Coolant System (RCS) piping is stainless steel. Therefore, the lowest service temperature requirements of Paragraph NB-2332 of ASME Code Section III [Ref. 17] do not apply to the Surry Units 1 and 2 reactor vessels. See Appendix C for additional details.

4.5 CLOSURE HEAD/VESSEL FLANGE REQUIREMENTS

10 CFR Part 50, Appendix G [Ref. 5] addresses the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure head regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the preservice hydrostatic test pressure, which is calculated to be 621 psig. The initial RT_{NDT} values of the reactor vessel closure head and vessel flange are documented in Tables 3-5 and 3-6. The limiting unirradiated RT_{NDT} of -60°F is associated with the Surry Unit 2 replacement reactor vessel closure head, so the minimum allowable temperature of this region is 60°F at pressures greater than 621 psig (without margins for instrument uncertainties). This limit is shown in Tables 6-1 and 6-2.

4.6 BOLTUP TEMPERATURE REQUIREMENTS

The minimum boltup temperature is the minimum allowable temperature at which the reactor vessel closure head bolts can be preloaded. It is determined by the highest reference temperature, RT_{NDT}, in the closure flange region. This requirement is established in Appendix G to 10 CFR 50 [Ref. 5]. Per the NRC-approved methodology in WCAP-14040-A, Revision 4 [Ref. 3], the minimum boltup temperature is 60°F or the limiting unirradiated RT_{NDT} of the closure flange region, whichever is higher. Since the limiting unirradiated RT_{NDT} of this region is below 60°F per Tables 3-5 and 3-6, the recommended minimum boltup temperature for the Surry Units 1 and 2 reactor vessel is 60°F (without margins for instrument uncertainties). It is noted that the boltup temperature is procedurally controlled at Surry Units 1 and 2 independent from the Technical Specification curves.

5 CALCULATION OF ADJUSTED REFERENCE TEMPERATURE

From Regulatory Guide 1.99, Revision 2 [Ref. 2], the adjusted reference temperature (ART) for each material in the beltline region is given by the following expression:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \qquad (10)$$

Initial RT_{NDT} is the reference temperature for the unirradiated material as defined in Paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code [Ref. 17]. If measured values of the initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may be used, provided there are sufficient test results to establish a mean and standard deviation for the class.

ΔRT_{NDT} is the mean value of the adjustment in reference temperature caused by irradiation and should be calculated as follows:

$$\Delta RT_{NDT} = CF * f^{(0.28 - 0.10 \log f)} \qquad (11)$$

To calculate ΔRT_{NDT} at any depth (e.g., at 1/4T or 3/4T), the following formula must first be used to attenuate the fluence at the specific depth:

$$f_{(depth\ x)} = f_{surface} * e^{(-0.24x)} \qquad (12)$$

where x inches (reactor vessel cylindrical shell beltline thickness is 8.05 inches) is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then placed in Equation 11 to calculate the ΔRT_{NDT} at the specific depth.

The projected reactor vessel neutron fluence was updated for this analysis and documented in Section 2 of this report. The evaluation methods used in Section 2 are consistent with the methods presented in WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" [Ref. 3].

Tables 5-1 and 5-2 contain the surface fluence values at 68 EFPY, which were used for the development of the P-T limit curves contained in this report. Tables 5-1 and 5-2 also contain the 1/4T and 3/4T calculated fluence values and fluence factors (FFs), per Regulatory Guide 1.99, Revision 2. The values in this table will be used to calculate the 68 EFPY ART values for the Surry Units 1 and 2 reactor vessel materials.

Margin is calculated as $M = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2}$. The standard deviation for the initial RT_{NDT} margin term (σ_I) is 0°F when the initial RT_{NDT} is a measured value and 17°F when a generic value is available, unless a material-specific σ_I is calculated. The standard deviation for the ΔRT_{NDT} margin term, σ_Δ, is 17°F for plates or forgings when surveillance data is not used or is non-credible, and 8.5°F (half the value) for plates or forgings when credible surveillance data is used. For welds, σ_Δ is equal to 28°F when surveillance capsule data is not used or is non-credible, and is 14°F (half the value) when credible surveillance capsule data is used. The value for σ_Δ need not exceed 0.5 times the mean value of ΔRT_{NDT}.

However, for the welds utilizing "Master Curve"-based initial RT_{NDT} values, σ_{Δ} is set equal to 28°F per the safety evaluations (SEs) associated with BAW-2308, Revision 1 and 2 [Refs. 26 and 27].

Contained in Tables 5-3 through 5-6 are the 68 EFPY ART calculations at the 1/4T and 3/4T locations for generation of the Surry Units 1 and 2 heatup and cooldown curves.

Surry Unit 1 Inlet Nozzle 1 and Surry Unit 2 Inlet Nozzle 1 and Outlet Nozzle 3 have projected fluence values that exceed the 1 x 10^{17} n/cm^2 fluence threshold at the 1/4T flaw location at 68 EFPY per Tables 2-3 and 2-4. Therefore, per NRC RIS 2014-11 [Ref. 8], neutron radiation embrittlement must be considered herein for these nozzle forging materials. For conservatism, embrittlement is considered for each nozzle forging material. The nozzle forging ART values are calculated using surface fluence values at the 1/4T flaw location for each specific nozzle. Thus, ART calculations for the Surry Units 1 and 2 inlet and outlet nozzle forging materials utilizing the 1/4T and 3/4T fluence values are excluded from Tables 5-3 through 5-6. ART values for the nozzle forging materials are contained in Appendix B.

Finally, the second conclusion of TLR-RES/DE/CIB-2013-01 [Ref. 28] states that if ΔRT_{NDT} is calculated to be less than 25°F, then embrittlement need not be considered. This conclusion was applied, as necessary, to the ART calculations documented in Tables 5-3 through 5-6.

The limiting ART values for Surry Units 1 and 2 to be used in the generation of the P-T limit curves are based on multiple materials, since a combination of axial and circumferential flaw materials have the most limiting 1/4T and 3/4T ART values. The limiting ART values for Surry Units 1 and 2 are summarized in Table 5-7. The limiting ART values are <u>less</u> than the ART values utilized to develop the current Surry Units 1 and 2 Technical Specification curves [Ref. 1]. <u>Thus, the applicability of the P-T limit curves in the Surry Units 1 and 2 Technical Specifications (based on WCAP-14177 [Ref. 23]) can be extended to 68 EFPY.</u>

Section 7 provides further justification that the applicability of the current Surry Units 1 and 2 Technical Specifications P-T limit curves can be extended to 68 EFPY by directly comparing the 68 EFPY P-T limit curves developed in Section 6 to the 48 EFPY curves in the Surry Units 1 and 2 Technical Specifications.

Table 5-1 Fluence Values and Fluence Factors for the Vessel Surface, 1/4T and 3/4T Locations for the Surry Unit 1 Reactor Vessel Materials at 68 EFPY

Reactor Vessel Material	Surface Fluence, f [(a)] (x 10^{19} n/cm^2, E > 1.0 MeV)	1/4T f (x 10^{19} n/cm^2, E > 1.0 MeV)	1/4T FF	3/4T f (x 10^{19} n/cm^2, E > 1.0 MeV)	3/4T FF
Reactor Vessel Beltline Materials					
Upper Shell Forging 122V109VA1	0.754	0.465	0.787	0.177	0.541
Upper to Intermediate Shell Circ. Weld (Heat # 25017)	0.754	0.465	0.787	0.177	0.541
Intermediate Shell Plates C4326-1 and C4326-2	6.29	3.88	1.350	1.48	1.108
Lower Shell Plates C4415-1 and C4415-2	6.35	3.92	1.352	1.49	1.111
Intermediate Shell Longitudinal Welds L3 and L4 (Heat # 8T1554)	1.25	0.771	0.927	0.294	0.665
Lower Shell Longitudinal Welds L1 (Heat # 8T1554) and L2 (Heat # 299L44)	1.26	0.777	0.929	0.296	0.667
Intermediate to Lower Shell Circumferential Weld (Heat # 72445)	6.31	3.89	1.350	1.48	1.109
Reactor Vessel Extended Beltline Materials [(b)]					
Inlet Nozzle 1 to Upper Shell Weld (Heats # 299L44 and # 8T1762)	0.0304	0.0188	0.165	0.00714	0.087
Inlet Nozzle 2 to Upper Shell Weld (Heats # 299L44 and # 8T1762)	0.00784	0.00484	0.065	0.00184	0.031
Inlet Nozzle 3 to Upper Shell Weld (Heats # 299L44 and # 8T1762)	0.0109	0.00672	0.083	0.00256	0.040
Outlet Nozzle 1 to Upper Shell Weld (Heats # # 8T1762 and # 8T1554B)	0.00813	0.00502	0.067	0.00191	0.032
Outlet Nozzle 2 to Upper Shell Weld (Heats # 8T1762 and # 8T1554B)	0.00586	0.00362	0.052	0.00138	0.024
Outlet Nozzle 3 to Upper Shell Weld (Heats # 8T1762 and # 8T1554B)	0.0227	0.0140	0.137	0.00533	0.070

Notes:

(a) 68 EFPY fluence values are documented in Table 2-3.

(b) Reactor vessel nozzle forgings are excluded from this table – see Appendix B.

Table 5-2 Fluence Values and Fluence Factors for the Vessel Surface, 1/4T and 3/4T Locations for the Surry Unit 2 Reactor Vessel Materials at 68 EFPY

Reactor Vessel Material	Surface Fluence, f [(a)] (x 10^{19} n/cm^2, E > 1.0 MeV)	1/4T f (x 10^{19} n/cm^2, E > 1.0 MeV)	1/4T FF	3/4T f (x 10^{19} n/cm^2, E > 1.0 MeV)	3/4T FF
Reactor Vessel Beltline Materials					
Upper Shell Forging 123V303VA1	0.865	0.534	0.825	0.203	0.573
Upper to Intermediate Shell Circ. Weld (Heat # 4275)	0.865	0.534	0.825	0.203	0.573
Intermediate Shell Plates C4331-2 and C4339-2	7.20	4.44	1.378	1.69	1.145
Lower Shell Plates C4208-2 and C4339-1	7.26	4.48	1.380	1.70	1.147
Intermediate Shell Longitudinal Welds L3 and L4 (Heats # 72445 and 8T1762)	1.29	0.796	0.936	0.303	0.673
Lower Shell Longitudinal Welds L1 and L2 (Heat # 8T1762)	1.30	0.802	0.938	0.305	0.675
Intermediate to Lower Shell Circumferential Weld (Heat # 0227)	7.22	4.45	1.379	1.70	1.145
Reactor Vessel Extended Beltline Materials[(b)]					
Inlet Nozzle 1 to Upper Shell Weld (Heat # 8T1762)	0.0340	0.0210	0.177	0.00798	0.094
Inlet Nozzle 2 to Upper Shell Weld (Heat # 8T1762)	0.00784	0.00484	0.065	0.00184	0.031
Inlet Nozzle 3 to Upper Shell Weld (Heat # 8T1762)	0.0107	0.00660	0.082	0.00251	0.039
Outlet Nozzle 1 to Upper Shell Weld (Rotterdam)	0.00796	0.00491	0.066	0.00187	0.031
Outlet Nozzle 2 to Upper Shell Weld (Rotterdam)	0.00585	0.00361	0.052	0.00137	0.024
Outlet Nozzle 3 to Upper Shell Weld (Rotterdam)	0.0253	0.0156	0.147	0.00594	0.076

Notes:

(a) 68 EFPY fluence values are documented in Table 2-4.

(b) Reactor vessel nozzle forgings are excluded from this table – see Appendix B.

Table 5-3 Adjusted Reference Temperature Evaluation for the Surry Unit 1 Reactor Vessel Beltline and Extended Beltline Materials through 68 EFPY at the 1/4T Location

RPV Material	R.G. 1.99, Rev. 2 Position	Wt. % Cu(a)	Wt. % Ni(a)	CF(a) (°F)	1/4T Fluence(b) (x 10^{19} n/cm², E > 1.0 MeV)	1/4T FF(b)	$RT_{NDT(U)}$(c) (°F)	ΔRT_{NDT}(d) (°F)	σ_I(c) (°F)	σ_Δ(e) (°F)	Margin (°F)	1/4T ART (°F)
Reactor Vessel Beltline Materials												
Upper Shell Forging 122V109VA1	1.1	0.11	0.74	76.1	0.465	0.787	40	59.9	0.0	17.0	34.0	133.9
Upper to Intermediate Shell Circumferential Weld (Heat # 25017)	1.1	0.33	0.10	152.0	0.465	0.787	0	119.6	20.0	28.0	68.8	188.4
Intermediate Shell Plate C4326-1	1.1	0.11	0.55	73.5	3.88	1.350	10	99.2	0.0	17.0	34.0	143.2
Intermediate Shell Plate C4326-2	1.1	0.11	0.55	73.5	3.88	1.350	11.4	99.2	0.0	17.0	34.0	144.6
Intermediate Shell Longitudinal Welds L3 and L4 (Heat # 8T1554)	1.1	0.16	0.57	167.0	0.771	0.927	-48.6	154.8	18.0	28.0	66.6	172.8
Intermediate to Lower Shell Circumferential Weld (Heat # 72445)	1.1	0.22	0.54	167.0	3.89	1.350	-72.5	225.5	12.0	28.0	60.9	**213.9**
Using credible surveillance data	*2.1*	---	---	*167.0*	*3.89*	*1.350*	*-72.5*	*225.5*	*12.0*	*28.0*	*60.9*	***213.9***
Lower Shell Plate C4415-1	1.1	0.102	0.493	66.6	3.92	1.352	20	90.0	0.0	17.0	34.0	144.0
Using credible surveillance data	*2.1*	---	---	*83.1*	*3.92*	*1.352*	*20*	*112.3*	*0.0*	*8.5*	*17.0*	*149.3*
Lower Shell Plate C4415-2	1.1	0.11	0.50	73.0	3.92	1.352	4.6	98.7	0.0	17.0	34.0	137.3
Using credible surveillance data	*2.1*	---	---	*83.1*	*3.92*	*1.352*	*4.6*	*112.3*	*0.0*	*8.5*	*17.0*	*133.9*
Lower Shell Longitudinal Weld L1 (Heat # 8T1554)	1.1	0.16	0.57	167.0	0.777	0.929	-48.6	155.2	18.0	28.0	66.6	173.2
Lower Shell Longitudinal Weld L2 (Heat # 299L44)	1.1	0.34	0.68	220.6	0.777	0.929	-74.3	205.0	12.8	28.0	61.6	192.3
Using credible surveillance data	*2.1*	---	---	*249.8*	*0.777*	*0.929*	*-74.3*	*232.1*	*12.8*	*28.0*	*61.6*	***219.4***

Table 5-3 Adjusted Reference Temperature Evaluation for the Surry Unit 1 Reactor Vessel Beltline and Extended Beltline Materials through 68 EFPY at the 1/4T Location

RPV Material	R.G. 1.99, Rev. 2 Position	Wt. % Cu[(a)]	Wt. % Ni[(a)]	CF[(a)] (°F)	1/4T Fluence[(b)] (x 10^{19} n/cm², E > 1.0 MeV)	1/4T FF[(b)]	$RT_{NDT(U)}$[(c)] (°F)	ΔRT_{NDT}[(d)] (°F)	σ_I[(c)] (°F)	σ_Δ[(e)] (°F)	Margin (°F)	1/4T ART (°F)
Reactor Vessel Extended Beltline Materials												
Inlet Nozzle 1 to Upper Shell Weld (Heat # 299L44)	1.1	0.34	0.68	220.6	0.0188	0.165	-7.0	36.5	20.6	18.2	55.0	84.5
Using credible surveillance data	*2.1*	---	---	*249.8*	*0.0188*	*0.165*	*-7.0*	*41.3*	*20.6*	*14.0*	*49.8*	*84.1*
Inlet Nozzle 2 to Upper Shell Weld (Heat # 299L44)	1.1	0.34	0.68	220.6	0.00484	0.065	-7.0	0.0 (14.4)	20.6	0.0	41.2	34.2
Using credible surveillance data	*2.1*	---	---	*249.8*	*0.00484*	*0.065*	*-7.0*	*0.0 (16.3)*	*20.6*	*0.0*	*41.2*	*34.2*
Inlet Nozzle 3 to Upper Shell Weld (Heat # 299L44)	1.1	0.34	0.68	220.6	0.00672	0.083	-7.0	0.0 (18.3)	20.6	0.0	41.2	34.2
Using credible surveillance data	*2.1*	---	---	*249.8*	*0.00672*	*0.083*	*-7.0*	*0.0 (20.8)*	*20.6*	*0.0*	*41.2*	*34.2*
Inlet Nozzle 1 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.0188	0.165	-4.9	25.2	19.7	12.6	46.8	67.1
Inlet Nozzle 2 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.00484	0.065	-4.9	0.0 (10.0)	19.7	0.0	39.4	34.5
Inlet Nozzle 3 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.00672	0.083	-4.9	0.0 (12.7)	19.7	0.0	39.4	34.5
Outlet Nozzle 1 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.00502	0.067	-4.9	0.0 (10.2)	19.7	0.0	39.4	34.5
Outlet Nozzle 2 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.00362	0.052	-4.9	0.0 (8.0)	19.7	0.0	39.4	34.5
Outlet Nozzle 3 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.0140	0.137	-4.9	0.0 (20.9)	19.7	0.0	39.4	34.5
Outlet Nozzle 1 to Upper Shell Weld (Heat # 8T1554B)	1.1	0.16	0.57	143.9	0.00502	0.067	-4.9	0.0 (9.7)	19.7	0.0	39.4	34.5

Table 5-3 Adjusted Reference Temperature Evaluation for the Surry Unit 1 Reactor Vessel Beltline and Extended Beltline Materials through 68 EFPY at the 1/4T Location

RPV Material	R.G. 1.99, Rev. 2 Position	Wt. % Cu(a)	Wt. % Ni(a)	CF(a) (°F)	1/4T Fluence(b) (x 10^{19} n/cm^2, E > 1.0 MeV)	1/4T FF(b)	$RT_{NDT(U)}$(c) (°F)	ΔRT_{NDT}(d) (°F)	σ_I(c) (°F)	σ_Δ(e) (°F)	Margin (°F)	1/4T ART (°F)
Outlet Nozzle 2 to Upper Shell Weld (Heat # 8T1554B)	1.1	0.16	0.57	143.9	0.00362	0.052	-4.9	0.0 (7.6)	19.7	0.0	39.4	34.5
Outlet Nozzle 3 to Upper Shell Weld (Heat # 8T1554B)	1.1	0.16	0.57	143.9	0.0140	0.137	-4.9	0.0 (19.7)	19.7	0.0	39.4	34.5

Notes:

(a) Chemical composition data taken from Tables 3-1 and 3-2 of this report. Chemistry factor values taken from Table 3-10 of this report.

(b) The 1/4T fluence and 1/4T FF values were taken from Table 5-1.

(c) Initial RT_{NDT} values and σ_I values are from Tables 3-1 and 3-2 of this report.

(d) Calculated ΔRT_{NDT} values less than 25°F have been reduced to zero per TLR-RES/DE/CIB-2013-01 [Ref. 28]; actual calculated ΔRT_{NDT} values are listed in parentheses for these materials.

(a) As summarized in Appendix G of this report, all surveillance data for Surry Unit 1 were deemed credible. Per the guidance of Regulatory Guide 1.99, Revision 2 [Ref. 2], the base metal σ_Δ = 17°F for Position 1.1, and σ_Δ = 8.5°F for Position 2.1 with credible surveillance data. Also per Regulatory Guide 1.99, Revision 2, the weld metal σ_Δ = 28°F for Position 1.1, and with credible surveillance data σ_Δ = 14°F for Position 2.1. However, σ_Δ need not exceed 0.5*ΔRT_{NDT}. For welds utilizing initial RT_{NDT} values based on BAW-2308, σ_Δ = 28°F per References 26 and 27.

Table 5-4 Adjusted Reference Temperature Evaluation for the Surry Unit 1 Reactor Vessel Beltline and Extended Beltline Materials through 68 EFPY at the 3/4T Location

RPV Material	R.G. 1.99, Rev. 2 Position	Wt. % Cu[(a)]	Wt. % Ni[(a)]	CF[(a)] (°F)	3/4T Fluence[(b)] (x 10^{19} n/cm², E > 1.0 MeV)	3/4T FF[(b)]	$RT_{NDT(U)}$[(c)] (°F)	ΔRT_{NDT}[(d)] (°F)	σ_I[(c)] (°F)	σ_Δ[(e)] (°F)	Margin (°F)	3/4T ART (°F)
Reactor Vessel Beltline Materials												
Upper Shell Forging 122V109VA1	1.1	0.11	0.74	76.1	0.177	0.541	40	41.1	0.0	17.0	34.0	115.1
Upper to Intermediate Shell Circumferential Weld (Heat # 25017)	1.1	0.33	0.10	152.0	0.177	0.541	0	82.2	20.0	28.0	68.8	151.0
Intermediate Shell Plate C4326-1	1.1	0.11	0.55	73.5	1.48	1.108	10	81.4	0.0	17.0	34.0	125.4
Intermediate Shell Plate C4326-2	1.1	0.11	0.55	73.5	1.48	1.108	11.4	81.4	0.0	17.0	34.0	126.8
Intermediate Shell Longitudinal Welds L3 and L4 (Heat # 8T1554)	1.1	0.16	0.57	167.0	0.294	0.665	-48.6	111.0	18.0	28.0	66.6	129.0
Intermediate to Lower Shell Circumferential Weld (Heat # 72445)	1.1	0.22	0.54	167.0	1.48	1.109	-72.5	185.2	12.0	28.0	60.9	**173.6**
Using credible surveillance data	*2.1*	---	---	*167.0*	*1.48*	*1.109*	*-72.5*	*185.2*	*12.0*	*28.0*	*60.9*	***173.6***
Lower Shell Plate C4415-1	1.1	0.102	0.493	66.6	1.49	1.111	20	74.0	0.0	17.0	34.0	128.0
Using credible surveillance data	*2.1*	---	---	*83.1*	*1.49*	*1.111*	*20*	*92.3*	*0.0*	*8.5*	*17.0*	*129.3*
Lower Shell Plate C4415-2	1.1	0.11	0.50	73.0	1.49	1.111	4.6	81.1	0.0	17.0	34.0	119.7
Using credible surveillance data	*2.1*	---	---	*83.1*	*1.49*	*1.111*	*4.6*	*92.3*	*0.0*	*8.5*	*17.0*	*113.9*
Lower Shell Longitudinal Weld L1 (Heat # 8T1554)	1.1	0.16	0.57	167.0	0.296	0.667	-48.6	111.3	18.0	28.0	66.6	129.3
Lower Shell Longitudinal Weld L2 (Heat # 299L44)	1.1	0.34	0.68	220.6	0.296	0.667	-74.3	147.1	12.8	28.0	61.6	134.3
Using credible surveillance data	*2.1*	---	---	*249.8*	*0.296*	*0.667*	*-74.3*	*166.5*	*12.8*	*28.0*	*61.6*	***153.8***

Table 5-4 Adjusted Reference Temperature Evaluation for the Surry Unit 1 Reactor Vessel Beltline and Extended Beltline Materials through 68 EFPY at the 3/4T Location

RPV Material	R.G. 1.99, Rev. 2 Position	Wt. % Cu[(a)]	Wt. % Ni[(a)]	CF[(a)] (°F)	3/4T Fluence[(b)] (x 10^{19} n/cm², E > 1.0 MeV)	3/4T FF[(b)]	$RT_{NDT(U)}$[(c)] (°F)	ΔRT_{NDT}[(d)] (°F)	σ_I[(c)] (°F)	σ_Δ[(e)] (°F)	Margin (°F)	3/4T ART (°F)
Reactor Vessel Extended Beltline Materials												
Inlet Nozzle 1 to Upper Shell Weld (Heat # 299L44)	1.1	0.34	0.68	220.6	0.00714	0.087	-7.0	0.0 (19.1)	20.6	0.0	41.2	34.2
Using credible surveillance data	*2.1*	---	---	*249.8*	*0.00714*	*0.087*	*-7.0*	*0.0 (21.7)*	*20.6*	*0.0*	*41.2*	*34.2*
Inlet Nozzle 2 to Upper Shell Weld (Heat # 299L44)	1.1	0.34	0.68	220.6	0.00184	0.031	-7.0	0.0 (6.8)	20.6	0.0	41.2	34.2
Using credible surveillance data	*2.1*	---	---	*249.8*	*0.00184*	*0.031*	*-7.0*	*0.0 (7.7)*	*20.6*	*0.0*	*41.2*	*34.2*
Inlet Nozzle 3 to Upper Shell Weld (Heat # 299L44)	1.1	0.34	0.68	220.6	0.00256	0.040	-7.0	0.0 (8.8)	20.6	0.0	41.2	34.2
Using credible surveillance data	*2.1*	---	---	*249.8*	*0.00256*	*0.040*	*-7.0*	*0.0 (10.0)*	*20.6*	*0.0*	*41.2*	*34.2*
Inlet Nozzle 1 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.00714	0.087	-4.9	0.0 (13.2)	19.7	0.0	39.4	34.5
Inlet Nozzle 2 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.00184	0.031	-4.9	0.0 (4.7)	19.7	0.0	39.4	34.5
Inlet Nozzle 3 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.00256	0.040	-4.9	0.0 (6.1)	19.7	0.0	39.4	34.5
Outlet Nozzle 1 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.00191	0.032	-4.9	0.0 (4.8)	19.7	0.0	39.4	34.5
Outlet Nozzle 2 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.00138	0.024	-4.9	0.0 (3.7)	19.7	0.0	39.4	34.5
Outlet Nozzle 3 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.00533	0.070	-4.9	0.0 (10.7)	19.7	0.0	39.4	34.5
Outlet Nozzle 1 to Upper Shell Weld (Heat # 8T1554B)	1.1	0.16	0.57	143.9	0.00191	0.032	-4.9	0.0 (4.5)	19.7	0.0	39.4	34.5

Table 5-4 Adjusted Reference Temperature Evaluation for the Surry Unit 1 Reactor Vessel Beltline and Extended Beltline Materials through 68 EFPY at the 3/4T Location

RPV Material	R.G. 1.99, Rev. 2 Position	Wt. % Cu[(a)]	Wt. % Ni[(a)]	CF[(a)] (°F)	3/4T Fluence[(b)] (x 10^{19} n/cm^2, E > 1.0 MeV)	3/4T FF[(b)]	$RT_{NDT(U)}$[(c)] (°F)	ΔRT_{NDT}[(d)] (°F)	σ_I[(c)] (°F)	σ_Δ[(e)] (°F)	Margin (°F)	3/4T ART (°F)
Outlet Nozzle 2 to Upper Shell Weld (Heat # 8T1554B)	1.1	0.16	0.57	143.9	0.00138	0.024	-4.9	0.0 (3.5)	19.7	0.0	39.4	34.5
Outlet Nozzle 3 to Upper Shell Weld (Heat # 8T1554B)	1.1	0.16	0.57	143.9	0.00533	0.070	-4.9	0.0 (10.1)	19.7	0.0	39.4	34.5

Notes:

(a) Chemical composition data taken from Tables 3-1 and 3-2 of this report. Chemistry factor values taken from Table 3-10 of this report.

(b) The 3/4T fluence and 3/4T FF values were taken from Table 5-1.

(c) Initial RT_{NDT} values and σ_I values are from Tables 3-1 and 3-2 of this report.

(d) Calculated ΔRT_{NDT} values less than 25°F have been reduced to zero per TLR-RES/DE/CIB-2013-01 [Ref. 28]; actual calculated ΔRT_{NDT} values are listed in parentheses for these materials.

(e) As summarized in Appendix G of this report, all surveillance data for Surry Unit 1 were deemed credible. Per the guidance of Regulatory Guide 1.99, Revision 2 [Ref. 2], the base metal σ_Δ = 17°F for Position 1.1, and σ_Δ = 8.5°F for Position 2.1 with credible surveillance data. Also per Regulatory Guide 1.99, Revision 2, the weld metal σ_Δ = 28°F for Position 1.1, and with credible surveillance data σ_Δ = 14°F for Position 2.1. However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$. For welds utilizing initial RT_{NDT} values based on BAW-2308, σ_Δ = 28°F per References 26 and 27.

Table 5-5 Adjusted Reference Temperature Evaluation for the Surry Unit 2 Reactor Vessel Beltline and Extended Beltline Materials through 68 EFPY at the 1/4T Location

RPV Material	R.G. 1.99, Rev. 2 Position	Wt. % Cu[a]	Wt. % Ni[a]	CF[a] (°F)	1/4T Fluence[b] (x 10^{19} n/cm², E > 1.0 MeV)	1/4T FF[b]	$RT_{NDT(U)}$[c] (°F)	ΔRT_{NDT}[d] (°F)	σ_I[c] (°F)	σ_Δ[e] (°F)	Margin (°F)	1/4T ART (°F)
Reactor Vessel Beltline Materials												
Upper Shell Forging 123V303VA1	1.1	0.11	0.72	75.8	0.534	0.825	30	62.5	0.0	17.0	34.0	126.5
Upper to Intermediate Shell Circumferential Weld (Heat # 4275)	1.1	0.35	0.10	160.5	0.534	0.825	0	132.3	20.0	28.0	68.8	201.2
Intermediate Shell Plate C4331-2	1.1	0.12	0.60	83.0	4.44	1.378	15.0	114.4	0.0	17.0	34.0	163.4
Intermediate Shell Plate C4339-2	1.1	0.11	0.54	73.4	4.44	1.378	7.8	101.2	0.0	17.0	34.0	143.0
Using non-credible surveillance data	*2.1*	---	---	*75.7*	*4.44*	*1.378*	*7.8*	*104.3*	*0.0*	*17.0*	*34.0*	*146.1*
Intermediate Shell Longitudinal Welds L3 and L4 (OD 50%) (Heat # 72445)	1.1	0.22	0.54	167.0	0.796	0.936	-72.5	156.3	12.0	28.0	60.9	144.7
Using credible surveillance data	*2.1*	---	---	*167.0*	*0.796*	*0.936*	*-72.5*	*156.3*	*12.0*	*28.0*	*60.9*	*144.7*
Intermediate Shell Longitudinal Weld L4 (ID 50%) (Heat # 8T1762)	1.1	0.19	0.57	167.0	0.796	0.936	-48.6	156.3	18.0	28.0	66.6	174.3
Intermediate to Lower Shell Circumferential Weld (Heat # 0227)	1.1	0.187	0.545	147.5	4.45	1.379	0	203.4	0.0	28.0	56.0	259.4
Using credible surveillance data	*2.1*	---	---	*132.5*	*4.45*	*1.379*	*0*	*182.7*	*0.0*	*14.0*	*28.0*	***210.7***
Lower Shell Plate C4208-2	1.1	0.15	0.55	107.3	4.48	1.380	-30	148.1	0.0	17.0	34.0	152.1
Lower Shell Plate C4339-1	1.1	0.107	0.53	70.8	4.48	1.380	-4.4	97.7	0.0	17.0	34.0	127.3
Using non-credible surveillance data	*2.1*	---	---	*75.7*	*4.48*	*1.380*	*-4.4*	*104.5*	*0.0*	*17.0*	*34.0*	*134.1*

Table 5-5 Adjusted Reference Temperature Evaluation for the Surry Unit 2 Reactor Vessel Beltline and Extended Beltline Materials through 68 EFPY at the 1/4T Location

RPV Material	R.G. 1.99, Rev. 2 Position	Wt. % Cu[(a)]	Wt. % Ni[(a)]	CF[(a)] (°F)	1/4T Fluence[(b)] (x 10^{19} n/cm^2, E > 1.0 MeV)	1/4T FF[(b)]	$RT_{NDT(U)}$[(c)] (°F)	ΔRT_{NDT}[(d)] (°F)	σ_I[(c)] (°F)	σ_Δ[(e)] (°F)	Margin (°F)	1/4T ART (°F)
Lower Shell Longitudinal Welds L1 and L2 (Heat # 8T1762)	1.1	0.19	0.57	167.0	0.802	0.938	-48.6	156.7	18.0	28.0	66.6	**174.6**
Reactor Vessel Extended Beltline Materials												
Inlet Nozzle 1 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.0210	0.177	-4.9	27.0	19.7	13.5	47.8	69.9
Inlet Nozzle 2 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.00484	0.065	-4.9	0.0 (10.0)	19.7	0.0	39.4	34.5
Inlet Nozzle 3 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.00660	0.082	-4.9	0.0 (12.5)	19.7	0.0	39.4	34.5
Outlet Nozzle 1 to Upper Shell Weld (Rotterdam)	1.1	0.35	1.0	272.0	0.00491	0.066	30	0.0 (18.0)	0.0	0.0	0.0	30.0
Outlet Nozzle 2 to Upper Shell Weld (Rotterdam)	1.1	0.35	1.0	272.0	0.00361	0.052	30	0.0 (14.3)	0.0	0.0	0.0	30.0
Outlet Nozzle 3 to Upper Shell Weld (Rotterdam)	1.1	0.35	1.0	272.0	0.0156	0.147	30	40.0	0.0	20.0	40.0	110.0

Notes:

(a) Chemical composition values taken from Tables 3-3 and 3-4 of this report. Chemistry Factor values taken from Table 3-12 of this report.

(b) The 1/4T fluence and 1/4T FF values were taken from Table 5-2.

(c) Initial RT_{NDT} values and σ_I values are from Tables 3-3 and 3-4 of this report.

(d) Calculated ΔRT_{NDT} values less than 25°F have been reduced to zero per TLR-RES/DE/CIB-2013-01 [Ref. 28]; actual calculated ΔRT_{NDT} values are listed in parentheses for these materials.

(e) Per Appendix G of this report, the surveillance plate data were deemed non-credible, whereas the surveillance data for the weld materials were deemed credible. Per the guidance of Regulatory Guide 1.99, Revision 2 [Ref. 2], the base metal σ_Δ = 17°F for Position 1.1 and for Position 2.1 with non-credible surveillance data. Also per Regulatory Guide 1.99, Revision 2, the weld metal σ_Δ = 28°F for Position 1.1, and with credible surveillance data σ_Δ = 14°F for Position 2.1. However, σ_Δ need not exceed 0.5*ΔRT_{NDT}. For welds utilizing initial RT_{NDT} values based on BAW-2308, σ_Δ = 28°F per References 26 and 27.

Table 5-6 Adjusted Reference Temperature Evaluation for the Surry Unit 2 Reactor Vessel Beltline and Extended Beltline Materials through 68 EFPY at the 3/4T Location

RPV Material	R.G. 1.99, Rev. 2 Position	Wt. % Cu(a)	Wt. % Ni(a)	CF(a) (°F)	3/4T Fluence(b) (x 10^{19} n/cm^2, E > 1.0 MeV)	3/4T FF(b)	$RT_{NDT(U)}$(c) (°F)	ΔRT_{NDT}(d) (°F)	σ_I(c) (°F)	σ_Δ(e) (°F)	Margin (°F)	3/4T ART (°F)
Reactor Vessel Beltline Materials												
Upper Shell Forging 123V303VA1	1.1	0.11	0.72	75.8	0.203	0.573	30	43.4	0.0	17.0	34.0	107.4
Upper to Intermediate Shell Circumferential Weld (Heat # 4275)	1.1	0.35	0.10	160.5	0.203	0.573	0	92.0	20.0	28.0	68.8	160.8
Intermediate Shell Plate C4331-2	1.1	0.12	0.60	83.0	1.69	1.145	15.0	95.0	0.0	17.0	34.0	**144.0**
Intermediate Shell Plate C4339-2	1.1	0.11	0.54	73.4	1.69	1.145	7.8	84.0	0.0	17.0	34.0	125.8
Using non-credible surveillance data	*2.1*	---	---	*75.7*	1.69	*1.145*	*7.8*	*86.6*	*0.0*	*17.0*	34.0	*128.4*
Intermediate Shell Longitudinal Welds L3 and L4 (OD 50%) (Heat # 72445)	1.1	0.22	0.54	167.0	0.303	0.673	-72.5	112.4	12.0	28.0	60.9	100.8
Using credible surveillance data	*2.1*	---	---	*167.0*	*0.303*	*0.673*	*-72.5*	*112.4*	*12.0*	*28.0*	*60.9*	*100.8*
Intermediate Shell Longitudinal Weld L4 (ID 50%) (Heat # 8T1762)	1.1	0.19	0.57	167.0	0.303	0.673	-48.6	112.4	18.0	28.0	66.6	130.3
Intermediate to Lower Shell Circumferential Weld (Heat # 0227)	1.1	0.187	0.545	147.5	1.70	1.145	0	168.9	0.0	28.0	56.0	224.9
Using credible surveillance data	*2.1*	---	---	*132.5*	*1.70*	*1.145*	*0*	*151.8*	*0.0*	*14.0*	*28.0*	***179.8***
Lower Shell Plate C4208-2	1.1	0.15	0.55	107.3	1.70	1.147	-30	123.1	0.0	17.0	34.0	127.1
Lower Shell Plate C4339-1	1.1	0.107	0.53	70.8	1.70	1.147	-4.4	81.2	0.0	17.0	34.0	110.8
Using non-credible surveillance data	*2.1*	---	---	*75.7*	*1.70*	*1.147*	*-4.4*	*86.8*	*0.0*	*17.0*	*34.0*	*116.4*
Lower Shell Longitudinal Welds L1 and L2 (Heat # 8T1762)	1.1	0.19	0.57	167.0	0.305	0.675	-48.6	112.7	18.0	28.0	66.6	130.7

Table 5-6 Adjusted Reference Temperature Evaluation for the Surry Unit 2 Reactor Vessel Beltline and Extended Beltline Materials through 68 EFPY at the 3/4T Location

RPV Material	R.G. 1.99, Rev. 2 Position	Wt. % Cu[(a)]	Wt. % Ni[(a)]	CF[(a)] (°F)	3/4T Fluence[(b)] (x 10^{19} n/cm^2, E > 1.0 MeV)	3/4T FF[(b)]	$RT_{NDT(U)}$[(c)] (°F)	ΔRT_{NDT}[(d)] (°F)	σ_I[(c)] (°F)	σ_Δ[(e)] (°F)	Margin (°F)	3/4T ART (°F)
Reactor Vessel Extended Beltline Materials												
Inlet Nozzle 1 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.00798	0.094	-4.9	0.0 (14.3)	19.7	0.0	39.4	34.5
Inlet Nozzle 2 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.00184	0.031	-4.9	0.0 (4.7)	19.7	0.0	39.4	34.5
Inlet Nozzle 3 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.00251	0.039	-4.9	0.0 (6.0)	19.7	0.0	39.4	34.5
Outlet Nozzle 1 to Upper Shell Weld (Rotterdam)	1.1	0.35	1.0	272.0	0.00187	0.031	30	0.0 (8.4)	0.0	0.0	0.0	30.0
Outlet Nozzle 2 to Upper Shell Weld (Rotterdam)	1.1	0.35	1.0	272.0	0.00137	0.024	30	0.0 (6.5)	0.0	0.0	0.0	30.0
Outlet Nozzle 3 to Upper Shell Weld (Rotterdam)	1.1	0.35	1.0	272.0	0.00594	0.076	30	0.0 (20.7)	0.0	0.0	0.0	30.0

Notes:

(a) Chemical composition values taken from Tables 3-3 and 3-4 of this report. Chemistry Factor values taken from Table 3-12 of this report.

(b) The 3/4T fluence and 3/4T FF values were taken from Table 5-2.

(c) Initial RT_{NDT} values and σ_I values are from Tables 3-3 and 3-4 of this report.

(d) Calculated ΔRT_{NDT} values less than 25°F have been reduced to zero per TLR-RES/DE/CIB-2013-01 [Ref. 28]; actual calculated ΔRT_{NDT} values are listed in parentheses for these materials.

(e) Per Appendix G of this report, the surveillance plate data were deemed non-credible, whereas the surveillance data for the weld materials were deemed credible. Per the guidance of Regulatory Guide 1.99, Revision 2 [Ref. 2], the base metal σ_Δ = 17°F for Position 1.1 and for Position 2.1 with non-credible surveillance data. Also per Regulatory Guide 1.99, Revision 2, the weld metal σ_Δ = 28°F for Position 1.1, and with credible surveillance data σ_Δ = 14°F for Position 2.1. However, σ_Δ need not exceed 0.5*ΔRT_{NDT}. For welds utilizing initial RT_{NDT} values based on BAW-2308, σ_Δ = 28°F per References 26 and 27.

Table 5-7 Summary of the Limiting ART Values for Surry Units 1 and 2 at 68 EFPY[a]

Plant	Limiting Material	1/4T Limiting ART (°F)		3/4T Limiting ART (°F)	
		Existing 48 EFPY Curves Documented in Technical Specifications[b]	**TLAA Evaluation at 68 EFPY**	**Existing 48 EFPY Curves Documented in Technical Specifications[b]**	**TLAA Evaluation at 68 EFPY**
Surry Unit 1	**<u>(Circ Flaw) Circ. Weld:</u>** Intermediate to Lower Shell Circ. Weld, Heat # 72445	228.4	213.9	189.5	173.6
	<u>(Axial Flaw) Long. Weld:</u> Lower Shell Long. Weld L2 Heat # 299L44 (Position 2.1)		**219.4**		153.8
Surry Unit 2	**<u>(Circ Flaw) Circ. Weld:</u>** Intermediate to Lower Shell Circ. Weld, Heat # 0227 (Position 2.1)		210.7		**179.8**
	<u>(Axial Flaw) Plate:</u> Intermediate Shell Plate C4331-2		Not Limiting		144.0
	<u>Axial Flaw) Weld:</u> Lower Shell Longitudinal Weld L1 and L2 Heat # 8T1762		174.6		Not Limiting

Notes:

(a) The overall limiting ART values are shown in bold. Since the limiting 1/4T ART value is an axial flaw material and the limiting 3/4T ART value is a circumferential flaw material, both the limiting axial flaw and limiting circumferential flaw P-T limits were considered. See Section 6 and Appendix H for details.

(b) The limiting 48 EFPY 1/4T and 3/4T ART values in the Technical Specifications correspond to the Surry Unit 1 Intermediate to Lower Shell Circumferential Weld (Heat # 72445). The basis for the P-T limit curves is contained in WCAP-14177, Revision 0 [Ref. 23]; however, the applicability was extended to 48 EFPY in a later analysis. See Appendix F for details.

6 HEATUP AND COOLDOWN PRESSURE-TEMPERATURE LIMIT CURVES

Pressure-temperature limit curves for normal heatup and cooldown of the primary reactor coolant system have been calculated for the pressure and temperature in the reactor vessel cylindrical beltline region using the methods discussed in Sections 4 and 5 of this report. This approved methodology is also presented in WCAP-14040-A, Revision 4 [Ref. 3]. The curves are generated for the purpose of comparing the current Technical Specifications P-T limit curves to P-T limit curves developed using modern techniques with the goal of extending the applicability of the current Surry Units 1 and 2 Technical Specifications P-T limit curves to 68 EFPY.

Figure 6-1 presents the limiting heatup curves without margins for possible instrumentation errors using heatup rates of 20, 40, and 60°F/hr applicable for 68 EFPY, with the flange requirements and using the "Axial Flaw" methodology and the limiting "Axial Flaw" ART values summarized in Table 5-7. Figure 6-2 presents the limiting cooldown curves without margins for possible instrumentation errors using cooldown rates of 0 (steady-state), 20, 40, 60, and 100°F/hr applicable for 68 EFPY, with the flange requirements and using the "Axial Flaw" methodology and the limiting "Axial Flaw" ART values summarized in Table 5-7. The heatup and cooldown curves were generated using the 1998 Edition through the 2000 Addenda ASME Code Section XI, Appendix G. Note that a "Circumferential Flaw" evaluation was also completed to confirm that the "Axial Flaw" methodology and ART values are bounding. See Appendix H for details.

Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown in Figures 6-1 and 6-2. This is in addition to other criteria, which must be met before the reactor is made critical, as discussed in the following paragraphs.

The reactor must not be made critical until pressure-temperature combinations are to the right of the criticality limit line shown in Figure 6-1 (heatup curve only). The straight-line portion of the criticality limit is at the minimum permissible temperature for the 2485 psig in-service hydrostatic test as required by Appendix G to 10 CFR Part 50. The governing equation for the hydrostatic test is defined in the 1998 Edition through the 2000 Addenda ASME Code Section XI, Appendix G as follows.

$$1.5\ K_{Im} < K_{Ic} \qquad (13)$$

where,

K_{Im} is the stress intensity factor covered by membrane (pressure) stress [see page 4-2, Equation (3)],

$K_{Ic} = 33.2 + 20.734\ e^{[0.02\ (T - RT_{NDT})]}$ [see page 4-1 Equation (1)],

T is the minimum permissible metal temperature, and

RT_{NDT} is the metal reference nil-ductility temperature.

The criticality limit curve specifies pressure-temperature limits for core operation in order to provide additional margin during actual power production. The pressure-temperature limits for core operation (except for low power physics tests) are that: 1) the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and 2) the reactor vessel must be at least 40°F higher than the minimum permissible temperature in the corresponding pressure-temperature curve for heatup and cooldown calculated as described in Section 4 of this report. For the heatup and cooldown curves without margins for instrumentation errors, the minimum temperature for the inservice hydrostatic leak tests for the Surry Units 1 and 2 reactor vessel at 68 EFPY is 274°F. This temperature is the minimum permissible temperature at which design pressure can be reached during a hydrostatic test per Equation (13). The vertical line drawn from these points on the pressure-temperature curve, intersecting a curve 40°F higher than the pressure-temperature limit curve, constitutes the limit for core operation for the reactor vessel.

Figures 6-1 and 6-2 define all of the above limits for ensuring prevention of non-ductile failure for the Surry Units 1 and 2 reactor vessel for 68 EFPY with the flange requirements and without instrumentation uncertainties. The data points used for developing the heatup and cooldown P-T limit curves shown in Figures 6-1 and 6-2 are presented in Tables 6-1 and 6-2. The P-T limit curves shown in Figures 6-1 and 6-2 were generated based on the limiting "Axial Flaw" ART values for the cylindrical beltline and extended beltline reactor vessel materials. As discussed in Appendix B, the P-T limits developed for the cylindrical beltline region bound the P-T limits for the reactor vessel inlet and outlet nozzles for Surry Units 1 and 2 at 68 EFPY.

The curves developed in this Section are compared to the P-T limit curves in the current Surry Power Station Technical Specifications [Ref. 1] in Section 7 with the goal of showing that the current Surry Power Station Technical Specifications P-T limit curves are bounding and appropriate for continued use to 68 EFPY. To allow direct comparison, the curves in the current Technical Specifications have been adjusted by 10% to account for the differences in methodology between the utilization of the K_{Ia} and K_{Ic} curves.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Surry Unit 1 Lower Shell Longitudinal Weld L2 (Heat # 299L44, Position 2.1)
LIMITING ART VALUES AT 68 EFPY: 1/4T, 219.4°F (Axial Flaw)
3/4T, 153.8°F (Axial Flaw)

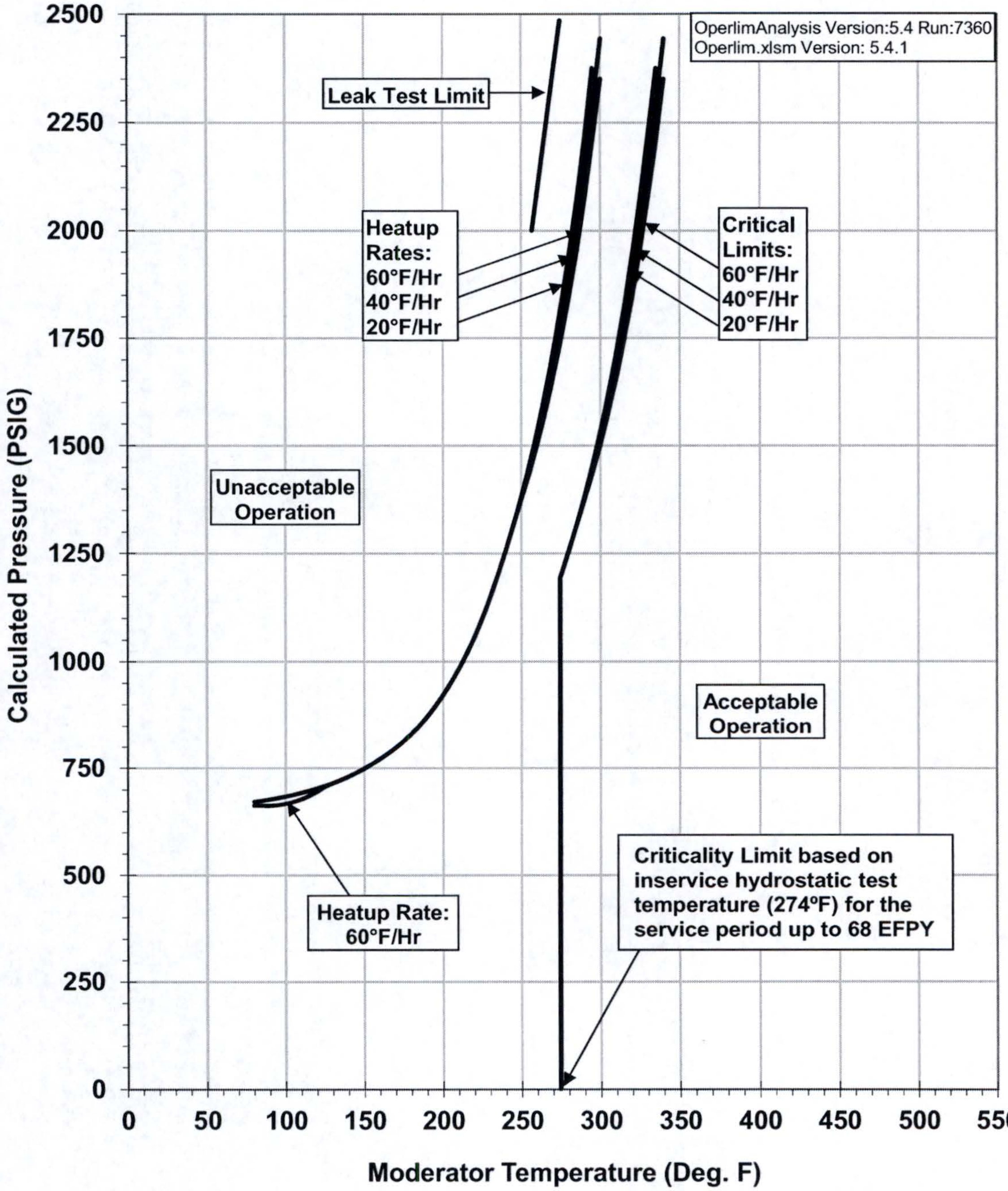


Figure 6-1 Surry Units 1 and 2 Reactor Coolant System Heatup Limitations (Heatup Rates of 20, 40, and 60°F/hr) Applicable for 68 EFPY (with Flange Requirements and without Margins for Instrumentation Errors) using the 1998 Edition through the 2000 Addenda App. G "Axial Flaw" Methodology (w/ K_{Ic})

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Surry Unit 1 Lower Shell Longitudinal Weld L2 (Heat # 299L44, Position 2.1)
LIMITING ART VALUES AT 68 EFPY: 1/4T, 219.4°F (Axial Flaw)
3/4T, 153.8°F (Axial Flaw)

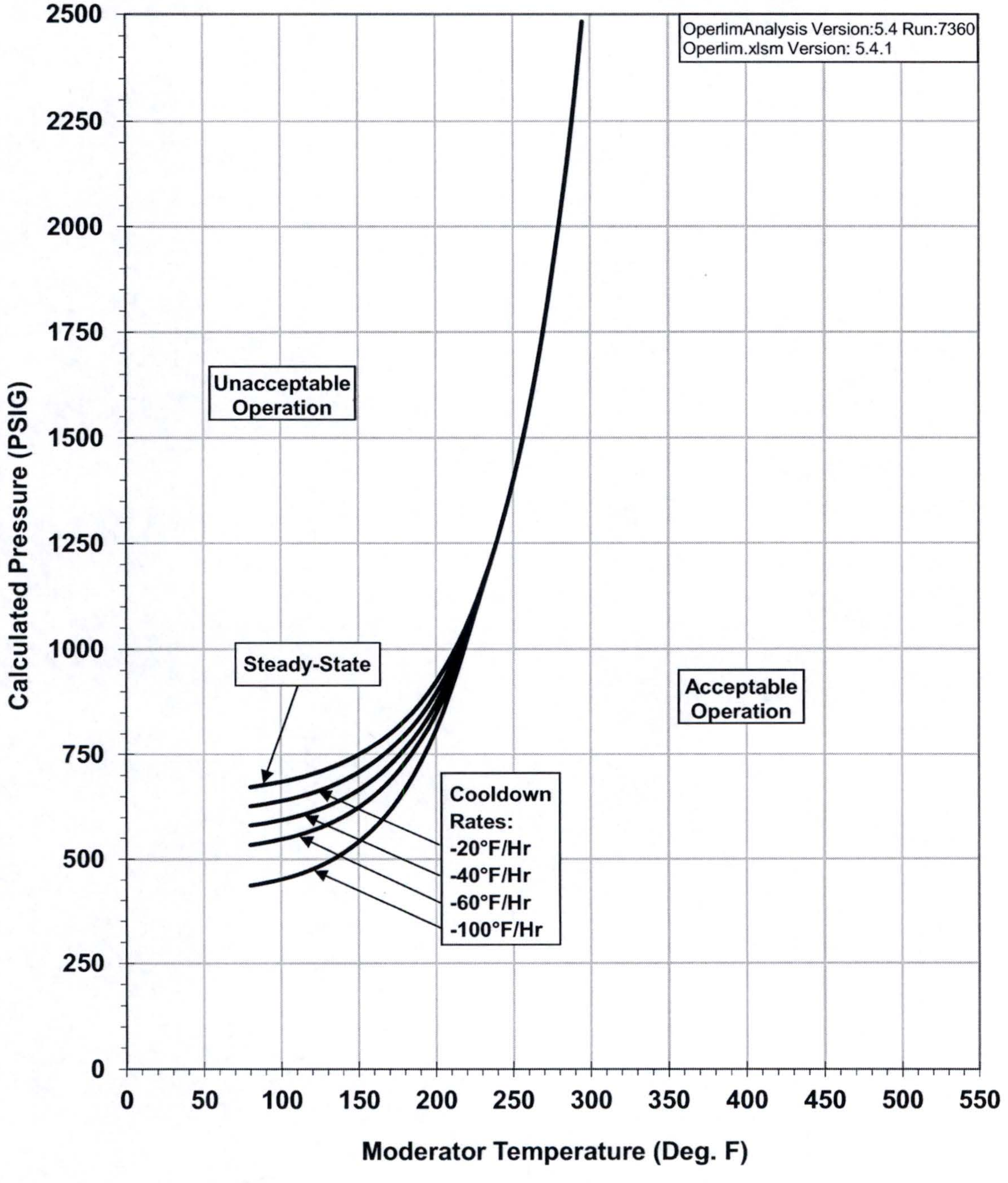


Figure 6-2 Surry Units 1 and 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 20, 40, 60, and 100°F/hr) Applicable for 68 EFPY (with Flange Requirements and without Margins for Instrumentation Errors) using the 1998 Edition through the 2000 Addenda App. G "Axial Flaw" Methodology (w/ K_{Ic})

Table 6-1 Surry Units 1 and 2 68 EFPY Heatup Curve Data Points using the 1998 Edition through the 2000 Addenda App. G Methodology (w/ K_{Ic}, w/ Flange Requirements, and w/o Margins for Instrumentation Errors)

20°F/hr Heatup		20°F/hr Criticality		40°F/hr Heatup		40°F/hr Criticality		60°F/hr Heatup		60°F/hr Criticality	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
60	621	274	0	60	621	274	0	60	621	274	0
60	665	274	1194	60	665	274	1194	60	664	274	1194
65	666	275	1201	65	666	275	1201	65	664	275	1201
70	668	280	1259	70	668	280	1259	70	664	280	1259
75	670	285	1319	75	670	285	1319	75	664	285	1319
80	673	290	1383	80	673	290	1378	80	664	290	1378
85	675	295	1455	85	675	295	1444	85	664	295	1438
90	678	300	1534	90	678	300	1516	90	664	300	1504
95	681	305	1622	95	681	305	1596	95	665	305	1577
100	685	310	1718	100	685	310	1684	100	668	310	1658
105	689	315	1825	105	689	315	1782	105	673	315	1747
110	693	320	1943	110	693	320	1889	110	679	320	1845
115	698	325	2073	115	698	325	2008	115	687	325	1954
120	703	330	2217	120	703	330	2139	120	696	330	2073
125	709	335	2375	125	709	335	2283	125	707	335	2205
130	716			130	716	340	2443	130	716	340	2351
135	723			135	723			135	723		
140	730			140	730			140	730		
145	739			145	739			145	739		
150	749			150	749			150	749		
155	759			155	759			155	759		
160	771			160	771			160	771		
165	784			165	784			165	784		

Table 6-1 Surry Units 1 and 2 68 EFPY Heatup Curve Data Points using the 1998 Edition through the 2000 Addenda App. G Methodology (w/ K_{Ic}, w/ Flange Requirements, and w/o Margins for Instrumentation Errors)

20°F/hr Heatup		20°F/hr Criticality		40°F/hr Heatup		40°F/hr Criticality		60°F/hr Heatup		60°F/hr Criticality	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
170	798			170	798			170	798		
175	814			175	814			175	814		
180	832			180	832			180	832		
185	851			185	851			185	851		
190	873			190	873			190	873		
195	896			195	896			195	896		
200	922			200	922			200	922		
205	951			205	951			205	951		
210	983			210	983			210	983		
215	1018			215	1018			215	1018		
220	1057			220	1057			220	1057		
225	1100			225	1100			225	1100		
230	1148			230	1148			230	1148		
235	1201			235	1201			235	1201		
240	1259			240	1259			240	1259		
245	1319			245	1319			245	1319		
250	1383			250	1378			250	1378		
255	1455			255	1444			255	1438		
260	1534			260	1516			260	1504		
265	1622			265	1596			265	1577		
270	1718			270	1684			270	1658		
275	1825			275	1782			275	1747		
280	1943			280	1889			280	1845		

Table 6-1 Surry Units 1 and 2 68 EFPY Heatup Curve Data Points using the 1998 Edition through the 2000 Addenda App. G Methodology (w/ K_{Ic}, w/ Flange Requirements, and w/o Margins for Instrumentation Errors)

20°F/hr Heatup		20°F/hr Criticality		40°F/hr Heatup		40°F/hr Criticality		60°F/hr Heatup		60°F/hr Criticality	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
285	2073			285	2008			285	1954		
290	2217			290	2139			290	2073		
295	2375			295	2283			295	2205		
				300	2443			300	2351		

Leak Test Limit	
T (°F)	P (psig)
157	2000
274	2485

Table 6-2 Surry Units 1 and 2 68 EFPY Cooldown Curve Data Points using the 1998 Edition through the 2000 Addenda App. G Methodology (w/ K_{Ic}, w/ Flange Requirements, and w/o Margins for Instrumentation Errors)

Steady-State		20°F/hr Cooldown		40°F/hr Cooldown		60°F/hr Cooldown		100°F/hr Cooldown	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
60	621	60	619	60	573	60	526	60	428
60	665	65	621	65	575	65	527	65	429
65	666	70	623	70	576	70	529	70	431
70	668	75	625	75	579	75	531	75	434
75	670	80	627	80	581	80	534	80	436
80	673	85	630	85	584	85	536	85	439
85	675	90	633	90	587	90	540	90	443
90	678	95	636	95	590	95	543	95	446
95	681	100	640	100	594	100	547	100	451
100	685	105	644	105	598	105	551	105	456
105	689	110	648	110	603	110	556	110	462
110	693	115	653	115	608	115	562	115	468
115	698	120	659	120	614	120	568	120	475
120	703	125	665	125	620	125	575	125	483
125	709	130	672	130	628	130	583	130	492
130	716	135	679	135	636	135	592	135	502
135	723	140	688	140	645	140	601	140	514
140	730	145	697	145	655	145	612	145	527
145	739	150	707	150	666	150	624	150	541
150	749	155	719	155	678	155	637	155	557
155	759	160	731	160	692	160	652	160	574
160	771	165	745	165	707	165	669	165	594
165	784	170	761	170	724	170	687	170	616
170	798	175	778	175	742	175	707	175	640
175	814	180	797	180	763	180	730	180	667
180	832	185	818	185	786	185	755	185	697
185	851	190	841	190	811	190	782	190	730
190	873	195	867	195	839	195	813	195	767
195	896	200	895	200	870	200	847	200	808
200	922	205	927	205	905	205	885	205	854
205	951	210	962	210	943	210	926	210	905
210	983	215	1000	215	985	215	973	215	961

Table 6-2 Surry Units 1 and 2 68 EFPY Cooldown Curve Data Points using the 1998 Edition through the 2000 Addenda App. G Methodology (w/ K_{Ic}, w/ Flange Requirements, and w/o Margins for Instrumentation Errors)

Steady-State		20°F/hr Cooldown		40°F/hr Cooldown		60°F/hr Cooldown		100°F/hr Cooldown	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
215	1018	220	1043	220	1031	220	1024	220	1023
220	1057	225	1090	225	1083	225	1081	225	1081
225	1100	230	1142	230	1140	230	1140	230	1140
230	1148	235	1200	235	1200	235	1200	235	1200
235	1201	240	1259	240	1259	240	1259	240	1259
240	1259	245	1323	245	1323	245	1323	245	1323
245	1323	250	1394	250	1394	250	1394	250	1394
250	1394	255	1472	255	1472	255	1472	255	1472
255	1472	260	1559	260	1559	260	1559	260	1559
260	1559	265	1655	265	1655	265	1655	265	1655
265	1655	270	1761	270	1761	270	1761	270	1761
270	1761	275	1878	275	1878	275	1878	275	1878
275	1878	280	2007	280	2007	280	2007	280	2007
280	2007	285	2150	285	2150	285	2150	285	2150
285	2150	290	2308	290	2308	290	2308	290	2308
290	2308	295	2483	295	2483	295	2483	295	2483

7 APPLICABILITY OF CURRENT HEATUP AND COOLDOWN LIMITS

The applicability of the current Surry Units 1 and 2 P-T limit curves was determined based on a comparison of the available operating margin between the P-T limits developed in this report at 68 EFPY with those based on WCAP-14177 [Ref. 23], which are contained in the Surry Power Station Technical Specifications (Figures 3.1-1 and 3.1-2) [Ref. 1]. A summary of the applicability of the Surry Power Station P-T limit curves is provided in Appendix F. The P-T limit curves presented in Figures 3.1-1 and 3.1-2 of the Surry Power Station Technical Specifications do not contain margins for instrumentation error; however, these curves do contain a pressure adjustment of 21.5 psi. In order to provide a direct comparison between the current Technical Specification P-T limit curves and those developed in this report, the pressure adjustment is removed from the Technical Specification curves for comparison purposes only.

The methodology of the 1998 Edition through 2000 Addenda of ASME B&PV Code, Section XI, Appendix G, along with ASME Code Case N-641 was used in the development of the P-T limit curves contained in this report. Code Case N-641 removes some of the conservatism in P-T limit curves by allowing the use of the K_{Ic} reference stress intensity factor, instead of the older, more conservative K_{Ia} reference stress intensity factor, which was used in the development of the Surry Power Station current P-T limit curves. Additionally, the 1998 through the Summer 2000 Addenda Edition of ASME Code Section XI, Appendix G methodology allows use of the less restrictive "Circ-Flaw" methodology, which postulates circumferentially oriented reference defects in circumferential weld materials. Therefore, the P-T limit curves developed in this report took advantage of these updates to the ASME P-T limit methodology and are predicted to contain additional operating margin not present in the curves developed using the older K_{Ia} methodology.

However, when K_{Ic} methodology is used, the LTOP system shall limit the maximum pressure in the vessel to 100% of the pressure determined to satisfy Equation (2) of Section 4. Previously, while using K_{Ia}, the maximum pressure determined from Equation (2) of Section 4 could be exceeded by 10% by the LTOP system. Therefore, since the current curves utilized the K_{Ia} reference stress intensity factor, the P-T limit curve pressure values (without margins for instrumentation error) contained in the Technical Specifications were increased by 10% in order to determine if margin exists between this data and the P-T limit curves developed herein using the K_{Ic} reference stress intensity factor. This 10% increase to the pressure values contained in the Technical Specifications is for comparison purposes only. The increased pressure values are not to be used in actual plant operation. Note that before the 10% increase is applied, the current Surry Power Station P-T limit curve data points were increased by 21.5 psi to remove the pressure adjustment so that direct comparison could be made between these pressure values (current curves without pressure adjustment plus 10% margin) and the pressure values for the curves developed in this report. These adjusted values are shown below in Tables 7-1 and 7-2, for heatup and cooldown, respectively.

Additionally, in order for the current Surry Power Station P-T limit curves to be bounded by the curves developed in this report, the criticality temperatures shown in Section 6 must be found to be lower than the minimum criticality temperature in Technical Specifications. In the Surry Power Station Technical Specifications, the minimum criticality temperature was determined to be 538°F. This value of 538°F bounds the criticality curves developed in Section 6. Therefore, based on this analysis, significant margin exists between the current Surry Units 1 and 2 criticality temperature and the criticality curves determined in this report.

The pressure and temperature values contained in Tables 7-1 and 7-2 (current curves without pressure adjustment plus 10% margin) were plotted together with the data points from Tables 6-1 and 6-2 of this report, which were developed using the K_{Ic} reference stress intensity factor, in Figures 7-1 through 7-3. In Figures 7-1 through 7-3, the curves developed in this report (through 68 EFPY; without margins for instrumentation errors) are shown as solid lines, while the curves developed from the data points in Tables 7-1 and 7-2 (current curves without pressure adjustment plus 10% margin) are shown as dashed lines. The color scheme in the Figures correlates so that the solid and dashed lines have an identical color for each heatup or cooldown rate.

Figure 7-1 shows the comparison of the heatup curves. The corresponding data points, along with the margin between the *current Surry Power Station P-T limit curves +10% margin* and the *P-T limit curves developed in this report* are contained in Table 7-3.

Figure 7-2 shows the comparison of the cooldown curves. Figure 7-3 shows a magnified version of Figure 7-2 in the lower pressure and temperature region. The corresponding data points, along with the margin between the *current Surry Power Station P-T limit curves +10% margin* and the *P-T limit curves developed in this report* are contained in Table 7-4.

Tables 7-5 and 7-6 contain a summary of the available margin between the P-T limits developed in Section 6 of this report (through 68 EFPY; without margins for instrumentation errors) and the current Surry Power Station P-T limits, contained in the Technical Specifications without pressure adjustment, plus 10% margin.

Per Tables 7-5 and 7-6, the minimum pressure difference (at constant temperature) between the current Technical Specifications [Ref. 1] P-T limit curves (plus a 10% margin) and the new curves developed herein is 109 psi. This 109 psi margin applies to the steady-state curves at 80°F and 85°F, as well as the -20°F/hr cooldown rate at 80°F. Using visual comparison of the current Technical Specifications P-T limit curves (plus a 10% margin) and the new curves, shown in Figures 7-1, 7-2, and 7-3 herein, a minimum temperature difference (at constant pressure) of no less than 50°F is identified. These margins of 109 psi and at least 50°F illustrate that adequate margins exist in the current Technical Specifications P-T limit curves to cover typical instrument uncertainties.

P-T Limit Curve Applicability Conclusion

Tables 7-5 and 7-6 show that adequate margin exists between the current Surry Power Station P-T limit curves plus 10% margin (to account for the methodology change between K_{Ia} to K_{Ic}) and the P-T limit curves developed in this report for 68 EFPY. Therefore, the continued use of the current Surry Power Station P-T limit curves as documented in Figures F-1 and F-2 is justified through SLR (68 EFPY).

Low Temperature Overpressure Protection (LTOP) Applicability Conclusion

The maximum allowable Low Temperature Overpressure Protection System (LTOPS) pressurizer Power Operated Relief Valve (PORV) setpoint was calculated to be 399.6 psig for the Surry Units 1 and 2 Subsequent License Renewal (SLR) program. The calculation was performed in accordance with the WCAP-14040-A, Revision 4 [Ref. 3] methodology using critical LTOPS input parameters provided by Dominion, and the limiting axial flaw steady state Appendix G limits calculated for the SLR program at 68 Effective Full Power Years (EFPY).

The evaluation showed that the current Technical Specification value of $\leq$ 390.0 psig is bounding and will remain valid for the SLR program. Since the maximum allowable PORV setpoint for the SLR program was determined using the methodology in Reference 3, this demonstrates that the current licensing basis PORV setpoint that was developed using K_{Ia} Appendix G limits without applying uncertainties was sufficiently conservative.

Summary of Conclusions

- The current P-T limit curves in the Surry Power Station Technical Specifications [Ref. 1] remain valid through 68 EFPY.
- The 21.5 psi adjustment applied to the Technical Specification P-T limit curves remains applicable per Dominion calculations [Refs. 29 and 30]. Note that the LTOP PORV setpoint calculation utilized a conservative 40 psi adjustment.
- The margin between the current P-T limit curves in the Surry Power Station Technical Specifications [Ref. 1] plus 10% and the new K_{Ic} curves developed herein is sufficient to cover typical instrument uncertainties.
- The nozzle P-T limit curves (documented in Appendix B) are bounded by the current Surry Power Station Technical Specifications [Ref. 1] P-T limit curves through 68 EFPY, and other Reactor Coolant Pressure Boundary ferritic components have been addressed (see Appendix C).
- The current Technical Specification PORV setpoint of $\leq$ 390.0 psig remains valid through 68 EFPY.

Table 7-1 Current Surry Power Station P-T Limit Curve Data Points without Pressure Adjustment Plus 10% Margin for Heatup[a]

T (°F)	0°F/hr	20°F/hr	40°F/hr	60°F/hr
	P (psig)	P (psig)	P (psig)	P (psig)
80	564	553	528	503
85	566	553	528	503
90	568	553	528	503
95	571	556	528	503
100	573	559	528	503
105	576	563	528	503
110	579	567	530	503
115	583	572	532	503
120	586	578	536	503
125	590	584	540	505
130	594	590	544	507
135	599	597	550	510
140	604	604	555	514
145	609	609	562	518
150	614	614	569	524
155	620	620	576	529
160	627	627	585	536
165	634	634	594	543
170	641	641	603	551
175	649	649	613	560
180	658	658	625	569
185	667	667	637	579
190	677	677	650	590
195	688	688	663	602
200	699	699	678	615
205	711	711	694	628
210	725	725	712	643
215	739	739	730	659
220	754	754	750	676
225	771	771	771	695
230	788	788	788	715
235	807	807	807	736
240	828	828	828	760
245	850	850	850	784
250	873	873	873	811
255	899	899	899	840
260	926	926	926	871
265	955	955	955	904
270	987	987	987	939
275	1020	1020	1020	978

Table 7-1 Current Surry Power Station P-T Limit Curve Data Points without Pressure Adjustment Plus 10% Margin for Heatup[(a)]

T (°F)	0°F/hr	20°F/hr	40°F/hr	60°F/hr
	P (psig)	P (psig)	P (psig)	P (psig)
280	1057	1057	1057	1018
285	1096	1096	1096	1062
290	1138	1138	1138	1110
295	1183	1183	1183	1161
300	1231	1231	1231	1215
305	1283	1283	1283	1273
310	1339	1339	1339	1336
315	1399	1398	1398	1398
320	1463	1456	1456	1456
325	1532	1514	1514	1514
330	1606	1575	1575	1575
335	1686	1639	1639	1639
340	1771	1709	1709	1709
345	1862	1783	1783	1783
350	1960	1863	1863	1863
355	2065	1949	1949	1949
360	2178	2040	2040	2040
365	2298	2138	2138	2138
370	2426	2242	2242	2242
375	2564	2355	2355	2355
380	2710	2474	2474	2474
385		2602	2602	2602

Note:

(a) Data is associated with the Surry Power Station current heatup curves contained in the Technical Specifications and based on WCAP-14177 [Ref. 23] evaluations. Ten-percent margin was added to the pressure values after the 21.5 psi pressure adjustment was removed. This ten-percent margin on the pressure values is for comparison purposes only and is not to be used in actual plant operation.

Table 7-2 Current Surry Power Station P-T Limit Curve Data Points without Pressure Adjustment Plus 10% Margin for Cooldown[(a)]

T (°F)	0°F/hr	-20°F/hr	-40°F/hr	-60°F/hr	-100°F/hr
	P (psig)	P (psig)	P (psig)	P (psig)	P (psig)
80	564	518	471	423	324
85	566	520	473	425	325
90	568	522	475	427	328
95	571	525	478	430	330
100	573	527	480	432	333
105	576	530	483	435	335
110	579	533	486	438	339
115	583	537	490	442	342
120	586	540	493	445	346
125	590	544	497	450	350
130	594	549	502	454	355
135	599	553	506	459	360
140	604	558	511	464	366
145	609	563	517	470	372
150	614	569	523	476	379
155	620	575	529	482	386
160	627	582	536	490	394
165	634	589	544	497	403
170	641	597	552	506	412
175	649	605	560	515	422
180	658	614	570	525	433
185	667	624	580	536	445
190	677	634	591	547	458
195	688	645	603	560	472
200	699	657	615	573	487
205	711	670	629	587	504
210	725	684	644	603	521
215	739	699	660	620	541
220	754	716	677	638	561
225	771	733	695	658	584
230	788	752	715	679	608
235	807	772	737	702	634
240	828	793	760	727	662
245	850	817	785	753	693
250	873	842	811	782	726
255	899	869	840	813	761
260	926	898	871	846	800
265	955	929	905	882	841
270	987	963	941	920	885
275	1020	999	979	961	933

Table 7-2 Current Surry Power Station P-T Limit Curve Data Points without Pressure Adjustment Plus 10% Margin for Cooldown[(a)]

T (°F)	0°F/hr	-20°F/hr	-40°F/hr	-60°F/hr	-100°F/hr
	P (psig)	P (psig)	P (psig)	P (psig)	P (psig)
280	1057	1038	1021	1006	985
285	1096	1080	1065	1054	1040
290	1138	1124	1113	1106	1100
295	1183	1173	1165	1161	1164
300	1231	1224	1221	1221	1231
305	1283	1280	1281	1283	1283
310	1339	1339	1339	1339	1339
315	1399	1399	1399	1399	1399
320	1463	1463	1463	1463	1463
325	1532	1532	1532	1532	1532
330	1606	1606	1606	1606	1606
335	1686	1686	1686	1686	1686
340	1771	1771	1771	1771	1771
345	1862	1862	1862	1862	1862
350	1960	1960	1960	1960	1960
355	2065	2065	2065	2065	2065
360	2178	2178	2178	2178	2178
365	2298	2298	2298	2298	2298
370	2426	2426	2426	2426	2426
375	2564	2564	2564	2564	2564
380	2710	2710	2710	2710	2710

Note:

(a) Data is associated with the Surry Power Station current cooldown curves contained in the Technical Specifications and based on WCAP-14177 [Ref. 23] evaluations. Ten-percent margin was added to the pressure values after the 21.5 psi pressure adjustment was removed. This ten-percent margin on the pressure values is for comparison purposes only and is not to be used in actual plant operation.

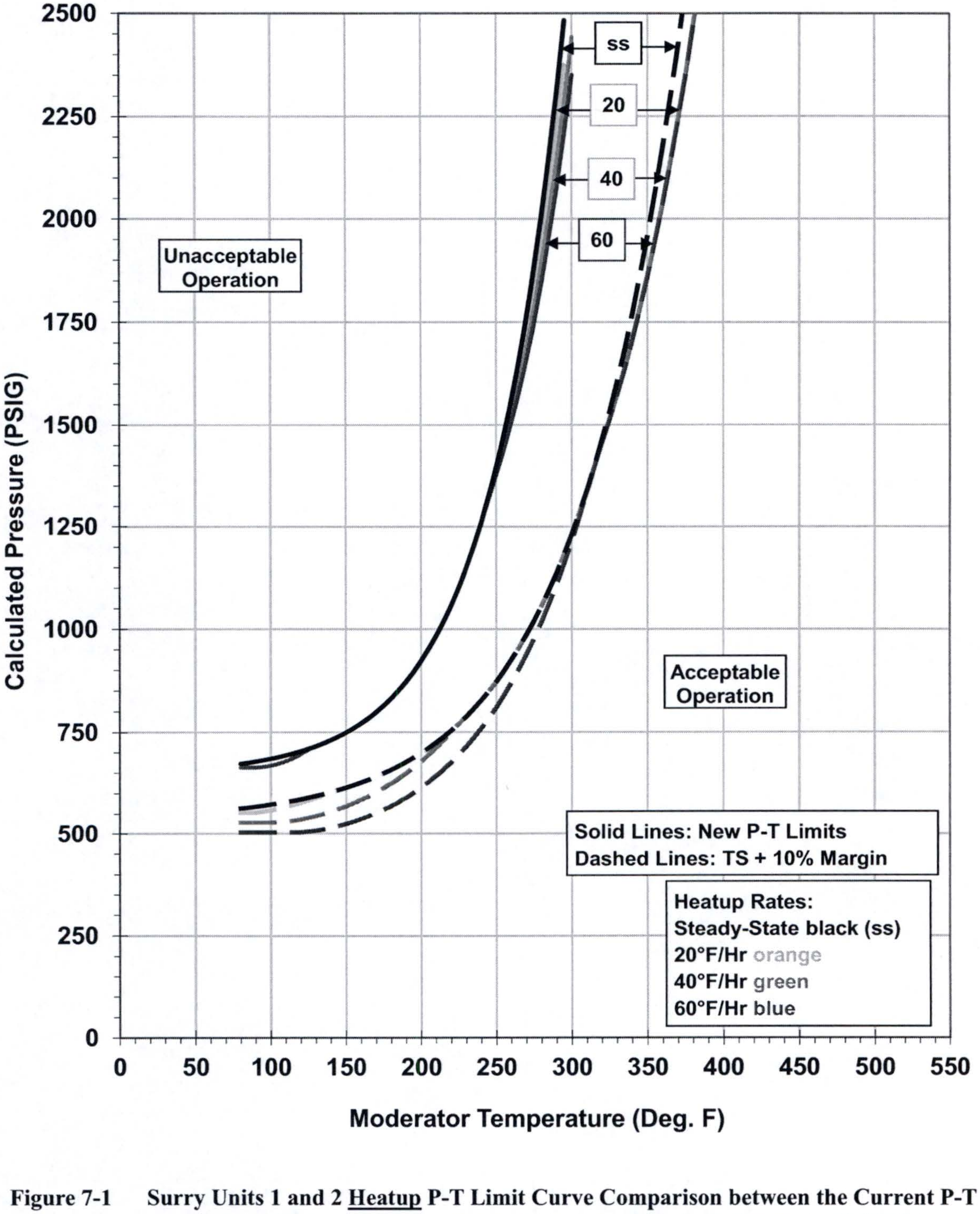


Figure 7-1 Surry Units 1 and 2 Heatup P-T Limit Curve Comparison between the Current P-T Limit Curves + 10% Margin and the New P-T Limit Curves to 68 EFPY

Note: "New P-T Limits" determined in Section 6. TS = Technical Specifications.

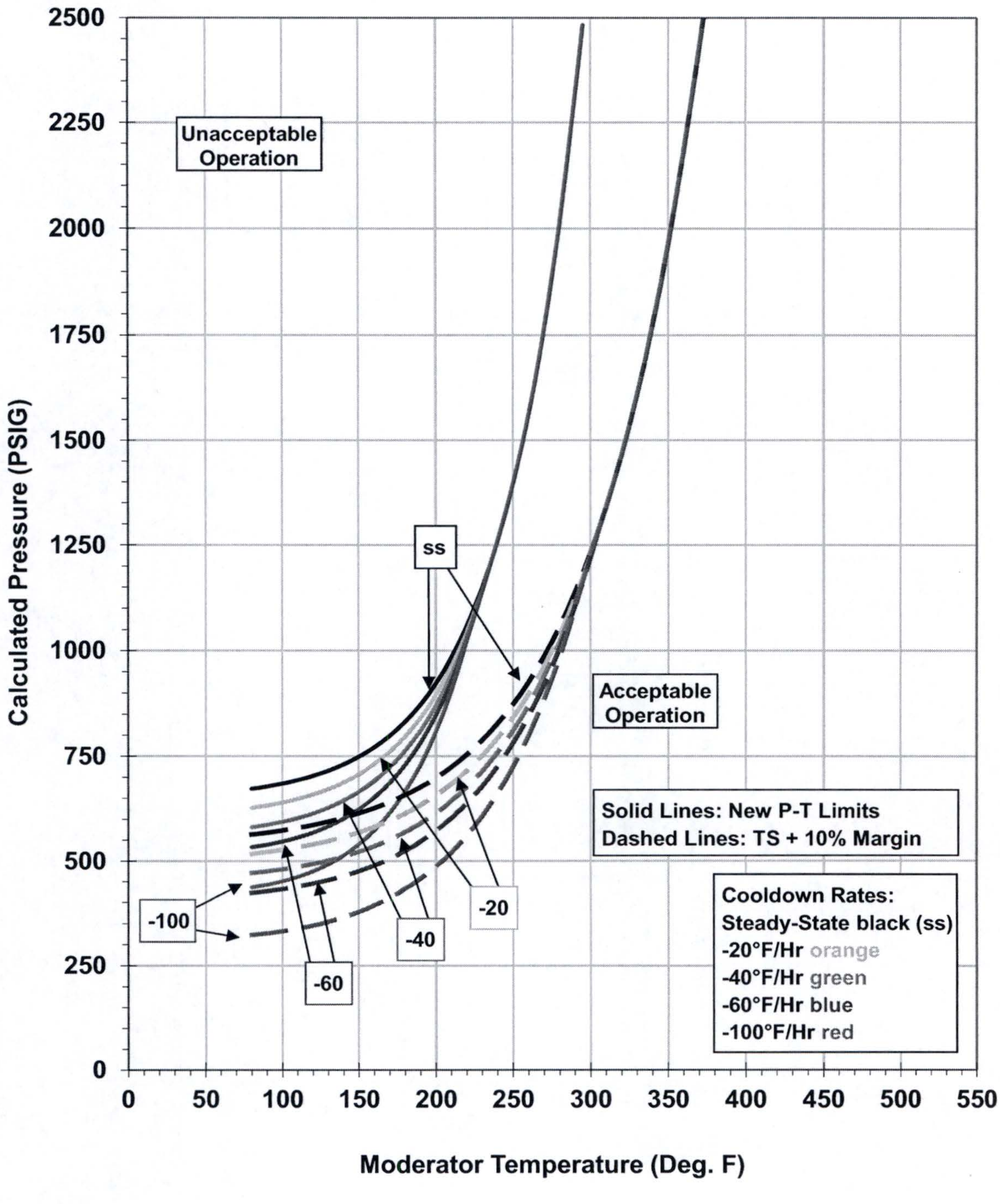


Figure 7-2 Surry Units 1 and 2 <u>Cooldown</u> P-T Limit Curve Comparison between the Current P-T Limit Curves + 10% Margin and the New P-T Limit Curves to 68 EFPY

Note: "New P-T Limits" determined in Section 6. TS = Technical Specifications.

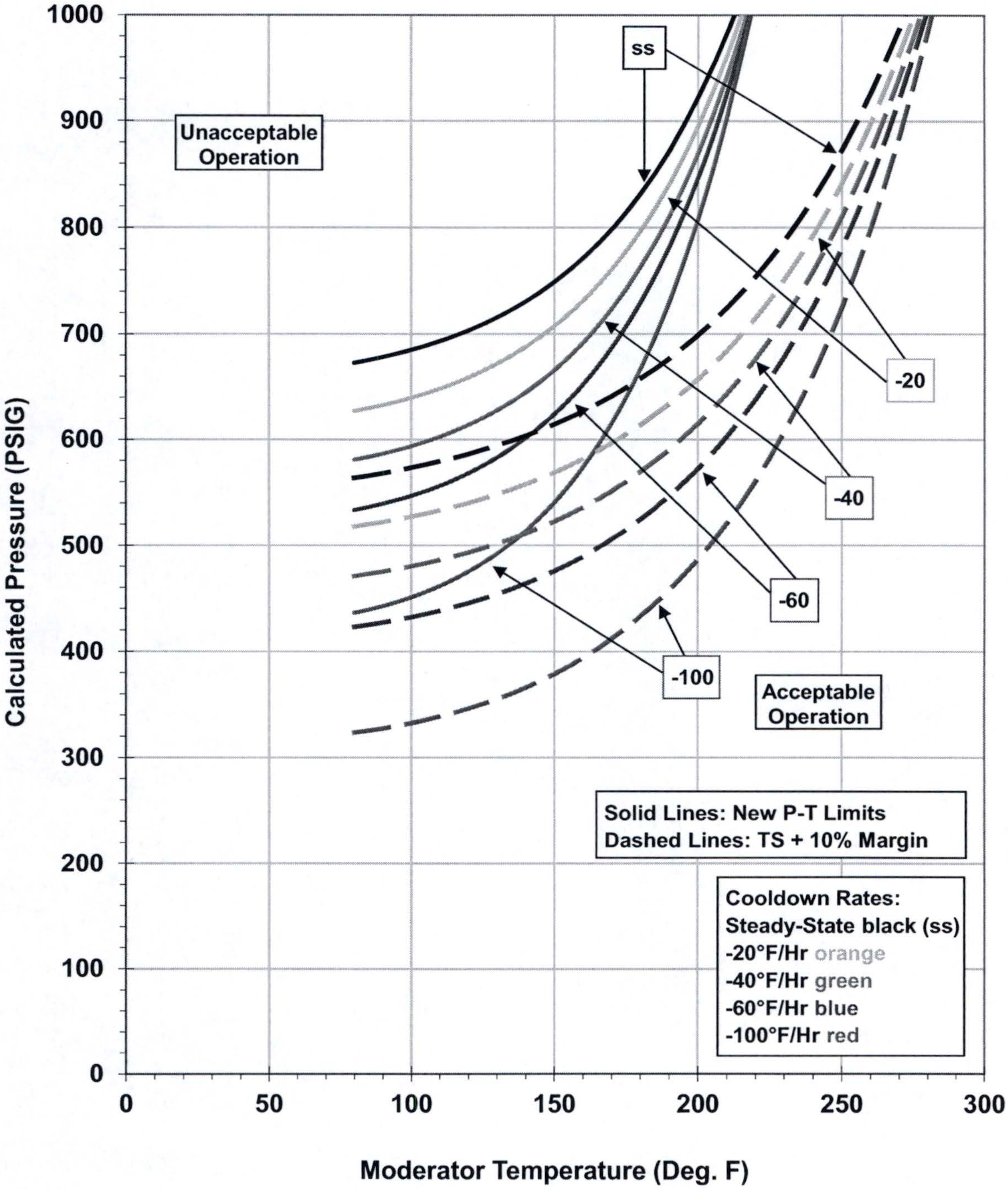


Figure 7-3 Surry Units 1 and 2 Cooldown P-T Limit Curve Comparison between the Current P-T Limit Curves + 10% Margin and the New P-T Limit Curves to 68 EFPY Magnified

Note: "New P-T Limits" determined in Section 6. TS = Technical Specifications.

Table 7-3 Data Points for Surry Units 1 and 2 Heatup P-T Limit Curve Comparison between the Current P-T Limit Curves + 10% Margin and the New P-T Limit Curves to 68 EFPY

T (°F)	0°F/hr			20°F/hr			40°F/hr			60°F/hr		
	+10% Current Curves (psig)	New P-T Curves (psig)	Margin(a) (psi)	+10% Current Curves (psig)	New P-T Curves (psig)	Margin(a) (psi)	+10% Current Curves (psig)	New P-T Curves (psig)	Margin(a) (psi)	+10% Current Curves (psig)	New P-T Curves (psig)	Margin(a) (psi)
80	564	673	109	553	673	120	528	673	145	503	664	161
85	566	675	109	553	675	122	528	675	148	503	664	161
90	568	678	110	553	678	125	528	678	151	503	664	161
95	571	681	111	556	681	125	528	681	154	503	665	162
100	573	685	112	559	685	126	528	685	157	503	668	165
105	576	689	113	563	689	126	528	689	161	503	673	170
110	579	693	114	567	693	126	530	693	164	503	679	176
115	583	698	115	572	698	126	532	698	166	503	687	184
120	586	703	117	578	703	125	536	703	168	503	696	193
125	590	709	119	584	709	125	540	709	169	505	707	203
130	594	716	121	590	716	126	544	716	171	507	716	209
135	599	723	124	597	723	126	550	723	173	510	723	213
140	604	730	127	604	730	127	555	730	175	514	730	217
145	609	739	130	609	739	130	562	739	177	518	739	221
150	614	749	134	614	749	134	569	749	180	524	749	225
155	620	759	139	620	759	139	576	759	183	529	759	230
160	627	771	144	627	771	144	585	771	187	536	771	235
165	634	784	150	634	784	150	594	784	191	543	784	241
170	641	798	157	641	798	157	603	798	195	551	798	247
175	649	814	165	649	814	165	613	814	201	560	814	255
180	658	832	174	658	832	174	625	832	207	569	832	263
185	667	851	184	667	851	184	637	851	215	579	851	272
190	677	873	196	677	873	196	650	873	223	590	873	283
195	688	896	209	688	896	209	663	896	233	602	896	294
200	699	922	223	699	922	223	678	922	244	615	922	308
205	711	951	240	711	951	240	694	951	257	628	951	323
210	725	983	258	725	983	258	712	983	271	643	983	340
215	739	1018	280	739	1018	280	730	1018	288	659	1018	359
220	754	1057	303	754	1057	303	750	1057	307	676	1057	381

Table 7-3 Data Points for Surry Units 1 and 2 Heatup P-T Limit Curve Comparison between the Current P-T Limit Curves + 10% Margin and the New P-T Limit Curves to 68 EFPY

T (°F)	0°F/hr			20°F/hr			40°F/hr			60°F/hr		
	+10% Current Curves (psig)	New P-T Curves (psig)	Margin[a] (psi)	+10% Current Curves (psig)	New P-T Curves (psig)	Margin[a] (psi)	+10% Current Curves (psig)	New P-T Curves (psig)	Margin[a] (psi)	+10% Current Curves (psig)	New P-T Curves (psig)	Margin[a] (psi)
225	771	1100	330	771	1100	330	771	1100	330	695	1100	405
230	788	1148	360	788	1148	360	788	1148	360	715	1148	433
235	807	1201	393	807	1201	393	807	1201	393	736	1201	464
240	828	1259	431	828	1259	431	828	1259	431	760	1259	499
245	850	1323	473	850	1319	469	850	1319	469	784	1319	534
250	873	1394	521	873	1383	510	873	1378	505	811	1378	566
255	899	1472	574	899	1455	557	899	1444	545	840	1438	598
260	926	1559	633	926	1534	608	926	1516	590	871	1504	633
265	955	1655	700	955	1622	667	955	1596	641	904	1577	673
270	987	1761	774	987	1718	731	987	1684	697	939	1658	718
275	1020	1878	858	1020	1825	804	1020	1782	761	978	1747	769
280	1057	2007	951	1057	1943	886	1057	1889	832	1018	1845	827
285	1096	2150	1055	1096	2073	977	1096	2008	912	1062	1954	891
290	1138	2308	1171	1138	2217	1079	1138	2139	1001	1110	2073	963
295	1183	2483	1300	1183	2375	1192	1183	2283	1100	1161	2205	1045
300	1231	2676	1445				1231	2443	1212	1215	2351	1136

Note:

(a) Margin equals *New P-T limit curve* data point minus the *current (Technical Specifications) P-T limit curves + 10%* data point for each temperature and rate.

Table 7-4 Data Points for Surry Units 1 and 2 Cooldown P-T Limit Curve Comparison between the Current P-T Limit Curves + 10% Margin and the New P-T Limit Curves to 68 EFPY

T (°F)	0°F/hr			-20°F/hr			-40°F/hr			-60°F/hr			-100°F/hr		
	+10% Current Curves (psig)	New P-T Curves (psig)	M[(a)] (psi)	+10% Current Curves (psig)	New P-T Curves (psig)	M[(a)] (psi)	+10% Current Curves (psig)	New P-T Curves (psig)	M[(a)] (psi)	+10% Current Curves (psig)	New P-T Curves (psig)	M[(a)] (psi)	+10% Current Curves (psig)	New P-T Curves (psig)	M[(a)] (psi)
80	564	673	109	518	627	109	471	581	110	423	534	111	324	436	113
85	566	675	109	520	630	110	473	584	111	425	536	111	325	439	114
90	568	678	110	522	633	111	475	587	111	427	540	112	328	443	115
95	571	681	111	525	636	111	478	590	112	430	543	113	330	446	117
100	573	685	112	527	640	112	480	594	113	432	547	115	333	451	118
105	576	689	113	530	644	114	483	598	115	435	551	116	335	456	120
110	579	693	114	533	648	115	486	603	116	438	556	118	339	462	123
115	583	698	115	537	653	117	490	608	118	442	562	120	342	468	126
120	586	703	117	540	659	118	493	614	120	445	568	123	346	475	129
125	590	709	119	544	665	121	497	620	123	450	575	126	350	483	133
130	594	716	121	549	672	123	502	628	126	454	583	129	355	492	137
135	599	723	124	553	679	126	506	636	129	459	592	133	360	502	142
140	604	730	127	558	688	130	511	645	133	464	601	137	366	514	148
145	609	739	130	563	697	134	517	655	138	470	612	143	372	527	155
150	614	749	134	569	707	138	523	666	143	476	624	148	379	541	162
155	620	759	139	575	719	144	529	678	149	482	637	155	386	557	171
160	627	771	144	582	731	150	536	692	156	490	652	163	394	574	180
165	634	784	150	589	745	156	544	707	163	497	669	171	403	594	191
170	641	798	157	597	761	164	552	724	172	506	687	181	412	616	204
175	649	814	165	605	778	173	560	742	182	515	707	192	422	640	218
180	658	832	174	614	797	183	570	763	193	525	730	205	433	667	234
185	667	851	184	624	818	194	580	786	206	536	755	219	445	697	252
190	677	873	196	634	841	207	591	811	220	547	782	235	458	730	272
195	688	896	209	645	867	222	603	839	236	560	813	253	472	767	295
200	699	922	223	657	895	238	615	870	255	573	847	274	487	808	321
205	711	951	240	670	927	257	629	905	276	587	885	297	504	854	350
210	725	983	258	684	962	277	644	943	299	603	926	323	521	905	383
215	739	1018	280	699	1000	301	660	985	325	620	973	353	541	961	420
220	754	1057	303	716	1043	327	677	1031	355	638	1024	386	561	1023	462

Table 7-4 Data Points for Surry Units 1 and 2 Cooldown P-T Limit Curve Comparison between the Current P-T Limit Curves + 10% Margin and the New P-T Limit Curves to 68 EFPY

T (°F)	0°F/hr			-20°F/hr			-40°F/hr			-60°F/hr			-100°F/hr		
	+10% Current Curves (psig)	New P-T Curves (psig)	M[a] (psi)	+10% Current Curves (psig)	New P-T Curves (psig)	M[a] (psi)	+10% Current Curves (psig)	New P-T Curves (psig)	M[a] (psi)	+10% Current Curves (psig)	New P-T Curves (psig)	M[a] (psi)	+10% Current Curves (psig)	New P-T Curves (psig)	M[a] (psi)
225	771	1100	330	733	1090	357	695	1083	388	658	1081	423	584	1081	497
230	788	1148	360	752	1142	390	715	1140	425	679	1140	461	608	1140	532
235	807	1201	393	772	1200	428	737	1200	463	702	1200	498	634	1200	565
240	828	1259	431	793	1259	465	760	1259	499	727	1259	532	662	1259	596
245	850	1323	473	817	1323	506	785	1323	538	753	1323	570	693	1323	630
250	873	1394	521	842	1394	552	811	1394	583	782	1394	612	726	1394	668
255	899	1472	574	869	1472	603	840	1472	632	813	1472	660	761	1472	711
260	926	1559	633	898	1559	661	871	1559	688	846	1559	714	800	1559	760
265	955	1655	700	929	1655	726	905	1655	750	882	1655	774	841	1655	814
270	987	1761	774	963	1761	798	941	1761	820	920	1761	841	885	1761	876
275	1020	1878	858	999	1878	879	979	1878	899	961	1878	917	933	1878	945
280	1057	2007	951	1038	2007	970	1021	2007	987	1006	2007	1002	985	2007	1023
285	1096	2150	1055	1080	2150	1071	1065	2150	1085	1054	2150	1097	1040	2150	1110
290	1138	2308	1171	1124	2308	1184	1113	2308	1195	1106	2308	1203	1100	2308	1209
295	1183	2483	1300	1173	2483	1311	1165	2483	1318	1161	2483	1322	1164	2483	1319
300	1231	2676	1445												

Note:

(a) Margin equals *New P-T limit curve* data point minus the *current (Technical Specifications) P-T limit curves + 10%* data point for each temperature and rate.

Table 7-5 Surry Units 1 and 2 Heatup P-T Limit Curve Margin Summary between the Current P-T Limit Curves + 10% Margin and the New P-T Limit Curves to 68 EFPY

T (°F)	0°F/hr Margin (psi)	20°F/hr Margin (psi)	40°F/hr Margin (psi)	60°F/hr Margin (psi)
80	109	120	145	161
85	109	122	148	161
90	110	125	151	161
95	111	125	154	162
100	112	126	157	165
105	113	126	161	170
110	114	126	164	176
115	115	126	166	184
120	117	125	168	193
125	119	125	169	203
130	121	126	171	209
135	124	126	173	213
140	127	127	175	217
145	130	130	177	221
150	134	134	180	225
155	139	139	183	230
160	144	144	187	235
165	150	150	191	241
170	157	157	195	247
175	165	165	201	255
180	174	174	207	263
185	184	184	215	272
190	196	196	223	283
195	209	209	233	294
200	223	223	244	308
205	240	240	257	323
210	258	258	271	340
215	280	280	288	359
220	303	303	307	381
225	330	330	330	405
230	360	360	360	433
235	393	393	393	464
240	431	431	431	499
245	473	469	469	534
250	521	510	505	566
255	574	557	545	598
260	633	608	590	633
265	700	667	641	673
270	774	731	697	718
275	858	804	761	769
280	951	886	832	827
285	1055	977	912	891
290	1171	1079	1001	963
295	1300	1192	1100	1045
300	1445	-	1212	1136

Table 7-6 Surry Units 1 and 2 Cooldown P-T Limit Curve Margin Summary between the Current P-T Limit Curves + 10% Margin and the New P-T Limit Curves to 68 EFPY

T (°F)	0°F/hr Margin (psi)	-20°F/hr Margin (psi)	-40°F/hr Margin (psi)	-60°F/hr Margin (psi)	-100°F/hr Margin (psi)
80	109	109	110	111	113
85	109	110	111	111	114
90	110	111	111	112	115
95	111	111	112	113	117
100	112	112	113	115	118
105	113	114	115	116	120
110	114	115	116	118	123
115	115	117	118	120	126
120	117	118	120	123	129
125	119	121	123	126	133
130	121	123	126	129	137
135	124	126	129	133	142
140	127	130	133	137	148
145	130	134	138	143	155
150	134	138	143	148	162
155	139	144	149	155	171
160	144	150	156	163	180
165	150	156	163	171	191
170	157	164	172	181	204
175	165	173	182	192	218
180	174	183	193	205	234
185	184	194	206	219	252
190	196	207	220	235	272
195	209	222	236	253	295
200	223	238	255	274	321
205	240	257	276	297	350
210	258	277	299	323	383
215	280	301	325	353	420
220	303	327	355	386	462
225	330	357	388	423	497
230	360	390	425	461	532
235	393	428	463	498	565
240	431	465	499	532	596
245	473	506	538	570	630
250	521	552	583	612	668
255	574	603	632	660	711
260	633	661	688	714	760
265	700	726	750	774	814
270	774	798	820	841	876
275	858	879	899	917	945
280	951	970	987	1002	1023
285	1055	1071	1085	1097	1110
290	1171	1184	1195	1203	1209
295	1300	1311	1318	1322	1319
300	1445	-	-	-	-

8 REFERENCES

1. Surry Power Station Technical Specifications, Section 3.1.B, Amendments Nos. 285 and 285.
2. U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.
3. Westinghouse Report WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," May 2004.
4. Appendix G to the 1998 Edition through 2000 Addenda of ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Division 1, "Fracture Toughness Criteria for Protection Against Failure."
5. Code of Federal Regulations, 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Federal Register, Volume 60, No. 243, dated December 19, 1995.
6. U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001.
7. NUREG-2192, "Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, July 2017. *[Agencywide Documents Access and Management System (ADAMS) Accession Number ML17188A158]*
8. NRC Regulatory Issue Summary (RIS) 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components," U.S. Nuclear Regulatory Commission, October 2014. *[ADAMS Accession Number ML14149A165]*
9. Virginia Electric and Power Company Letter to the U.S. Nuclear Regulatory Commission (Serial No. 15-023), "Virginia Electric and Power Company Surry Power Station Units 1 and 2 Proposed License Amendment Request Clarification of Reactor Coolant System Heatup and Cooldown Limitations Technical Specification Figures Response to Additional Information," dated February 4, 2015. *[ADAMS Accession Number ML15041A720]*
10. Virginia Electric and Power Company Letter to the U.S. Nuclear Regulatory Commission (Serial No. 15-023B), "Virginia Electric and Power Company Surry Power Stations Units 1 and 2 Final Update Regarding Fluence Assessment for Reactor Vessel Inlet and Outlet Nozzles," dated October 26, 2015. *[ADAMS Accession Number ML15302A340]*
11. RSICC Computer Code Collection CCC-650, "DOORS 3.2: One, Two- and Three Dimensional Discrete Ordinates Neutron/Photon Transport Code System," April 1998.
12. RSICC Data Library Collection DLC-185, "BUGLE-96: Coupled 47 Neutron, 20 Gamma-Ray Group Cross-Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications," March 1996.
13. AREVA NP, Inc. Report BAW-2313, Revision 7, Supplement 1, Revision 1, "Supplement to B&W Fabricated Reactor Vessel Materials and Surveillance Data Information for Surry Unit 1 and Unit 2," AREVA Document No. 77-2313S-007-001, February 2017.
14. Pressurized Water Reactor Owners Group (PWROG) Report PWROG-16045-NP, Revision 0, "Determination of Unirradiated RT_{NDT} and Upper-Shelf Energy Values of the Surry Units 1 and 2 Reactor Vessel Materials," March 2017.
15. Westinghouse Report WCAP-18028-NP, Revision 0, "Extended Beltline Pressure Vessel Fluence Evaluations Applicable to Surry Units 1 & 2," September 2015.

16. NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Chapter 5 LWR Edition, Branch Technical Position 5-3, "Fracture Toughness Requirements," Revision 2, U.S. Nuclear Regulatory Commission, March 2007.

17. ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Division 1, Subarticle NB-2300, "Fracture Toughness Requirements for Material."

18. *BWRVIP-173-A: BWR Vessel and Internals Project: Evaluation of Chemistry Data for BWR Vessel Nozzle Forging Materials.* EPRI, Palo Alto, CA: 2011. 1022835.

19. Surry Power Station Updated Final Safety Analysis Report, Revision 48, September 2016.

20. Framatome ANP Report BAW-2494, Revision 1, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessel of Surry Units 1 and 2 for Extended Life through 48 Effective Full Power Years," September 2005.

21. Framatome ANP Report BAW-2324, Revision 0, "Analysis of Capsule X, Virginia Power Surry Unit No. 1, Reactor Vessel Material Surveillance Program," April 1998.

22. Westinghouse Report WCAP-16001, Revision 0, "Analysis of Capsule Y from Dominion Surry Unit 2 Reactor Vessel Radiation Surveillance Program," February 2003.

23. Westinghouse Report WCAP-14177, Revision 0, "Surry Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," October 1994.

24. PWROG Report ANP-2650, "Updated Results for Request for Additional Information Regarding Reactor Pressure Vessel Integrity," July 2007.

25. AREVA NP, Inc. Report AREVA-17-01417, "Transmittal of Information for Surry Specific Weld Wire Heat 299L44 (Capsule A5) from BAW-2313 Revision 7, PA-MSC-1200R0 Task 1," May 2017.

26. NRC Safety Evaluation, "Final Safety Evaluation for Topical Report BAW-2308, Revision 1 'Initial RT_{NDT} of Linde 80 Weld Materials'," August 2005. *[ADAMS Accession Number ML052070408]*

27. NRC Safety Evaluation, "Final Safety Evaluation for Pressurized Water Reactor Owners Group (PWROG) Topical Report (TR) BAW-2308, Revision 2, 'Initial RT_{NDT} of Linde 80 Weld Materials'," March 2008. *[ADAMS Accession Number ML080770349]*

28. U.S. NRC Technical Letter Report TLR-RES/DE/CIB-2013-01, "Evaluation of the Beltline Region for Nuclear Reactor Pressure Vessels," Office of Nuclear Regulatory Research [RES], November 2014. *[ADAMS Accession Number ML14318A177]*

29. Virginia Power Calculation SM-792, Revision 3, "Surry 1 & 2 Composite P/T Limits Curve," January 1996.

30. Virginia Power Calculation SM-945, Revision 0, "Surry Unit 1 and 2 Heatup/Cooldown Curves and LTOPS Setpoint," February 1995.

APPENDIX A THERMAL STRESS INTENSITY FACTORS (K_{It})

Tables A-1 and A-2 contain the thermal stress intensity factors (K_{It}) for the maximum heatup and cooldown rates at 68 EFPY for Surry Units 1 and 2 based on the Section 6 P-T limit curves. The reactor vessel cylindrical shell radii to the 1/4T and 3/4T locations are as follows:

- 1/4T Radius = 80.5 inches
- 3/4T Radius = 84.5 inches

Table A-1 K_{It} Values for Surry Units 1 and 2 at 68 EFPY 100°F/hr Heatup Curves (w/ Flange Requirements, and w/o Margins for Instrument Errors)

Water Temp. (°F)	Vessel Temperature at 1/4T Location for 60°F/hr Heatup (°F)	1/4T Thermal Stress Intensity Factor (ksi √in.)	Vessel Temperature at 3/4T Location for 60°F/hr Heatup (°F)	3/4T Thermal Stress Intensity Factor (ksi √in.)
60	56.538	-1.077	55.169	0.604
65	59.879	-2.410	55.956	1.618
70	63.411	-3.347	57.605	2.400
75	67.274	-4.168	59.939	3.035
80	71.373	-4.775	62.802	3.533
85	75.623	-5.293	66.103	3.936
90	80.052	-5.686	69.739	4.256
95	84.560	-6.022	73.647	4.515
100	89.192	-6.279	77.760	4.725
105	93.865	-6.504	82.039	4.898
110	98.619	-6.677	86.448	5.039
115	103.396	-6.832	90.960	5.157
120	108.226	-6.952	95.554	5.256
125	113.068	-7.062	100.213	5.339
130	117.946	-7.150	104.924	5.410
135	122.830	-7.232	109.675	5.472
140	127.738	-7.298	114.459	5.526
145	132.648	-7.363	119.269	5.574
150	137.574	-7.416	124.100	5.617
155	142.502	-7.470	128.947	5.657
160	147.439	-7.515	133.806	5.693
165	152.378	-7.561	138.677	5.727
170	157.322	-7.601	143.555	5.759
175	162.268	-7.642	148.440	5.790
180	167.217	-7.679	153.330	5.819
185	172.167	-7.718	158.225	5.847
190	177.119	-7.752	163.122	5.875
195	182.072	-7.789	168.022	5.902
200	187.025	-7.823	172.925	5.928
205	191.980	-7.858	177.828	5.954
210	196.934	-7.891	182.734	5.980

Table A-2 K_{It} Values for Surry Units 1 and 2 at 68 EFPY 100°F/hr Cooldown Curves (w/ Flange Requirements, and w/o Margins for Instrument Errors)

Water Temp. (°F)	Vessel Temperature at 1/4T Location for 100°F/hr Cooldown (°F)	100°F/hr Cooldown 1/4T Thermal Stress Intensity Factor (ksi √in.)
210	233.489	14.320
205	228.412	14.261
200	223.336	14.202
195	218.259	14.143
190	213.182	14.085
185	208.104	14.025
180	203.027	13.967
175	197.950	13.907
170	192.873	13.849
165	187.796	13.790
160	182.720	13.731
155	177.643	13.673
150	172.566	13.614
145	167.489	13.556
140	162.413	13.497
135	157.336	13.439
130	152.260	13.381
125	147.183	13.323
120	142.107	13.265
115	137.031	13.207
110	131.955	13.149
105	126.879	13.092
100	121.803	13.034
95	116.728	12.977
90	111.652	12.920
85	106.577	12.863
80	101.502	12.806
75	96.427	12.749
70	91.352	12.692
65	86.277	12.636
60	81.203	12.579

APPENDIX B REACTOR VESSEL INLET AND OUTLET NOZZLES

As described in NRC Regulatory Issue Summary (RIS) 2014-11 [Ref. B-1], reactor vessel non-beltline materials may define pressure-temperature (P-T) limit curves that are more limiting than those calculated for the reactor vessel cylindrical shell beltline materials. Reactor vessel nozzles, penetrations, and other discontinuities have complex geometries that can exhibit significantly higher stresses than those for the reactor vessel beltline shell region. These higher stresses can potentially result in more restrictive P-T limits, even if the reference temperatures (RT_{NDT}) for these components are not as high as those of the reactor vessel beltline shell materials that have simpler geometries.

The methodology contained in WCAP-14040-A, Revision 4 [Ref. B-2] was used in the main body of this report to develop P-T limit curves for the limiting Surry Units 1 and 2 cylindrical shell beltline material; however, WCAP-14040-A, Revision 4 does not consider ferritic materials in the area adjacent to the beltline, specifically the stressed inlet and outlet nozzles. Due to the geometric discontinuity, the inside corner regions of these nozzles are the most highly stressed ferritic component outside the beltline region of the reactor vessel; therefore, these components are analyzed in this Appendix. P-T limit curves are determined for the reactor vessel nozzle corner region for Surry Units 1 and 2 and compared to the P-T limit curves for the reactor vessel traditional beltline region in order to determine if the nozzles can be more limiting than the reactor vessel beltline as the plant ages and the vessel accumulates more neutron fluence. The increase in neutron fluence as the plant ages causes a concern for embrittlement of the reactor vessel above the beltline region. Therefore, the P-T limit curves are developed for the nozzle inside corner region since the geometric discontinuity results in high stresses due to internal pressure and the cooldown transient. The cooldown transient is analyzed as it results in tensile stresses at the inside surface of the nozzle corner.

A 1/4T axial flaw is postulated at the inside surface of the reactor vessel nozzle corner, and stress intensity factors are determined based on the rounded curvature of the nozzle geometry. The allowable pressure is then calculated based on the fracture toughness of the nozzle material and the stress intensity factors for the 1/4T flaw.

B.1 CALCULATION OF ADJUSTED REFERENCE TEMPERATURES

The fracture toughness (K_{Ic}) used for the inlet and outlet nozzle material is defined in Appendix G of the Section XI ASME Code, as discussed in Section 4 of this report. The K_{Ic} fracture toughness curve is dependent on the Adjusted Reference Temperature (ART) value for irradiated materials. The ART values for the inlet and outlet nozzle materials are determined using the methodology contained in Regulatory Guide 1.99, Revision 2 [Ref. B-3], which is described in Section 5 of this report, as well as weight percent (wt. %) copper (Cu) and nickel (Ni) values, initial RT_{NDT} values, and projected neutron fluence as inputs. The material properties for each of the reactor vessel inlet and outlet nozzle forging materials are documented in Tables B-1 and B-2 and a summary of the limiting inlet and outlet nozzle ART values for Surry Units 1 and 2 is presented in Table B-3.

Nozzle Material Properties

The Surry Units 1 and 2 nozzle material properties are provided in Tables B-1 and B-2. Copper (Cu) and Nickel (Ni) weight percent (wt. %) values were obtained from PWROG-16045-NP [Ref. B-4] for each of the Surry Units 1 and 2 reactor vessel inlet and outlet nozzles.

Surry Units 1 and 2 nozzle forging initial RT_{NDT} and initial USE values for the inlet and outlet nozzles were also taken from PWROG-16045-NP [Ref. B-4]. The Charpy V-Notch forging specimen orientation for the inlet and outlet nozzles was not reported in the Surry Units 1 and 2 Certified Material Test Reports (CMTRs); thus, it was conservatively assumed that the orientation was the "strong direction" for each nozzle forging. Since each of the nozzle forging materials lacked drop-weight test data, the initial RT_{NDT} values were determined for each of the Surry Units 1 and 2 reactor vessel inlet and outlet nozzle forging materials using the BWRVIP-173-A, Appendix B, Alternative Approach 2 Methodology [Ref. B-5]. The initial RT_{NDT} values for all of the nozzle materials were determined using CVGRAPH, Version 6.02 hyperbolic tangent curve fits through the Charpy data points, in accordance with BWRVIP-173-A, Appendix B, Alternative Approach 2 Methodology [Ref. B-5]. The initial USE values were determined in accordance with the methodology described in ASTM E185-82 [Ref. B-6]. For each of the nozzle forging materials, use of BTP 5-3 Paragraph 1.2 [Ref. B-7] was necessary. The Surry Units 1 and 2 initial RT_{NDT} and initial USE values for the inlet and outlet nozzles materials are summarized in Tables B-1 and B-2.

Nozzle Calculated Neutron Fluence Values

The maximum fast neutron (E > 1 MeV) exposure of the Surry Units 1 and 2 reactor vessel materials is discussed in Section 2 of this report. The fluence values used in the inlet and outlet nozzle ART calculations were calculated at a location corresponding to the postulated 1/4T flaw in nozzle forgings and were chosen at an elevation lower than the actual elevation of the postulated flaw and at the inside surface of the nozzle, for conservatism.

Per NRC RIS 2014-11 [Ref. B-1], embrittlement of reactor vessel materials, with projected fluence values less than 1×10^{17} n/cm^2 (E > 1.0 MeV), does not need to be considered. Per Tables 2-3 and 2-4, the only Surry Units 1 and 2 inlet and outlet nozzles determined to receive a projected maximum fluence of greater than 1×10^{17} n/cm^2 (E > 1 MeV) at the 1/4T flaw location at 68 EFPY are Surry Unit 1 Inlet Nozzle 1, Surry Unit 2 Inlet Nozzle 1, and Surry Unit 2 Outlet Nozzle 3. For conservatism, the ΔRT_{NDT} for each of the nozzle materials is calculated.

The second conclusion of TLR-RES/DE/CIB-2013-01 [Ref. B-8] states that if ΔRT_{NDT} is calculated to be less than 25°F, then embrittlement need not be considered. This conclusion is applicable to and is applied to each of the Surry Units 1 and 2 inlet and outlet nozzle forging materials. Therefore, the initial RT_{NDT} values documented in Tables B-1 and B-2 are identical to the nozzle ART values.

The neutron fluence values used in the ART calculations for the Surry Units 1 and 2 inlet and outlet nozzle forging materials are summarized in Tables B-1 and B-2. The limiting nozzle ART values used for determination of the nozzle P-T limit curves are summarized in Table B-3.

Table B-1 Calculation of the Surry Unit 1 Nozzle Forging ART Values at 68 EFPY

RPV Material	R.G. 1.99, Rev. 2 Position	Wt. % Cu[(a)]	Wt. % Ni[(a)]	CF[(a)] (°F)	Surface Fluence[(b)] (x 10^{19} n/cm^2, E > 1.0 MeV)	Surface FF[(b)]	$RT_{NDT(U)}$[(c)] (°F)	ΔRT_{NDT}[(d)] (°F)	σ_I[(c)] (°F)	σ_Δ[(e)] (°F)	Margin (°F)	ART (°F)
Inlet Nozzle 1 (Heat # 9-4787)	1.1	0.159	0.85	123.5	0.0124	0.127	10.3	0.0 (15.6)	0.0	0.0	0.0	10.3
Inlet Nozzle 2 (Heat # 9-5078)	1.1	0.159	0.87	123.7	0.00322	0.048	11.6	0.0 (5.9)	0.0	0.0	0.0	11.6
Inlet Nozzle 3 (Heat # 9-4819)	1.1	0.159	0.84	123.4	0.00446	0.062	-47.2	0.0 (7.6)	0.0	0.0	0.0	-47.2
Outlet Nozzle 1 (Heat # 9-4825-1)	1.1	0.159	0.85	123.5	0.00345	0.051	-44.9	0.0 (6.3)	0.0	0.0	0.0	-44.9
Outlet Nozzle 2 (Heat # 9-4762)	1.1	0.159	0.83	123.3	0.00249	0.039	-87.5	0.0 (4.8)	0.0	0.0	0.0	-87.5
Outlet Nozzle 3 (Heat # 9-4788)	1.1	0.159	0.84	123.4	0.00962	0.107	-50.2	0.0 (13.2)	0.0	0.0	0.0	-50.2

Notes:

(a) Chemical composition data taken from Tables 3-1 and 3-2 of this report. Chemistry factor values taken from Table 3-10 of this report.

(b) Surface fluence values taken from Section 2 of this report. FF = fluence factor = $f^{(0.28-0.10*\log(f))}$.

(c) Initial RT_{NDT} values and σ_I values are from Table 3-2 of this report.

(d) Calculated ΔRT_{NDT} values less than 25°F have been reduced to zero per TLR-RES/DE/CIB-2013-01 [Ref. B-8]; actual calculated ΔRT_{NDT} values are listed in parentheses for these materials.

(e) Per the guidance of Regulatory Guide 1.99, Revision 2 [Ref. B-3], the base metal σ_Δ = 17°F for Position 1.1. However, σ_Δ need not exceed 0.5*ΔRT_{NDT}.

Table B-2 Calculation of the Surry Unit 2 Nozzle Forging ART Values at 68 EFPY

RPV Material	R.G. 1.99, Rev. 2 Position	Wt. % Cu(a)	Wt. % Ni(a)	CF(a) (°F)	Surface Fluence(b) (x 10^{19} n/cm², E > 1.0 MeV)	Surface FF(b)	$RT_{NDT(U)}$(c) (°F)	ΔRT_{NDT}(d) (°F)	σ_I(c) (°F)	σ_Δ(e) (°F)	Margin (°F)	ART(f) (°F)
Inlet Nozzle 1 (Heat # 9-5104)	1.1	0.159	0.84	123.4	0.0139	0.137	-29.7	0.0 (16.8)	0.0	0.0	0.0	-29.7
Inlet Nozzle 2 (Heat # 9-4815)	1.1	0.159	0.87	123.7	0.00321	0.048	4.5	0.0 (5.9)	0.0	0.0	0.0	4.5
Inlet Nozzle 3 (Heat # 9-5205)	1.1	0.159	0.86	123.6	0.00437	0.061	6.5	0.0 (7.5)	0.0	0.0	0.0	6.5
Outlet Nozzle 1 (Heat # 9-4825-2)	1.1	0.159	0.85	123.5	0.00338	0.050	-58.1	0.0 (6.2)	0.0	0.0	0.0	-58.1
Outlet Nozzle 2 (Heat # 9-5086-1)	1.1	0.159	0.86	123.6	0.00248	0.039	-26.6	0.0 (4.8)	0.0	0.0	0.0	-26.6
Outlet Nozzle 3 (Heat # 9-5086-2)	1.1	0.159	0.87	123.7	0.0107	0.115	-33.8	0.0 (14.2)	0.0	0.0	0.0	-33.8

Notes:

(a) Chemical composition values taken from Tables 3-3 and 3-4 of this report. Chemistry Factor values taken from Table 3-12 of this report.

(b) Surface fluence values taken from Section 2 of this report. FF = fluence factor = $f^{(0.28-0.10*\log(f))}$.

(c) Initial RT_{NDT} values and σ_I values are from Table 3-4 of this report.

(d) Calculated ΔRT_{NDT} values less than 25°F have been reduced to zero per TLR-RES/DE/CIB-2013-01 [Ref. B-8]; actual calculated ΔRT_{NDT} values are listed in parentheses for these materials.

(e) Per the guidance of Regulatory Guide 1.99, Revision 2 [Ref. B-3], the base metal σ_Δ = 17°F for Position 1.1. However, σ_Δ need not exceed 0.5*ΔRT_{NDT}.

Table B-3 Summary of the Limiting ART Values for the Surry Units 1 and 2 Inlet and Outlet Nozzle Forging Materials

EFPY	Nozzle Material and ID Number	Limiting ART Value (°F)
68	Surry Unit 1 Inlet Nozzle 2 (Heat # 9-5078)	11.6
	Surry Unit 1 Outlet Nozzle 1 (Heat # 9-4825-1)	-44.9
	Surry Unit 2 Inlet Nozzle 3 (Heat # 9-5205)	6.5
	Surry Unit 2 Outlet Nozzle 2 (Heat # 9-5086-1)	-26.6

B.2 NOZZLE COOLDOWN PRESSURE-TEMPERATURE LIMITS

Allowable pressures are determined for a given temperature based on the fracture toughness of the limiting nozzle material along with the appropriate pressure and thermal stress intensity factors. The Surry Units 1 and 2 nozzle fracture toughness used to determine the P-T limits is calculated using the limiting inlet and outlet nozzle ART values from Table B-3. The stress intensity factor correlations used for the nozzle corners are provided in Oak Ridge National Laboratory study, ORNL/TM-2010/246 [Ref. B-9], and are consistent with ASME PVP2011-57015 [Ref. B-10]. The methodology includes postulating an inside surface 1/4T nozzle corner flaw, and calculating through-wall nozzle corner stresses for a cooldown rate of 100°F/hour.

The through-wall stresses at the nozzle corner location were fitted based on a third-order polynomial of the form:

$$\sigma = A_0 + A_1x + A_2x^2 + A_3x^3$$

where,

σ = through-wall stress distribution

x = through-wall distance from inside surface

A_0, A_1, A_2, A_3 = coefficients of polynomial fit for the third-order polynomial, used in the stress intensity factor expression discussed below

The stress intensity factors generated for a rounded nozzle corner for the pressure and thermal gradient were calculated based on the methodology provided in ORNL/TM-2010/246. The stress intensity factor expression for a rounded corner is:

$$K_I = \sqrt{\pi a}\left[0.706A_0 + 0.537\left(\frac{2a}{\pi}\right)A_1 + 0.448\left(\frac{a^2}{2}\right)A_2 + 0.393\left(\frac{4a^3}{3\pi}\right)A_3\right]$$

where,

K_I = stress intensity factor for a circular corner crack on a nozzle with a rounded inner radius corner

a = crack depth at the nozzle corner, for use with 1/4T (25% of the wall thickness)

The reactor vessel nozzle P-T limit curves for Surry Units 1 and 2 are shown in Figures B-1 and B-2, respectively, based on the stress intensity factor expression discussed above. Also shown in these figures are the current Surry Power Station Technical Specification (TS) beltline cooldown P-T limit curves [Ref. B-11] (without pressure adjustment + 10% margin) (represented with the dashed lines) and the beltline cooldown P-T limit curves developed in this report (represented with the solid lines). These beltline cooldown P-T limit curves are located in Figure 7-2. The current Surry Power Station curves are included in Figures B-1 and B-2 for informational purposes.

Note that the figures show the most limiting P-T limit curves of the inlet and outlet nozzle for each Unit. The nozzle P-T limits are provided for a cooldown rate of 100 °F/hr, along with a steady-state curve.

An outside surface flaw in the nozzle was not considered because the pressure stress is significantly lower at the outside surface than the inside surface. A heatup nozzle P-T limit curve is also not provided since it would be less limiting than the cooldown nozzle P-T limit curves shown in Figures B-1 and B-2 for an inside surface flaw. Additionally, the cooldown transient is more limiting than the heatup transient since it results in tensile stresses at the inside surface of the nozzle corner.

Conclusion

Based on the results shown in Figures B-1 and B-2, it is concluded that the nozzle P-T limits are bounded by the traditional cylindrical beltline curves. The minimum pressure difference between the newly developed beltline P-T limit curves and nozzle P-T limit curves is 459 psi for Surry Unit 1 and 545 psi for Surry Unit 2 (based on steady state conditions at 80°F). The minimum pressure difference between the current Surry Power Station Technical Specifications beltline P-T limit curves (plus 10% margin) and nozzle P-T limit curves is 568 psi for Surry Unit 1 and 654 psi for Surry Unit 2 (based on steady state conditions at 80°F). Therefore, the P-T limits provided in Section 6 and in the current Surry Power Station Technical Specifications [Ref. B-11] remain limiting for the beltline and non-beltline reactor vessel components.

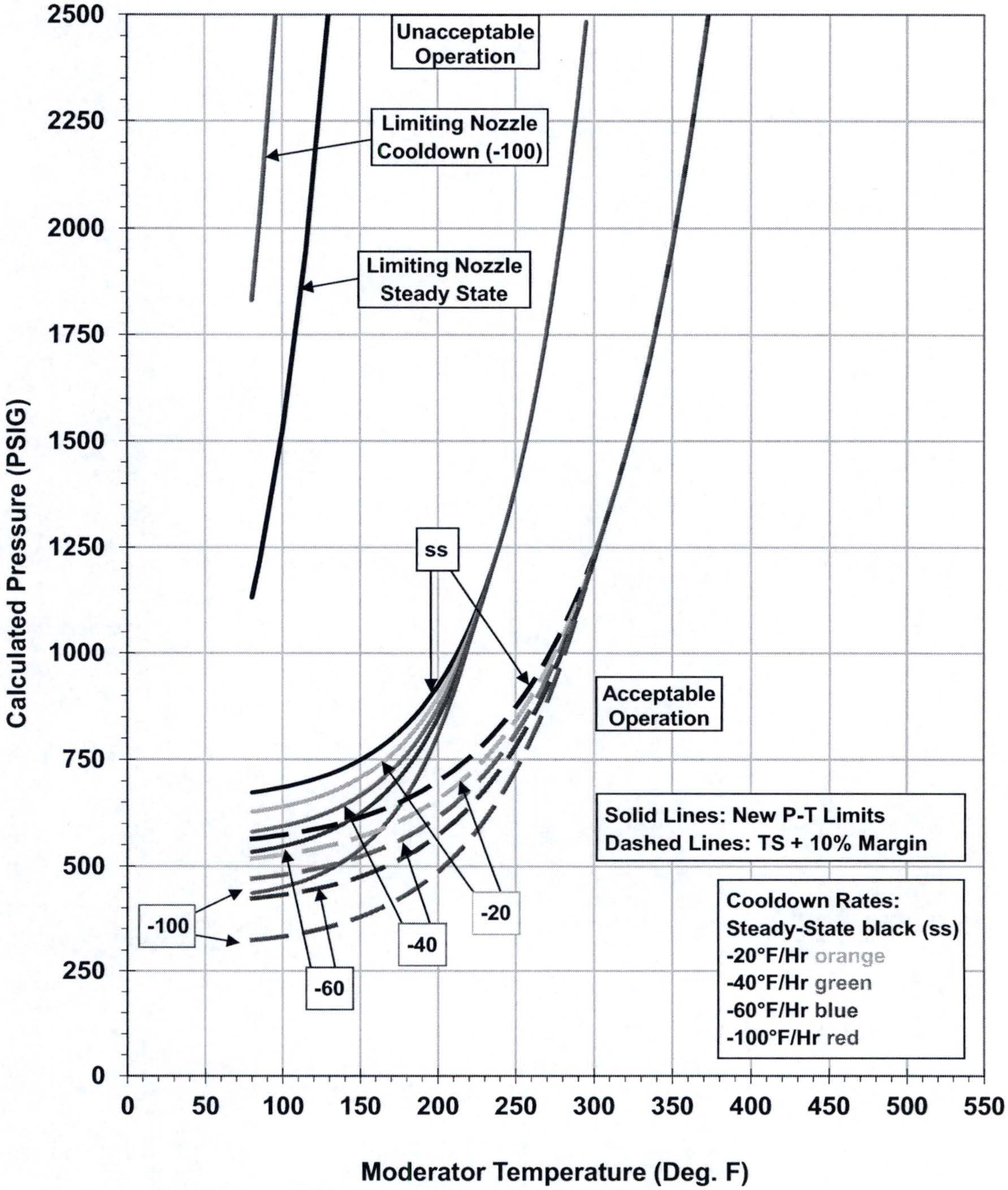


Figure B-1 **Comparison of Surry Unit 1 Beltline Cooldown P-T Limits (Including Current P-T Limits without Pressure Adjustment + 10% Margin and New 68 EFPY P-T Limits) to 68 EFPY Nozzle P-T Limits, Without Margins for Instrumentation Errors**

Note: "New P-T Limits" determined in Section 6. TS = Technical Specifications.

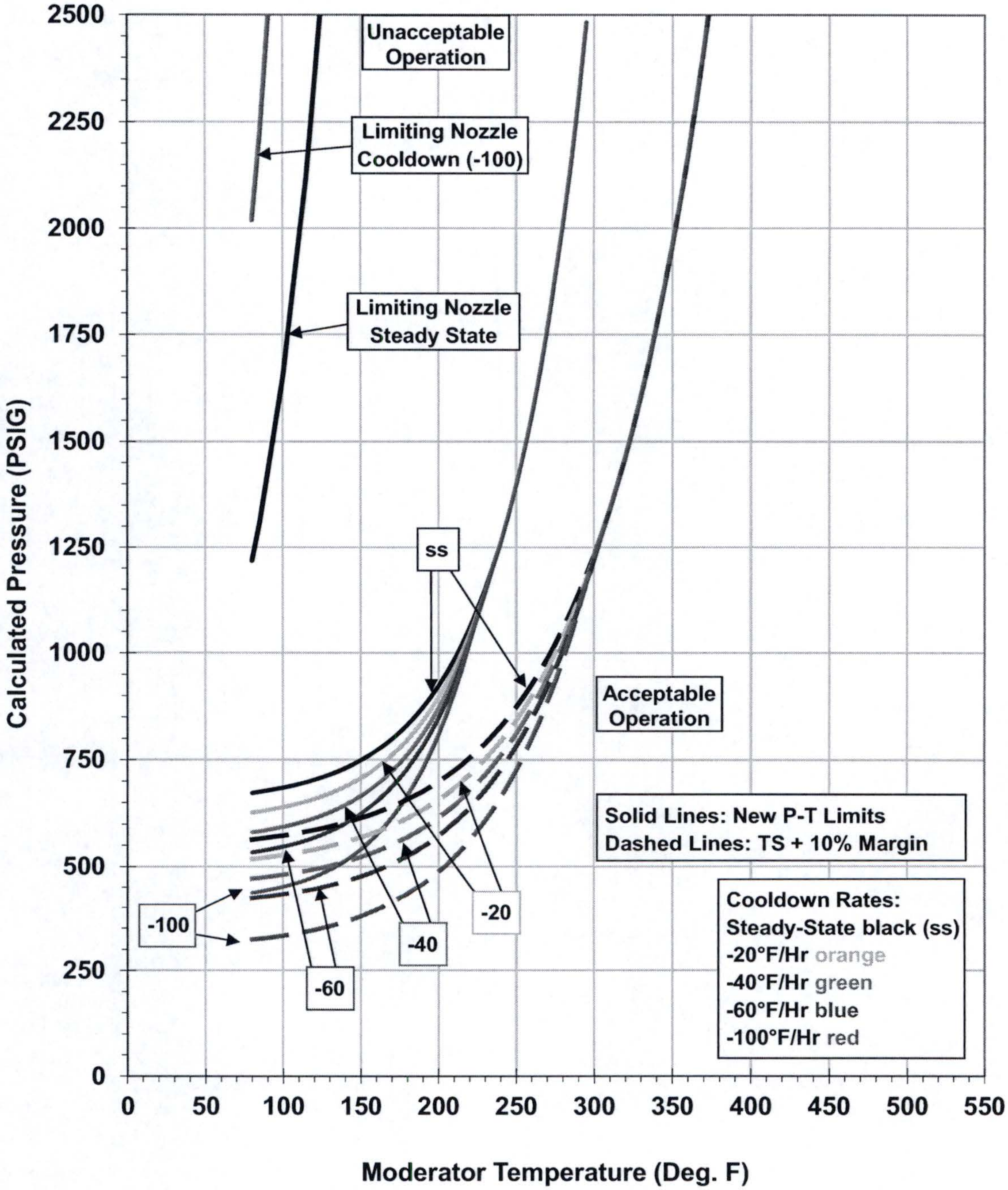


Figure B-2 Comparison of Surry Unit 2 Beltline Cooldown P-T Limits (Including Current P-T Limits without Pressure Adjustment + 10% Margin and New 68 EFPY P-T Limits) to 68 EFPY Nozzle P-T Limits, Without Margins for Instrumentation Errors

Note: "New P-T Limits" determined in Section 6. TS = Technical Specifications.

B.3 REFERENCES

B-1 NRC Regulatory Issue Summary 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components," U.S. Nuclear Regulatory Commission, October 2014. *[ADAMS Accession Number ML14149A165]*

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

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

B-4 Pressurized Water Reactor Owners Group (PWROG) Report PWROG-16045-NP, Revision 0, "Determination of Unirradiated RT_{NDT} and Upper-Shelf Energy Values of the Surry Units 1 and 2 Reactor Vessel Materials," March 2017.

B-5 *BWRVIP-173-A: BWR Vessel and Internals Project: Evaluation of Chemistry Data for BWR Vessel Nozzle Forging Materials.* EPRI, Palo Alto, CA: 2011. 1022835.

B-6 ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," American Society for Testing and Materials, July 1982.

B-7 NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Chapter 5 LWR Edition, Branch Technical Position 5-3, "Fracture Toughness Requirements," Revision 2, U.S. Nuclear Regulatory Commission, March 2007.

B-8 U.S. NRC Technical Letter Report TLR-RES/DE/CIB-2013-01, "Evaluation of the Beltline Region for Nuclear Reactor Pressure Vessels," Office of Nuclear Regulatory Research [RES], November 2014. *[ADAMS Accession Number ML14318A177]*

B-9 Oak Ridge National Laboratory Report, ORNL/TM-2010/246, "Stress and Fracture Mechanics Analyses of Boiling Water Reactor and Pressurized Water Reactor Pressure Vessel Nozzles – Revision 1," June 2012. *[ADAMS Accession Number ML110060164]*

B-10 ASME PVP2011-57015, "Additional Improvements to Appendix G of ASME Section XI Code for Nozzles," G. Stevens, H. Mehta, T. Griesbach, D. Sommerville, July 2011.

B-11 Surry Power Station Technical Specifications, Section 3.1.B, Amendments Nos. 285 and 285.

APPENDIX C OTHER RCPB FERRITIC COMPONENTS

10 CFR Part 50, Appendix G [Ref. C-1], requires that all Reactor Coolant Pressure Boundary (RCPB) components meet the requirements of Section III of the ASME Code. The lowest service temperature requirement (LST) for all RCPB components, which is specified in NB-3211 and NB-2332(b) of the Section III ASME Code, is the relevant requirement that would affect the pressure-temperature (P-T) limits. This requirement is applicable to ferritic materials outside of the reactor vessel with a nominal wall thickness greater than 2 ½ inches, such as piping, pumps and valves [Ref. C-2]. The Surry Unit 1 and 2 reactor coolant systems do not contain ferritic materials in the Class 1 piping, pumps and valves per Section 4.4 of this report. Therefore, the LST requirements of NB-2332(b) and NB-3211 are not applicable to the Surry Units 1 and 2 P-T limits.

The other ferritic RCPB components that are not part of the reactor vessel beltline or extended beltline in Surry Unit 1 and 2 consist of the replacement reactor vessel closure heads, replacement steam generators, and pressurizers.

The replacement reactor vessel closure head materials have been considered in the development of the P-T limits, see Section 4.5 of this report for the relevant inputs. Additionally, the Unit 1 replacement reactor vessel closure head was constructed to the French Construction Code (RCC-M) 1993 Edition with 1st Addenda June 1994, 2nd Addenda June 1995, 3rd Addenda June 1996 and Modification Sheets FM 797, 798, 801, 802, 803, 804, 805, 806, and 807. The sizing calculations, stress and fatigue analysis were performed to ASME Code Section III 1995 Edition through 1996 Addenda. The Design Report and Report of Reconciliation (References 14 and 15 of [C-3]) certify that the closure head meets the design requirements for the ASME Code Section III 1995 Edition through 1996 Addenda. The Unit 2 replacement reactor vessel closure head was constructed to the 1995 Edition through 1996 Addenda Section III ASME Code and met all applicable requirements at the time of construction.

The replacement steam generators were constructed to the 1974 Edition through Winter 1976 Addenda Section III ASME Code and met all applicable requirements at the time of construction. Therefore, no further consideration is necessary for these components with regards to P-T limits.

The pressurizers were constructed to the 1965 Edition through Winter 1965 Addenda Section III ASME Code and met all applicable requirements at the time of construction. No further consideration is necessary for these components with regards to P-T limits.

C.1 REFERENCES

C-1 Code of Federal Regulations, 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Federal Register, Volume 60, No. 243, dated December 19, 1995.

C-2 ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Division 1, Subsection NB, "Class 1 Components."

C-3 Surry Power Station Updated Final Safety Analysis Report (UFSAR), Revision 48, "Chapter 4: Reactor Coolant System," September 2016.

APPENDIX D LTOP SYSTEM ENABLE TEMPERATURE

D.1 ASME CODE CASE N-641

ASME Code Case N-641 [Ref. D-1] presents alternative procedures for calculating pressure-temperature relationships and low temperature overpressure protection (LTOP) system effective temperatures, T_e, and allowable pressures. The procedures provided in Code Case N-641 take into account alternative fracture toughness properties, circumferential and axial reference flaws, and plant-specific LTOP effective temperature calculations.

Per ASME Code Case N-641, the LTOP system shall be effective below the higher temperature determined in accordance with (1) and (2) below. Alternatively, LTOP systems shall be effective below the higher temperature determined in accordance with (1) and (3) below.

(1) a coolant temperature[(a)] of 200°F

(2) a coolant temperature[(a)] corresponding to a reactor vessel metal temperature[(b)], for all vessel beltline materials, where T_e is defined for inside axial surface flaws as RT_{NDT} + 40°F, and T_e is defined for inside circumferential surface flaws as RT_{NDT} - 85°F.

(3) a coolant temperature[(a)] corresponding to a reactor vessel metal temperature[(b)], for all vessel beltline materials, where T_e is calculated on a plant-specific basis for axial and circumferential reference flaws using the following equation:

$$T_e = RT_{NDT} + 50 \ln [((F * M_m (pR_i / t)) - 33.2) / 20.734]$$

Where,

F = 1.1, accumulation factor for safety relief valves

M_m = the value of M_m determined in accordance with G-2214.1, √in.

p = vessel design pressure, ksig

R_i = vessel inner radius, in.

t = vessel wall thickness, in.

Notes:

(a) The coolant temperature is the reactor coolant inlet temperature.

(b) The vessel metal temperature is the temperature at a distance 1/4 of the vessel section thickness from the clad/base metal interface in the vessel beltline region. RT_{NDT} is the highest adjusted reference temperature (for weld or base metal in the beltline region) at a distance 1/4 of the vessel section thickness from the vessel clad/base metal interface as determined by Regulatory Guide 1.99, Revision 2 [Ref. D-2].

Using the ASME Code Case N-641 equations and the following inputs, the Surry Units 1 and 2 LTOP system minimum enable temperature using Cases 2 and 3 was determined.

RT_{NDT}	= 219.4°F for 68 EFPY (at 1/4T per Table 5-7)
F	= 1.1
M_m	= 2.627 $\sqrt{in.}$ (See Section 4 for equations used to calculate M_m)
p	= 2.485 ksig (see Section 4.6, Item 6)
R_i	= 78.5 in. (see Section 4.6, Item 4)
t	= 8.05 in. (see Section 4.6, Item 4)

The LTOP system shall be effective below the higher temperature determined in accordance with (1) and (2) above, which has been determined to be 273°F for 68 EFPY. Alternatively, LTOP systems shall be effective below the higher temperature determined in accordance with (1) and (3) above, which has been determined to be 262°F for 68 EFPY. Therefore, the minimum required enable temperature (without margins for instrument uncertainty) for the Surry Units 1 and 2 reactor vessel is 262°F for 68 EFPY.

D.2 ASME CODE CASE N-514

The LTOP enable temperature can also be calculated based on ASME Code Case N-514 [Ref. D-3]. Per ASME Code Case N-514, the LTOP system shall be effective below the higher temperature determined in accordance with (A) and (B) below.

(A) a coolant temperature[(a)] of 200°F

(B) a coolant temperature[(a)] corresponding to a reactor vessel metal temperature[(b)] less than RT_{NDT} + 50°F

Notes:

(a) The coolant temperature is the reactor coolant inlet temperature.

(b) The vessel metal temperature is the temperature at a distance 1/4 of the vessel section thickness from the inside wetted surface in the vessel beltline region. RT_{NDT} is the highest adjusted reference temperature (for weld or base metal in the beltline region) at a distance 1/4 of the vessel section thickness from the vessel wetted inner surface as determined by Regulatory Guide 1.99, Revision 2. For the purpose of this calculation, the inside wetted surface is taken to be the clad/base metal interface.

Using the ASME Code Case N-514 equations and an RT_{NDT} value of 219.4°F for 68 EFPY (at 1/4T per Table 5-7), the Surry Units 1 and 2 LTOP system enable temperature using Cases (A) and (B) was determined to be 283°F.

D.3 LTOP ENABLE TEMPERATURE CONCLUSION

The Surry Power Station Technical Specifications [Ref. D-4] specifies an arming temperature of 350°F, which is conservative and remains valid for the Surry SLR period of operation. The margin to the 350°F value is sufficient to cover uncertainties utilizing either the Code Case N-641 [Ref. D-1] methodology or the more conservative Code Case N-514 [Ref. D-3] methodology.

D.4 REFERENCES

D-1 ASME Code Case N-641, "Alternative Pressure-Temperature Relationship and Low Temperature Overpressure Protection System Requirements Section XI, Division 1," ASME International, January 17, 2000.

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

D-3 ASME Code Case N-514, "Low Temperature Overpressure Protection Section XI, Division 1," ASME International, dated February 12, 1992.

D-4 Technical Specifications LCO 3.1.G.1.c.(4), Virginia Electric and Power Company, Docket No. 50-280, "Surry Power Station, Unit 1 Renewed Facility Operating License" Amendments 248 and 247.

APPENDIX E WELD MATERIAL HEAT # 0227 INITIAL RT_{NDT} AND UPPER-SHELF ENERGY DETERMINATION

Charpy V-notch data exists from multiple sources for the Surry Unit 2 Intermediate to Lower Shell Circumferential Weld (Heat # 0227, Grau Lo flux LW320). Table E-1 provides Charpy V-notch test data taken from the Record of Weld Material Qualification for Heat # 0227, Grau Lo flux LW320 per Certified Material Test Reports (CMTRs). Table E-2 provides supplemental Charpy V-notch test data also obtained from CMTRs. Table E-3 provides the Charpy V-notch test data taken from Reference E-1 for the Surry Unit 2 surveillance weld, which was fabricated using the same weld Heat and flux type as the Surry Unit 2 Intermediate to Lower Shell Circumferential Weld. Since the surveillance weld test data provides the most complete record of Charpy V-notch test information, it is appropriate to include this data for determination of the Surry Unit 2 Intermediate to Lower Shell Circumferential Weld initial material properties.

Table E-1 Weld Material Qualification Charpy V-Notch Test Data for Surry Unit 2 Intermediate to Lower Shell Circumferential Weld (Heat # 0227)[(a)]

Temperature (°C)	Temperature[(b)] (°F)	Energy (kgm/cm^2)	Energy[(b)] (ft-lb)
-12	10	11.4	66
-12	10	8.8	51
-12	10	8.0	46

Notes:

(a) Data obtained from CMTRs.

(b) Converted value. Energy values were converted from kgm/cm^2 to ft-lb utilizing the formula below. Note that 0.315 inch and 0.394 inch are the nominal dimensions of the Charpy specimen cross section per WCAP-8085 [Ref. E-1].

Energy (ft-lbs) = Energy (kgm/cm^2) $* \frac{14.223\ lb*cm^2}{kg*in^2} * \frac{3.28\ ft}{m}$ * (0.315 in. * 0.394 in.)

Table E-2 Supplemental Charpy V-Notch Test Data for Surry Unit 2 Intermediate to Lower Shell Circumferential Weld (Heat # 0227)[(a)]

Temperature (°C)	Temperature[(b)] (°F)	Energy (kgm/cm^2)	Energy[(b)] (ft-lb)
-12	10	8.1	47
-12	10	5.5	32
-12	10	6.6	38
-12	10	8.8	51
-12	10	7.5	43
-12	10	6.6	38
-12	10	11.4	66
-12	10	8.8	51
-12	10	8.8	51
-12	10	10.5	61
-12	10	10.4	60
-12	10	8.5	49
-12	10	9.5	55
-12	10	10.2	59
-12	10	10.0	58

Notes:

(a) Data obtained from CMTRs.

(b) Converted value. Energy values were converted from kgm/cm^2 to ft-lb utilizing the formula below. Note that 0.315 inch and 0.394 inch are the nominal dimensions of the Charpy specimen cross section per WCAP-8085 [Ref. E-1].

$$\text{Energy (ft-lbs)} = \text{Energy (kgm/cm}^2\text{)} * \frac{14.223\ lb*cm^2}{kg*in^2} * \frac{3.28\ ft}{m} * (0.315\text{ in.} * 0.394\text{ in.})$$

Table E-3 Charpy V-Notch Test Data for Surry Unit 2 Surveillance Weld (Heat # 0227)[(a)]

Temperature (°F)	Energy (ft-lb)	Shear (%)	Lateral Expansion (mils)
-100	7	9	5
-100	7.5	9	5
-100	7	5	3
-40	15.5	17	15
-40	24	37	20
-40	34	33	31
-20	31	53	29
-20	27.5	47	25
-20	29	33	25
10	53	68	47
10	47	58	40
10	35	47	33
40	50	74	50
40	55.5	74	51
40	53.5	68	51
73	75	100	68
73	81	100	72
73	78	100	71
210	69.5	100	66
210	72	100	70
210	86	100	80
300	91	100	82
300	91	100	81
300	91	100	83

Note:

(a) Data obtained from WCAP-8085 [Ref. E-1]. Since the surveillance weld test data provides the most complete record of Charpy V-notch test information, it is appropriate to include this data for determination of the Surry Unit 2 Intermediate to Lower Shell Circumferential Weld initial material properties.

Using the data summarized in Tables E-1 through E-3, the initial RT_{NDT} value for the Surry Unit 2 Intermediate to Lower Shell Circumferential Weld (Heat # 0227) must be determined using NUREG-0800, BTP 5-3 guidance [Ref. E-2] and in accordance with the ASME Code Section III, Subarticle NB-2331 requirements [Ref. E-3].

Following NUREG-0800, BTP 5-3 Position 1.1(1) guidance, T_{NDT} "may be assumed to be the temperature at which 41 J (30 ft-lbs) was obtained in Charpy V-notch tests, or -18°C (0°F), whichever was higher." To precisely determine the temperature at which 30 ft-lbs was obtained on the weld specimens, the unirradiated Charpy V-notch data was plotted and fit using a hyperbolic tangent curve-fitting software, CVGRAPH, Version 6.02. Only the minimum data points (from Tables E-1 through E-3) at each Charpy V-notch test temperature were used as input to the curve-fitting software, in accordance with ASME Code Section III, Subarticle NB-2331, Paragraph (a)(4). The resulting CVGRAPH figures are contained in the following pages for Charpy V-notch absorbed energy and lateral expansion.

Using these figures, the temperature at which 30 ft-lb absorbed energy was achieved was determined to be -5.6°F. Since this value is lower than 0°F, T_{NDT} for this weld material is set equal to 0°F per BTP 5-3 Position 1.1(1).

This estimate of T_{NDT} and the Charpy V-notch test data in Tables E-1 through E-3 are used to determine RT_{NDT}. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the Charpy V-notch test data is first checked at a temperature equal to the drop-weight T_{NDT} plus 60°F to determine if the material exhibits at least 50 ft-lb absorbed energy and 35 mils lateral expansion. Charpy V-notch tests were not performed at T_{NDT} + 60°F. However, multiple Charpy V-notch tests were conducted at T_{NDT} + 40°F (0°F + 40°F = 40°F) and did exhibit a minimum of 50 ft-lb absorbed energy and 35 mils lateral expansion. Thus, the test data are T_{NDT} limited. For completeness, the unirradiated Charpy V-notch data was plotted and fit using a hyperbolic tangent curve-fitting software, CVGRAPH, Version 6.02. Only the minimum data points (from Tables E-1 through E-3) at each Charpy V-notch test temperature were used as input to the curve-fitting software, in accordance with ASME Code Section III, Subarticle NB-2331, Paragraph (a)(4). The resulting CVGRAPH figures are contained in the following pages for Charpy V-notch absorbed energy and lateral expansion.

Using these figures, the temperatures at which 50 ft-lb absorbed energy and 35 mils lateral expansion were achieved may be obtained. In this case, the absorbed energy test data is more conservative than the lateral expansion test data; therefore, it becomes the dominant data set in defining initial RT_{NDT}.

$T_{50\ ft\text{-}lb}$ = 32.1°F
$T_{35\ mils}$ = 5.0°F
T_{Cv} = Max [$T_{50\ ft\text{-}lb}$, $T_{35\ mils}$]
T_{Cv} = Max [32.1°F, 5.0°F]
T_{Cv} = 32.1°F

Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(3), the initial RT_{NDT} is the higher of T_{NDT} (determined from the drop-weight tests) and T_{Cv} (determined above) minus 60°F.

RT_{NDT} = Max [T_{NDT}, T_{Cv} – 60°F]
RT_{NDT} = Max [0°F, 32.1 – 60°F]
RT_{NDT} = Max [0°F, -27.9°F]
RT_{NDT} = 0°F

Surry Unit 2 Intermediate to Lower Shell Circumferential Weld (Heat # 0227) Initial RT_{NDT} = 0°F

The current 10 CFR 50, Appendix G [Ref. E-4], requirements specify that USE be calculated based on ASTM E185-82 [Ref. E-5]. Herein, USE is calculated based on an interpretation of ASTM E185-82 that is best explained by the most recent version of the ASTM E185 manual (2016 version). Using the guidelines in ASTM E185-82 and E185-16 [Ref. E-6], the average of all Charpy data ≥ 95% shear is reported as the USE. In some instances, there may be data deemed 'out of family,' which are removed from the determination of the USE based on engineering judgment. However, the use of engineering judgment to remove 'out of family' data was not necessary for this material.

Intermediate to Lower Shell Circumferential Weld (Heat # 0227) USE = Average (75, 81, 78, 69.5, 72, 86, 91, 91, 91 ft-lbs) = **82 ft-lb**

Surry Unit 2 Intermediate to Lower Shell Circumferential Weld

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 5/18/2017 1:47 PM

A = 35.85 B = 33.65 C = 60.43 T0 = 5.00 D = 0.00

Correlation Coefficient = 0.962

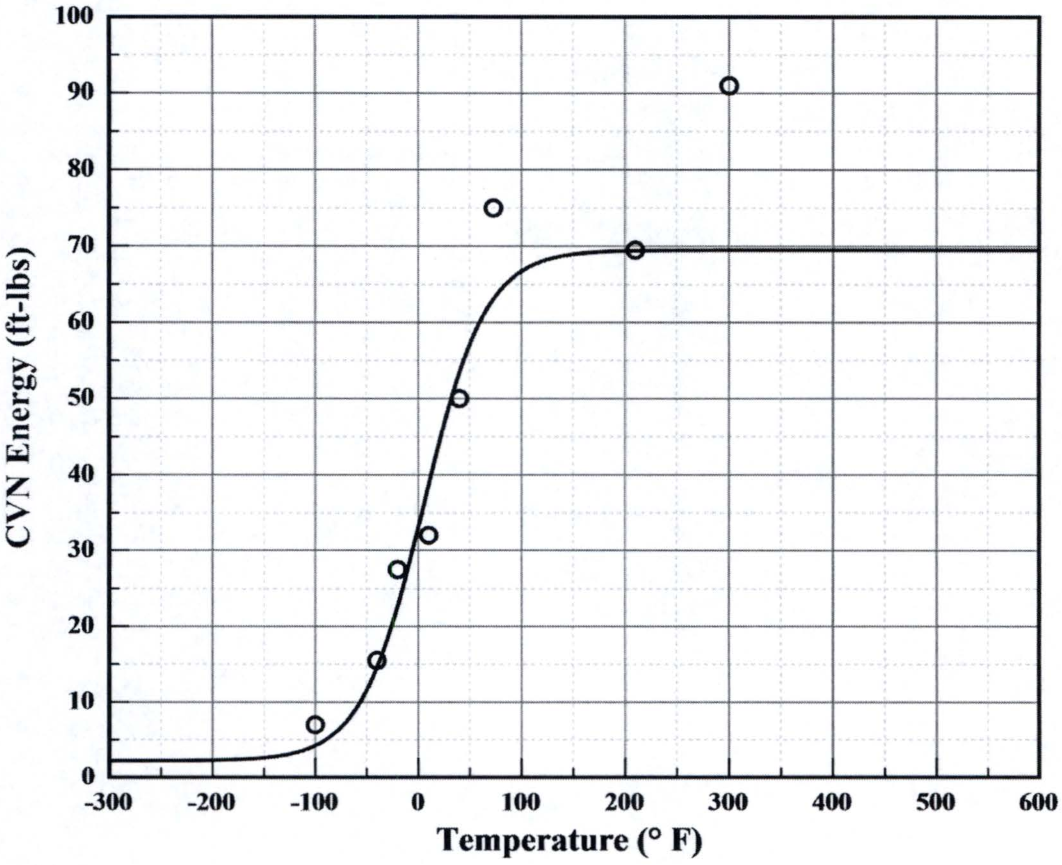
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf Energy = 69.50 (Fixed) Lower Shelf Energy = 2.20 (Fixed)

Temp@30 ft-lbs= -5.60° F Temp@35 ft-lbs= 3.50° F Temp@50 ft-lbs= 32.10° F

Plant: **Surry 2** Material: **WELD** Heat: **0227**

Orientation: **N/A** Capsule: **Unirradiated** Fluence: **0.00E+000 n/cm²**



CVGraph 6.02 05/18/2017 Page 1/2

Plant: **Surry 2** Material: **WELD** Heat: **0227**
Orientation: **N/A** Capsule: **Unirradiated** Fluence: **0.00E+000 n/cm²**

Surry Unit 2 Intermediate to Lower Shell Circumferential Weld

Charpy V-Notch Data

Temperature (° F)	Input CVN	Computed CVN	Differential
-100	7.0	4.2	2.78
-40	15.5	14.6	0.91
-20	27.5	22.7	4.83
10	32.0	38.6	-6.63
40	50.0	53.4	-3.42
73	75.0	63.1	11.91
210	69.5	69.4	0.08
300	91.0	69.5	21.50

CVGraph 6.02 05/18/2017 Page 2/2

Surry Unit 2 Intermediate to Lower Shell Circumferential Weld

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 5/18/2017 1:57 PM

A = 33.50 B = 32.50 C = 59.91 T0 = 2.18 D = 0.00

Correlation Coefficient = 0.980

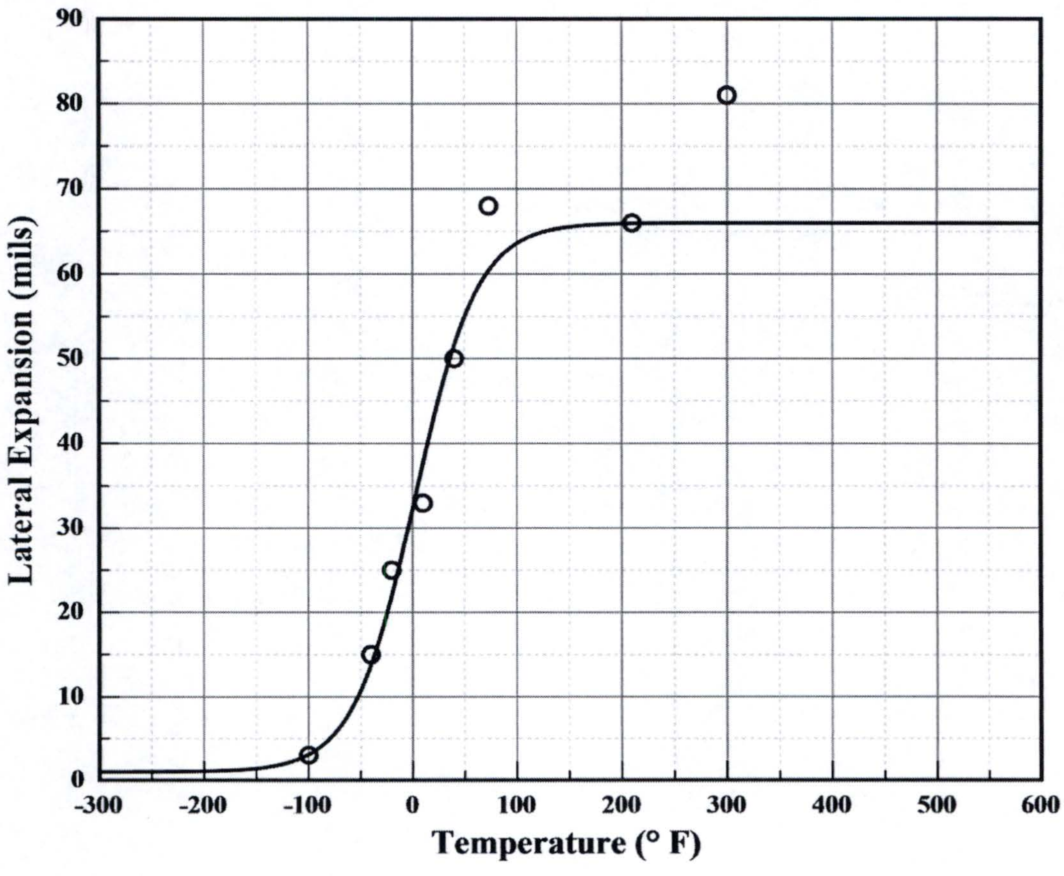
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf L.E. = 66.00 (Fixed) Lower Shelf L.E. = 1.00 (Fixed)

Temp@35 mils= 5.00° F

Plant: **Surry 2** Material: **WELD** Heat: **0227**

Orientation: **N/A** Capsule: **Unirradiated** Fluence: **0.00E+000 n/cm²**



CVGraph 6.02 05/18/2017 Page 1/2

Plant: **Surry 2** Material: **WELD** Heat: **0227**
Orientation: **N/A** Capsule: **Unirradiated** Fluence: **0.00E+000 n/cm²**

Surry Unit 2 Intermediate to Lower Shell Circumferential Weld

Charpy V-Notch Data

Temperature (° F)	Input L. E.	Computed L. E.	Differential
-100	3.0	3.1	-0.08
-40	15.0	13.8	1.23
-20	25.0	22.0	3.01
10	33.0	37.7	-4.72
40	50.0	51.7	-1.67
73	68.0	60.4	7.59
210	66.0	65.9	0.06
300	81.0	66.0	15.00

CVGraph 6.02 05/18/2017 Page 2/2

E.1 REFERENCES

E-1 Westinghouse Report WCAP-8085, Revision 0, "Virginia Electric & Power Co. Surry Unit No. 2 Reactor Vessel Radiation Surveillance Program," June 1973.

E-2 NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Chapter 5 of LWR Edition, Branch Technical Position 5-3, "Fracture Toughness Requirements," Revision 2, U.S. Nuclear Regulatory Commission, March 2007.

E-3 ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Division 1, Subarticle NB-2300, "Fracture Toughness Requirements for Material."

E-4 Code of Federal Regulations, 10 CFR 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Federal Register, Volume 60, No. 243, dated December 19, 1995.

E-5 ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," American Society for Testing and Materials, July 1982.

E-6 ASTM E185-16, "Standard Practice for Design of Surveillance Programs for Light-Water Moderated Nuclear Power Reactor Vessels," ASTM International, December 2016.

APPENDIX F SUMMARY OF THE APPLICABILITY OF P-T LIMIT CURVES FOR SURRY UNITS 1 AND 2

The Surry Units 1 and 2 P-T limit curves that are currently in Surry Power Station Technical Specifications (TS) [Ref. F-1] were first approved in WCAP-14177 [Ref. F-2] for End of License (EOL) and have applicability that was extended to 48 EFPY (per Reference F-12). Table F-1 contains a summary of the applicability of the Surry Units 1 and 2 P-T Limit curves. Figures F-1 and F-2 show the Surry Units 1 and 2 heatup and cooldown curves as currently depicted in the Surry Power Station Technical Specifications [Ref. F-12]. Tables F-2 and F-3 provide the data points corresponding to the heatup and cooldown curves, respectively, as currently depicted in the Surry Power Station Technical Specifications.

Table F-1 Surry Units 1 and 2 P-T Limit Curve Applicability History

Subject Document(s)	Content Relevant to Surry Units 1 and 2 P-T Limit Curves	Date	Reference Number(s)
WCAP-14177, Revision 0	P-T limit curves for 28.8 EFPY for Surry Unit 1 and 29.4 EFPY for Surry Unit 2 were created without inclusion of instrumentation errors. Note that this evaluation pre-dates the first approval of Westinghouse's current NRC-approved methodology in WCAP-14040-A, Revision 4 [Ref. F-3].	October 1994	F-2
SM-792, Revision 3 (Page 18/47) SM-945, Revision 0 (Page 26/102)	Per the subject documents, an adjustment of 21.5 psi to accommodate for the pressure difference between the pressurizer and reactor beltline was applied to the WCAP-14177 curves to create the TS curves. Additionally, the WCAP-14177 heatup curves are combined into one bounding heatup curve at temperatures of 315°F and above for the TS. These calculations also state that no instrumentation uncertainties were added to the P-T limit curves.	1995-1996	F-4 and F-5
NRC Letter Serial No. 95-197 (Page 14/47)	The subject document contains the original request to the NRC to incorporate the curves based on WCAP-14177 in the plant TS. It is stated that the curves do include a "correction for the effects of pressure measurement location" and repeats the statement that instrument uncertainties are not included in the curves. The differences between WCAP-14177 and the TS curves are a result of pressure measurement location adjustments.	June 1995	F-6
Letter from the NRC	NRC approved the P-T limits based on WCAP-14177 through amendment No. 207.	December 1995	F-7
WCAP-15130, Revision 1	P-T limit curves for End of License Extension (EOLE) were developed.	April 2001	F-8

Table F-1 Surry Units 1 and 2 P-T Limit Curve Applicability History

Subject Document(s)	Content Relevant to Surry Units 1 and 2 P-T Limit Curves	Date	Reference Number(s)
Letter from the NRC	The TS P-T Limits were changed to curves based on WCAP-15130, Revision 1.	January 2006	F-9
Letter from the NRC	The P-T Limits approved under amendment No. 207 were reinstated in the TS.	June 2006	F-10
Letter from the NRC	The applicability of the P-T Limits approved under amendment No. 207 was extended to 48 EFPY.	May 2011	F-11
Letter from the NRC	This reference represents the most recent TS P-T limit curve amendment (No. 285), which administratively alters the P-T limits.	June 2015	F-12

Thus, the limiting ART values used to create the TS curves (based on WCAP-14177) are those used for determination of applicability of the P-T limit curves at 48 and 68 EFPY with updated fluence, material properties, and Position 2.1 chemistry factor values.

In summary, the current Surry Units 1 and 2 Technical Specifications P-T limit curves are based on WCAP-14177 with minor administrative changes and an applied pressure measurement adjustment of 21.5 psi.

Surry Units 1 and 2
Reactor Coolant System Heatup Limitations

Material Property Basis
Limiting Material: Surry Unit 1 Intermediate to Lower Shell Circ Weld
Limiting ART Values for Surry 1 at 48 EFPY: 1/4-T, 228.4°F
3/4-T, 189.5°F
Limiting Boltup Temperature Surry 1 Initial RT_{NDT} Closure Flange Region: 10°F

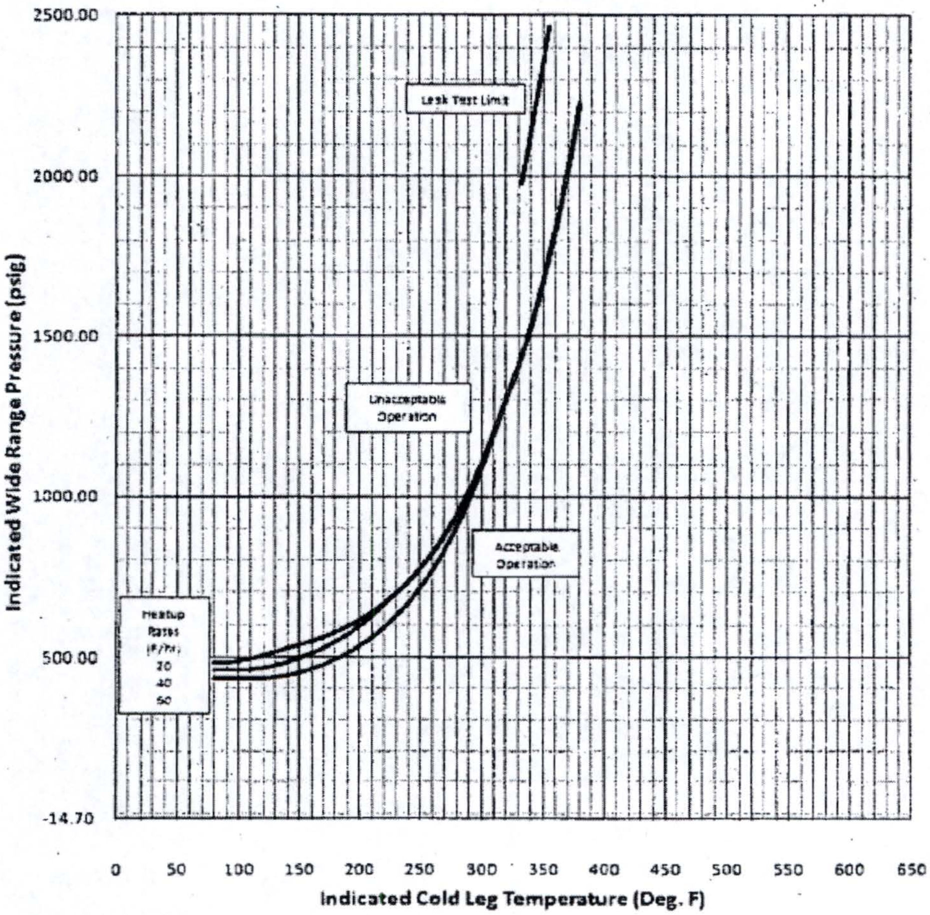


Figure 3.1-1 : Surry Units 1 and 2 Reactor Coolant System Heatup Limitations (Heatup Rates up to 60°F/hr) Applicable for 48 EFPY

Amendment Nos. 285, 285

Figure F-1 Surry Units 1 and 2 Heatup P-T Limit Curves as Depicted in the Surry Power Station Technical Specifications [Ref. F-12]

Surry Units 1 and 2
Reactor Coolant System Cooldown Limitations

Material Property Basis
Limiting Material: Surry Unit 1 Intermediate to Lower Shell Circ Weld
Limiting ART Values for Surry 1 at 48 EFPY: 1/4-T, 228.4°F
3/4-T, 189.5°F
Limiting Boltup Temperature Surry 1 Initial RT_{NDT} Closure Flange Region: 10°F

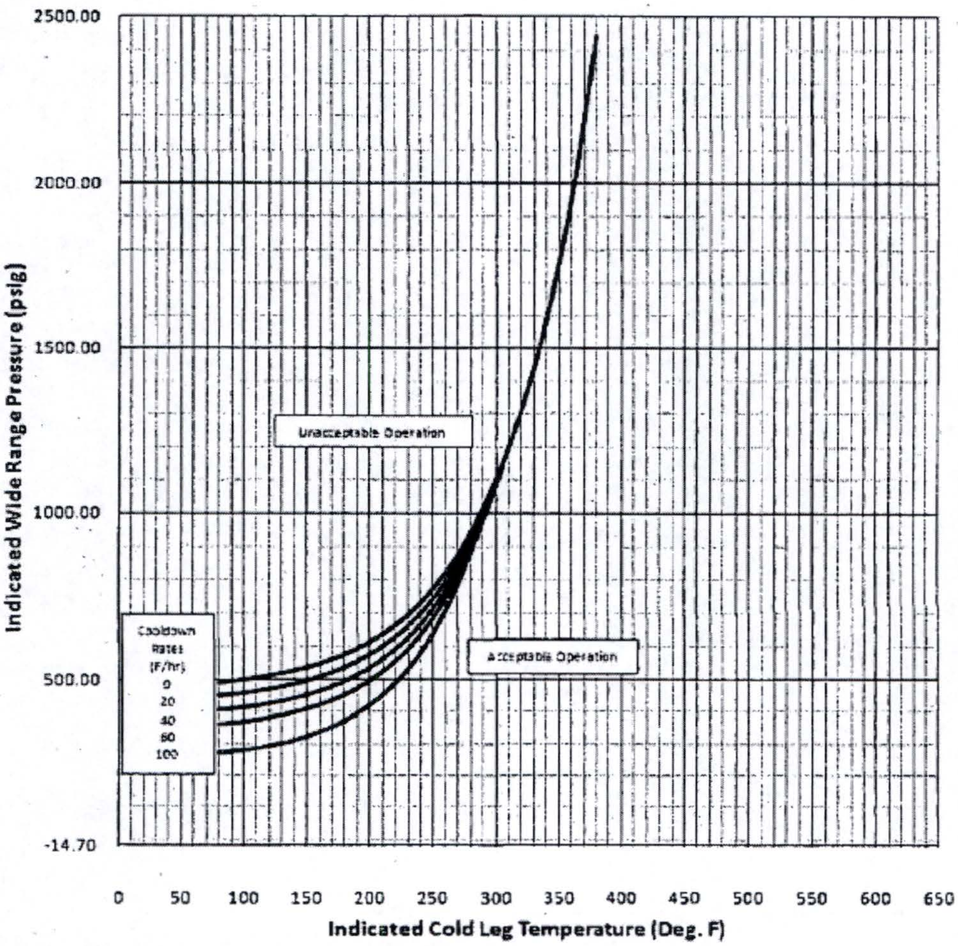


Figure 3.1-2 : Surry Units 1 and 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for 48 EFPY

Amendment Nos. 285, 285

Figure F-2 Surry Units 1 and 2 <u>Cooldown</u> P-T Limit Curves as Depicted in the Surry Power Station Technical Specifications [Ref. F-12]

Table F-2 Data Points for Surry Units 1 and 2 Current Technical Specifications Heatup P-T Limit Curves

20°F/hr Heatup		40°F/hr Heatup		60°F/hr Heatup	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
80	481.37	80	458.18	80	435.67
85	481.37	85	458.18	85	435.67
90	481.37	90	458.18	90	435.67
95	483.93	95	458.18	95	435.67
100	486.72	100	458.18	100	435.67
105	490.47	105	458.56	105	435.67
110	494.36	110	459.95	110	435.67
115	498.90	115	462.32	115	435.67
120	503.72	120	465.33	120	436.02
125	509.01	125	469.06	125	437.35
130	514.66	130	473.29	130	439.40
135	520.80	135	478.13	135	442.24
140	527.35	140	483.42	140	445.69
145	532.06	145	489.29	145	449.82
150	537.12	150	495.54	150	454.52
155	542.46	155	502.48	155	459.85
160	548.31	160	509.95	160	465.76
165	554.60	165	518.06	165	472.31
170	561.37	170	526.78	170	479.45
175	568.64	175	536.22	175	487.28
180	576.47	180	546.25	180	495.68
185	584.86	185	557.20	185	504.91
190	593.79	190	568.96	190	514.88
195	603.50	195	581.64	195	525.68
200	613.95	200	595.13	200	537.32
205	625.19	205	609.81	205	549.78
210	637.24	210	625.58	210	563.31
215	650.10	215	642.41	215	577.91
220	664.06	220	660.65	220	593.48
225	679.05	225	679.05	225	610.40
230	695.02	230	695.02	230	628.60
235	712.37	235	712.37	235	648.03
240	730.98	240	730.98	240	669.08
245	750.86	245	750.86	245	691.56
250	772.41	250	772.41	250	715.90
255	795.35	255	795.35	255	741.88
260	820.26	260	820.26	260	770.01
265	846.77	265	846.77	265	800.02
270	875.50	270	875.50	270	832.44

Table F-2 Data Points for Surry Units 1 and 2 Current Technical Specifications Heatup P-T Limit Curves

20°F/hr Heatup		40°F/hr Heatup		60°F/hr Heatup	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
275	906.20	275	906.20	275	867.18
280	939.14	280	939.14	280	904.40
285	974.78	285	974.78	285	944.39
290	1012.91	290	1012.91	290	987.33
295	1053.86	295	1053.86	295	1033.64
300	1097.82	300	1097.82	300	1083.17
305	1145.06	305	1145.06	305	1136.13
310	1195.82	310	1195.82	310	1193.21
315	1249.10	315	1249.10	315	1249.10
320	1302.07	320	1302.07	320	1302.07
325	1354.80	325	1354.80	325	1354.80
330	1409.89	330	1409.89	330	1409.89
335	1468.87	335	1468.87	335	1468.87
340	1531.93	340	1531.93	340	1531.93
345	1599.71	345	1599.71	345	1599.71
350	1672.05	350	1672.05	350	1672.05
355	1749.91	355	1749.91	355	1749.91
360	1833.09	360	1833.09	360	1833.09
365	1921.95	365	1921.95	365	1921.95
370	2017.08	370	2017.08	370	2017.08
375	2118.96	375	2118.96	375	2118.96
380	2227.79	380	2227.79	380	2227.79
385	2343.89	385	2343.89	385	2343.89

Leak Test Limit	
T (°F)	P (psig)
333	1978.5
355	2463.5

Table F-3 Data Points for Surry Units 1 and 2 Current Technical Specifications Cooldown P-T Limit Curves

Steady-State		20°F/hr Cooldown[a]		40°F/hr Cooldown[a]		60°F/hr Cooldown[a]		100°F/hr Cooldown[a]	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
80	491.03	80	449.28	80	406.68	80	363.08	80	272.64
85	492.91	85	451.20	85	408.55	85	364.90	85	274.35
90	495.04	90	453.26	90	410.56	90	366.88	90	276.26
95	497.32	95	455.51	95	412.78	95	369.07	95	278.43
100	499.77	100	457.93	100	415.17	100	371.44	100	280.80
105	502.41	105	460.56	105	417.80	105	374.07	105	283.47
110	505.25	110	463.38	110	420.62	110	376.91	110	286.38
115	508.30	115	466.45	115	423.72	115	380.04	115	289.62
120	511.58	120	469.75	120	427.05	120	383.42	120	293.15
125	515.10	125	473.33	125	430.69	125	387.14	125	297.07
130	518.89	130	477.17	130	434.60	130	391.15	130	301.33
135	522.97	135	481.33	135	438.80	135	395.53	135	306.02
140	527.35	140	485.81	140	443.39	140	400.27	140	311.12
145	532.06	145	490.65	145	448.38	145	405.37	145	316.68
150	537.12	150	495.77	150	453.76	150	410.94	150	322.74
155	542.46	155	501.40	155	459.60	155	417.02	155	329.39
160	548.31	160	507.45	160	465.88	160	423.56	160	336.58
165	554.60	165	513.99	165	472.69	165	430.68	165	344.44
170	561.37	170	521.01	170	480.02	170	438.28	170	352.94
175	568.64	175	528.61	175	487.96	175	446.61	175	362.16
180	576.47	180	536.76	180	496.41	180	455.58	180	372.17
185	584.86	185	545.46	185	505.66	185	465.32	185	383.07
190	593.79	190	554.93	190	515.60	190	475.80	190	394.84
195	603.50	195	565.14	195	526.36	195	487.16	195	407.57
200	613.95	200	576.12	200	537.82	200	499.30	200	421.38
205	625.19	205	587.83	205	550.33	205	512.54	205	436.28
210	637.24	210	600.55	210	563.77	210	526.80	210	452.44
215	650.10	215	614.27	215	578.30	215	542.11	215	469.96
220	664.06	220	629.02	220	593.79	220	558.70	220	488.86
225	679.05	225	644.76	225	610.66	225	576.64	225	509.23
230	695.02	230	661.84	230	628.80	230	595.82	230	531.28
235	712.37	235	680.23	235	648.22	235	616.67	235	555.04
240	730.98	240	699.86	240	669.26	240	638.97	240	580.76
245	750.86	245	721.17	245	691.79	245	663.19	245	608.44
250	772.41	250	743.88	250	716.20	250	689.09	250	638.25
255	795.35	255	768.54	255	742.32	255	717.23	255	670.62
260	820.26	260	794.84	260	770.62	260	747.30	260	705.32
265	846.77	265	823.36	265	800.90	265	779.91	265	742.77
270	875.50	270	853.82	270	833.62	270	814.86	270	783.25
275	906.20	275	886.57	275	868.75	275	852.48	275	826.80

Table F-3 Data Points for Surry Units 1 and 2 Current Technical Specifications Cooldown P-T Limit Curves

Steady-State		20°F/hr Cooldown[a]		40°F/hr Cooldown[a]		60°F/hr Cooldown[a]		100°F/hr Cooldown[a]	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
280	939.14	280	922.02	280	906.49	280	892.94	280	873.65
285	974.78	285	959.97	285	947.11	285	936.54	285	924.18
290	1012.91	290	1000.71	290	990.76	290	983.61	290	978.48
295	1053.86	295	1044.51	295	1037.77	295	1034.14	295	1036.84
300	1097.82	300	1091.61	300	1088.29	300	1088.27		
305	1145.06	305	1142.24	305	1142.68				
310	1195.82								
315	1250.37								
320	1308.86								
325	1371.62								
330	1438.89								
335	1511.21								
340	1588.69								
345	1671.46								
350	1760.72								
355	1856.03								
360	1958.14								
365	2067.32								
370	2184.34								
375	2308.98								
380	2442.42								

Note:

(a) The 20°F/hr and 40°F/hr cooldown curves are identical to the steady-state curve at 310°F and above. The 60°F/hr cooldown curve is identical to the steady-state curve at 305°F and above. The 100°F/hr cooldown curve is identical to the steady-state curve at 300°F and above.

F.1 REFERENCES

F-1 Surry Power Station Technical Specifications, Section 3.1.B, Amendments Nos. 285 and 285.

F-2 Westinghouse Report WCAP-14177, Revision 0, "Surry Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," October 1994.

F-3 Westinghouse Report WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," May 2004.

F-4 Virginia Power Calculation SM-792, Revision 3, "Surry 1 & 2 Composite P/T Limits Curve," January 1996.

F-5 Virginia Power Calculation SM-945, Revision 0, "Surry Unit 1 and 2 Heatup/Cooldown Curves and LTOPS Setpoint," February 1995.

F-6 Letter 95-197 from Virginia Electric and Power Company to the Nuclear Regulatory Commission, "Virginia Electric and Power Company, Surry Power Station Units 1 and 2, Request for Exemption – Code Case N-514, Proposed Technical Specifications Change, Revised Pressure/Temperature Limits and LTOPS Setpoint," dated June 8, 1995.

F-7 Letter from the NRC to Virginia Electric and Power Company, "Surry Units 1 and 2 – Issuance of Amendments RE: Surry, Units 1 and 2 Reactor Vessel Heatup and Cooldown Curves," dated December 28, 1995. *[ADAMS Accession Number ML012710054]*

F-8 Westinghouse Report WCAP-15130, Revision 1, "Surry Units 1 and 2 WOG Reactor Vessel 60-Year Evaluation Minigroup Heatup and Cooldown Limit Curves for Normal Operation," April 2001.

F-9 Letter from the NRC to Virginia Electric and Power Company, "Surry Power Station, Unit Nos. 1 and 2 – Issuance of Amendments on Reactor Coolant System Pressure and Temperature Limits," dated January 3, 2006. *[ADAMS Accession Number ML053550091]*

F-10 Letter from the NRC to Virginia Electric and Power Company, "Surry Power Station, Unit Nos. 1 and 2 – Issuance of Amendments to Reinstate Previous Reactor Coolant System Pressure and Temperature Limits," dated June 29, 2006. *[ADAMS Accession Number ML061710242]*

F-11 Letter from the NRC to Virginia Electric and Power Company, "Surry Power Station, Unit Nos. 1 and 2 – Issuance of Amendments regarding Reactor Vessel Heatup and Cooldown Curves for 48 Effective Full-Power Years," dated May 31, 2011. *[ADAMS Accession Number ML11110A111]*

F-12 Letter from the NRC to Virginia Electric and Power Company, "Surry Power Station, Unit Nos. 1 and 2, Issuance of Amendments Regarding Clarification of Reactor Coolant System Heatup and Cooldown Limitation Technical Specification Figures," dated June 26, 2015. *[ADAMS Accession Number ML15173A102]*

APPENDIX G CREDIBILITY EVALUATION OF THE SURRY UNITS 1 AND 2 SURVEILLANCE DATA

G.1 INTRODUCTION

Regulatory Guide 1.99, Revision 2 [Ref. G-1] describes general procedures acceptable to the NRC staff for calculating the effects of neutron radiation embrittlement of the low-alloy steels currently used for light-water-cooled reactor vessels. Positions 2.1 and 2.2 of Regulatory Guide 1.99, Revision 2, describe the method for calculating the adjusted reference temperature and Charpy upper-shelf energy of reactor vessel beltline materials using surveillance capsule data. The methods of Positions 2.1 and 2.2 can only be applied when two or more credible surveillance data sets become available from the reactor in question.

To date, there have been four surveillance capsules removed from the Surry Unit 1 reactor vessel; three were tested to provide Charpy data. Five plant-specific surveillance capsules were removed from the Surry Unit 2 reactor vessel; three were tested to provide Charpy data. Additional weld surveillance data will also be evaluated from other plants. To use the surveillance data, the data must be shown to be credible. In accordance with Regulatory Guide 1.99, Revision 2, the credibility of the surveillance data will be judged based on five criteria.

The purpose of this evaluation is to apply the credibility requirements of Regulatory Guide 1.99, Revision 2, to the Surry Units 1 and 2 reactor vessel surveillance data to determine if the surveillance data is credible.

G.2 EVALUATION

Criterion 1: Materials in the capsules should be those judged most likely to be controlling with regard to radiation embrittlement.

The beltline region of the reactor vessel is defined in Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements" [Ref. G-2], as follows:

> *"the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage."*

The Surry Unit 1 reactor vessel beltline region traditionally consists of the following materials:

1. Intermediate Shell Plates C4326-1 and C4326-2
2. Lower Shell Plates C4415-1 and C4415-2
3. Upper Shell Forging 122V109VA1
4. Upper to Intermediate Shell Circumferential Weld Seam (Heat # 25017, SAF 89 Flux Type, Flux Lot Number 1197).
5. Intermediate to Lower Shell Circumferential Weld Seam (Heat # 72445, Linde 80 Flux Type, (40%) Flux Lot Number 8597 and (60%) Flux Lot Number 8623)
6. Intermediate Shell Plate Longitudinal Weld Seams L3 and L4 (Heat # 8T1554, Linde 80 Flux Type, Flux Lot Number 8579)
7. Lower Shell Longitudinal Weld Seams L1 (Heat # 8T1554, Linde 80 Flux Type, Lot 8579) and L2 (Heat # 299L44, Linde 80 Flux Type, Lot 8596).

The Surry Unit 2 reactor vessel beltline region traditionally consists of the following materials:

1. Intermediate Shell Plates C4331-2 and C4339-2
2. Lower Shell Plates C4208-2 and C4339-1
3. Upper Shell Forging 123V303VA1
4. Upper to Intermediate Shell Circumferential Weld Seam (Heat # 4275, SAF 89 Flux Type, Flux Lot Number 02275)
5. Intermediate to Lower Shell Circumferential Weld Seam (Heat # 0227, Grau Lo Flux Type, Lot LW320)
6. Intermediate Shell Plate Longitudinal Weld Seams L3 (Heat # 72445, Linde 80 Flux Type, Flux Lot Number 8597) and L4 (50% - Heat # 72445, Linde 80 Flux Type, Flux Lot Number 8597 and 50% - Heat # 8T1762, Linde 80 Flux Type, Flux Lot Number 8597)
7. Lower Shell Longitudinal Weld Seams L1 (Heat # 8T1762, Linde 80 Flux Type, Flux Lot Number 8597) and L2 (Heat # 8T1762, Linde 80 Flux Type, (63%) Flux Lot Number 8597 and (37%) Flux Lot Number 8632).

Per WCAP-7723, Revision 0 [Ref. G-3] and WCAP-8085 Revision 0 [Ref. G-4], the Surry Units 1 and 2 respective surveillance programs were developed to the requirements of ASTM E185. WCAP-8085 specifically refers to the 1970 edition of ASTM E185 which states that the surveillance materials must be representative of materials in the highest flux region of the reactor.

Table 3-1 provides the initial material properties of the Surry Unit 1 reactor vessel beltline materials. Each of the beltline base metal materials has similar chemical properties. Lower Shell Plate C4415-1 has the highest initial RT_{NDT} value (other than the Upper Shell Forging), and is also representative of Lower Shell Plate C4415-2 which shares the same material heat number. Since this material is also in the high flux region of the reactor, this material meets the intent of Criterion 1. Per Table 3-1, each of the Surry Unit 1 beltline weld materials has similar USE and low initial RT_{NDT} values. Since Heat # 299L44 has the highest Cu wt. % value, and is located in the high flux region of the reactor, this material meets the intent of Criterion 1.

Table 3-3 provides the initial material properties of the Surry Unit 2 reactor vessel beltline materials. Each of the beltline base metal materials has similar chemical properties. Intermediate Shell Plate C4339-2 has the lowest initial USE value, and Upper Shell Forging 123V303VA1 has the highest initial RT_{NDT} value. Since Lower Shell Plate C4339-1 is also representative of Intermediate Shell Plate C4339-2, which shares the same material heat number, and this material is in the high flux region of the reactor, this material meets the intent of Criterion 1. Per Table 3-3, each of the Surry Unit 1 beltline weld materials has similar chemical properties and low USE values. Since Heat # 0227 has the highest initial RT_{NDT} value and is located in the high flux region of the reactor, this material meets the intent of Criterion 1.

<u>Based on the discussion above, Criterion 1 is met for the Surry Units 1 and 2 surveillance programs.</u>

Criterion 2: Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions should be small enough to permit the determination of the 30 ft-lb temperature and upper-shelf energy unambiguously.

Based on engineering judgment, the scatter in the data presented in the plots documented in BAW-2324 [Ref. G-5] and WCAP-16001 [Ref. G-6] is small enough to permit the determination of the 30 ft-lb temperature and the upper-shelf energy of the Surry Units 1 and 2 surveillance materials unambiguously.

<u>Hence, the Surry Units 1 and 2 surveillance programs meet this criterion.</u>

Criterion 3: When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Position 2.1 normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail this criterion for use in shift calculations, they may be credible for determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E185-82 [Ref. G-7].

The functional form of the least squares method as described in Regulatory Position 2.1 will be utilized to determine a best-fit line for this data and to determine if the scatter of these ΔRT_{NDT} values about this line is less than 28°F for welds and less than 17°F for the plates.

Following is the calculation of the best-fit lines. In addition, the recommended NRC methods for determining credibility will be followed. The NRC methods were presented to industry at a meeting held by the NRC on February 12 and 13, 1998 [Ref. G-8]. At this meeting, the NRC presented five cases. Of the five cases, three Cases will be used to represent the Surry Units 1 and 2 Surveillance Material:

Case 1: "Surveillance Data from Plant and No Other Source"

- Surry Unit 1 Lower Shell Plate C4415-1
- Surry Unit 2 Lower Shell Plate C4339-1
- Surry Unit 2 Weld Material Heat # 0227 – Intermediate to Lower Shell Circ. Weld

Case 4: "Surveillance Data from Plant and Other Sources"

- Weld Material Heat # 299L44 – Surry Unit 1 Lower Shell Longitudinal Weld L2 and Inlet Nozzle to Upper Shell Welds.

Case 5: "Surveillance Data from Other Sources Only"

- Weld Material Heat # 72445 from other Sources – Surry Unit 1 Intermediate to Lower Shell Circ. Weld and Surry Unit 2 Intermediate Shell Longitudinal Welds L3 and L4 (OD 50%).

Credibility Assessment Case 1: Lower Shell Plate C4415-1, Lower Shell Plate C4339-1, and Weld Heat # 0227

In accordance with the NRC guidelines, the plant-specific data from only Surry Units 1 and 2 will be analyzed first (Case 1). Case 1 interim chemistry factors are determined for both Surry Units 1 and 2 as summarized in Tables G-1 and G-2. Note that when evaluating the credibility of the surveillance weld data, the measured ΔRT_{NDT} values for the surveillance weld material do not include the adjustment ratio procedure of Regulatory Guide 1.99, Revision 2, Position 2.1, since this calculation is based on the actual surveillance weld material measured shift values. In addition, only plant-specific (Surry Unit 1 or Surry Unit 2) data is being considered; therefore, no temperature adjustment is required.

The Surry Unit 1 Lower Shell Plate C4415-1 surveillance material data and credibility conclusions pertain to the Lower Shell Plate C4415-1 and to Lower Shell Plate C4415-2 (same material heat). The Surry Unit 1, Case 1, chemistry factor is summarized in Table G-1.

Table G-1 Calculation of Interim Chemistry Factors for the Credibility Evaluation for Surry Unit 1

Material	Capsule	Capsule Fluence[(a)] (x 10^{19} n/cm^2, E > 1.0 MeV)	FF[(b)]	ΔRT_{NDT}[(c)] (°F)	FF*ΔRT_{NDT}(°F)	FF^2
Lower Shell Plate C4415-1 (Longitudinal)	T	0.271	0.644	50	32.21	0.415
	V	1.80	1.161	113	131.23	1.349
	X	2.11	1.203	86	103.46	1.447
				SUM:	266.91	3.211
	$CF_{C4415-1} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (266.91) \div (3.211) =$ **83.1°F**					

Notes:

(a) Capsule fluence values taken from Section 2.

(b) FF = fluence factor = $f^{(0.28 - 0.10*\log f)}$.

(c) ΔRT_{NDT} values obtained from Table 7-6 of BAW-2324 [Ref. G-5].

Surry Unit 2 Lower Shell Plate C4339-1 surveillance material data and credibility conclusions pertain to the Lower Shell Plate C4339-1 and Intermediate Shell Plate C4339-2 (same material heat). Surry Unit 2 Weld Material Heat # 0227 surveillance data and credibility conclusions only pertain to Surry Unit 2 Intermediate to Lower Shell Circumferential Weld. Surry Unit 2, Case 1, chemistry factors are summarized in Table G-2.

Table G-2 Calculation of Interim Chemistry Factors for the Credibility Evaluation for Surry Unit 2

Material	Capsule	Capsule Fluence(a) (x 10^{19} n/cm^2, E > 1.0 MeV)	FF(b)	ΔRT_{NDT}(c) (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Lower Shell Plate C4339-1 (Longitudinal)	X	0.297	0.668	59.08	39.45	0.446
	V	1.89	1.174	79.12	92.91	1.379
	Y	2.72	1.267	114.22	144.72	1.605
Lower Shell Plate C4339-1 (Transverse)	X	0.297	0.668	48.67	32.50	0.446
	V	1.89	1.174	63.60	74.68	1.379
	Y	2.72	1.267	106.81	135.33	1.605
				SUM:	519.59	6.860
	CF $_{C4339-1}$ = Σ(FF * ΔRT_{NDT}) ÷ Σ(FF^2) = (519.59) ÷ (6.860) = **75.7°F**					
Surveillance Weld Material (Heat # 0227)	X	0.297	0.668	95.65	63.86	0.446
	V	1.89	1.174	140.21	164.64	1.379
	Y	2.72	1.267	178.32	225.94	1.605
				SUM:	454.45	3.430
	CF $_{Heat\ \#\ 0227}$ = Σ(FF * ΔRT_{NDT}) ÷ Σ(FF^2) = (454.45) ÷ (3.430) = **132.5°F**					

Notes:

(a) Capsule fluence values taken from Section 2.

(b) FF = fluence factor = $f^{(0.28 - 0.10*\log f)}$.

(c) ΔRT_{NDT} values obtained from Table 5-12 of WCAP-16001 [Ref. G-6].

Credibility Assessment Case 4: Weld Heat # 299L44 (Surry Unit 1 and other sources)

Case 4 ("Surveillance Data from Plant and Other Sources") most closely represents the situation for the Surry Unit 1 Lower Shell Longitudinal Weld L2 and Inlet Nozzle to Upper Shell Welds (Heat # 299L44). In accordance with the NRC Case 4 guidelines, the data from Surry Unit 1 and all Capsules listed in Table 3-7 containing Weld Heat # 299L44 will be analyzed together. Data is adjusted to the mean chemical composition and operating temperature of the surveillance capsules. Table G-3 provides the chemistry and temperature adjustment for Weld Heat # 299L44 data from all sources. The average chemistry and temperature are used to calculate Adjusted ΔRT_{NDT} values and the interim CF for weld Heat # 299L44 data from all sources, as shown in Table G-4.

Table G-3 Mean Chemical Composition and Temperature for Weld Heat # 299L44[(a)]

Material	Capsule	Cu Wt. %	Ni Wt. %	Inlet Temperature during Period of Irradiation (°F)	Temperature Adjustment (°F)
Weld Metal Heat # 299L44 (Surry Unit 1 Data)	T	0.23	0.64	537	-13
	V			539	-11
	X			542	-8
Weld Metal Heat # 299L44 (Other Plant Data)	TMI2-LG1 (CR-3)	0.37	0.70	556	6
	W1(CR-3)			545	-5
	TMI1-E	0.33	0.67	556	6
	TMI1-C			556	6
	TMI2-LG1(TMI-2)			556	6
	CR3-LG1(ONS-3)	0.36	0.70	556	6
	A5	0.23	0.64	556	6
MEAN		**0.30**	**0.67**	**550**	

Note:

(a) Data obtained from Table 3-7 or calculated herein.

Table G-4 Calculation of Interim Chemistry Factor for the Credibility Evaluation of Weld Material Heat # 299L44

Capsule	Chemistry Factor Position 1.1	Capsule Fluence (x 10^{19} n/cm^2, E > 1.0 MeV)	FF$^{(a)}$	ΔRT_{NDT} (°F)	Adjusted $\Delta RT_{NDT}$$^{(b)}$ (°F)	FF*Adjusted ΔRT_{NDT} (°F)	FF^2
T	175.8	0.271	0.644	171	184.9	119.10	0.415
V	175.8	1.80	1.161	250	279.6	324.75	1.349
X	175.8	2.11	1.203	234	264.4	318.11	1.447
TMI2-LG1(CR-3)	234.0	0.830	0.948	216	195.4	185.15	0.898
W1	234.0	0.780	0.930	262	226.2	210.40	0.865
TMI1-E	215.2	0.107	0.431	74	76.0	32.72	0.185
TMI1-C	215.2	0.882	0.965	166	163.4	157.65	0.931
TMI2-LG1(TMI-2)	215.2	0.968	0.991	226	220.4	218.39	0.982
CR3-LG1	230.5	0.779	0.930	202	185.1	172.15	0.865
A5	175.8	2.75	1.270	246.6	295.5	375.26	1.612
					SUM:	2113.67	9.550
$CF_{\text{Heat \# 299L44}} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (2113.67) \div (9.550) =$ **221.3°F**							

Notes:

(a) FF = fluence factor = $f^{(0.28 - 0.10*\log f)}$.

(b) Adjusted ΔRT_{NDT} values are ΔRT_{NDT} values adjusted first to the mean operating temperature using the temperature adjustments in Table G-3, then to the mean chemical composition using the ratio procedure.

Credibility Assessment Case 5: Weld Heat # 72445 (other sources only)

Case 5 ("Surveillance Data from Other Sources Only") most closely represents the situation for the Surry Units 1 and 2 reactor vessels use of Weld Heat # 72445. Surry Unit 1 Intermediate to Lower Shell Circumferential Weld and Surry Unit 2 Intermediate Shell Longitudinal Welds L3 and L4 (OD 50%) are fabricated from Weld Heat # 72445, but neither plant included this weld metal heat in their original surveillance programs.

In accordance with the NRC Case 5 guidelines, the data from all capsules listed in Table 3-8 containing Weld Heat # 72445 will be analyzed together. Data is adjusted to the mean chemical composition and operating temperature of the surveillance capsules. Table G-4 provides the chemistry and temperature adjustment for Weld Heat # 72445 data from all sources. The average chemistry and temperature will be used to calculate Adjusted ΔRT_{NDT} values and the interim CF for Weld Heat # 72445 data from all sources, as shown in Table G-6.

Table G-5 Mean Chemical Composition and Temperature for Weld Heat # 72445[(a)]

Material	Capsule	Cu Wt. %	Ni Wt. %	Inlet Temperature during Period of Irradiation (°F)	Temperature Adjustment(°F)
Weld Metal Heat # 72445 (Other Plant Data)	CR3-LG1	0.22	0.59	556	11
	CR3-LG2	0.22	0.59	556	11
	W1	0.22	0.59	545	0
	Point Beach Unit 1: Capsule V	0.23	0.62	542	-3
	Point Beach Unit 1: Capsule S	0.23	0.62	542	-3
	Point Beach Unit 1: Capsule R	0.23	0.62	541.6	-3.4
	Point Beach Unit 1: Capsule T	0.23	0.62	533.4	-11.6
MEAN		**0.23**	**0.61**	**545**	

Note:

(a) Data obtained from Table 3-8 or calculated herein.

Table G-6 Calculation of Interim Chemistry Factor for the Credibility Evaluation of Weld Material Heat # 72445

Capsule	Chemistry Factor Position 1.1	Capsule Fluence (x 10^{19} n/cm^2, E > 1.0 MeV)	FF[(a)]	ΔRT_{NDT} (°F)	Adjusted ΔRT_{NDT}[(b)] (°F)	FF*Adjusted ΔRT_{NDT} (°F)	FF^2
CR3-LG1	165.5	0.510	0.812	139	154.5	125.46	0.659
CR3-LG2	165.5	1.67	1.141	164	180.3	205.72	1.303
W1	165.5	0.780	0.930	138	142.1	132.23	0.865
PB-1: Capsule V	172.4	0.634	0.872	107	103.0	89.81	0.761
PB-1: Capsule S	172.4	0.829	0.947	165	160.4	151.94	0.898
PB-1: Capsule R	172.4	2.19	1.213	155	150.1	182.00	1.471
PB-1: Capsule T	172.4	2.23	1.217	181	167.7	204.15	1.482
					SUM:	1091.31	7.438
$CF_{\text{Heat \# 72445}} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (1091.31) \div (7.438) =$ **146.7°F**							

Notes:

(a) FF = fluence factor = $f^{(0.28 - 0.10*\log f)}$.

(b) Adjusted ΔRT_{NDT} values are ΔRT_{NDT} values adjusted first to the mean operating temperature using the temperature adjustments in Table G-5, then to the mean chemical composition using the ratio procedure.

The scatter of ΔRT_{NDT} values about the functional form of a best-fit line drawn as described in Regulatory Guide 1.99, Revision 2, Position 2.1 [Ref. G-1] is presented in Table G-7 for Surry Unit 1 and in G-8 for Surry Unit 2.

Table G-7 Surry Unit 1 Calculated Surveillance Capsule Data Scatter about the Best-Fit Line

Material	Capsule	CF (Slope $_{best-fit}$) (°F)	Capsule Fluence (x 10^{19} n/cm^2, E > 1.0 MeV)	FF[(a)]	Measured ΔRT_{NDT} (°F)	Adjusted[(b)] ΔRT_{NDT} (°F)	Predicted ΔRT_{NDT} (°F)	Scatter ΔRT_{NDT}[(c)] (°F)	<17°F (Base Metal) <28°F (Weld)
Lower Shell Plate C4415-1 (Longitudinal)	T	83.1	0.271	0.644	50	50.00	53.54	3.54	Yes
	V	83.1	1.80	1.161	113	113.00	96.51	16.49	Yes
	X	83.1	2.11	1.203	86	86.00	99.97	13.97	Yes
Surveillance Weld Material (Heat # 299L44)	T	221.3	0.271	0.644	171	184.86	142.58	42.28	No
	V	221.3	1.80	1.161	250	279.63	257.00	22.63	Yes
	X	221.3	2.11	1.203	234	264.42	266.23	1.81	Yes
	TMI2-LG1	221.3	0.830	0.948	216	195.36	209.73	14.37	Yes
	W1	221.3	0.780	0.930	262	226.16	205.87	20.29	Yes
	TMI1-E	221.3	0.107	0.431	74	76.00	95.28	19.28	Yes
	TMI1-C	221.3	0.882	0.965	166	163.40	213.51	50.11	No
	TMI2-LG1	221.3	0.968	0.991	226	220.40	219.28	1.12	Yes
	CR3-LG1	221.3	0.779	0.930	202	185.12	205.80	20.68	Yes
	A5	221.3	2.75	1.270	246.6	295.54	280.99	14.55	Yes
Surveillance Weld Material (Heat # 72445)	CR3-LG1	146.7	0.510	0.812	139	154.50	119.12	35.38	No
	CR3-LG2	146.7	1.67	1.141	164	180.25	167.43	12.82	Yes
	W1	146.7	0.780	0.930	138	142.14	136.47	5.67	Yes
	PB-1: V	146.7	0.634	0.872	107	102.96	127.97	25.01	Yes
	PB-1: S	146.7	0.829	0.947	165	160.38	138.98	21.40	Yes
	PB-1: R	146.7	2.19	1.213	155	150.08	177.90	27.81	Yes
	PB-1: T	146.7	2.23	1.217	181	167.71	178.58	10.87	Yes

Notes:

(a) FF = fluence factor = $f^{(0.28 - 0.10*\log f)}$.

(b) Adjusted to mean temperature and chemistry, as applicable.

(c) Scatter ΔRT_{NDT} = Absolute Value [Predicted ΔRT_{NDT} – Adjusted ΔRT_{NDT}].

Table G-8 Surry Unit 2 Calculated Surveillance Capsule Data Scatter about the Best-Fit Line

Material	Capsule	CF (Slope $_{best-fit}$) (°F)	Capsule Fluence (x 10^{19} n/cm^2, E > 1.0 MeV)	FF[(a)]	Measured ΔRT_{NDT} (°F)	Adjusted[(b)] ΔRT_{NDT} (°F)	Predicted ΔRT_{NDT} (°F)	Scatter ΔRT_{NDT}[(c)] (°F)	<17°F (Base Metal) <28°F (Weld)
Lower Shell Plate C4339-1 (Longitudinal)	X	75.7	0.297	0.668	59.08	59.08	50.54	8.54	Yes
	V	75.7	1.89	1.174	79.12	79.12	88.89	9.77	Yes
	Y	75.7	2.72	1.267	114.22	114.22	95.92	18.30	No
Lower Shell Plate C4339-1 (Transverse)	X	75.7	0.297	0.668	48.67	48.67	50.54	1.87	Yes
	V	75.7	1.89	1.174	63.60	63.60	88.89	25.29	No
	Y	75.7	2.72	1.267	106.81	106.81	95.92	10.89	Yes
Surveillance Weld Material (Heat # 0227)	X	132.5	0.297	0.668	95.65	95.65	88.47	7.18	Yes
	V	132.5	1.89	1.174	140.21	140.21	155.59	15.38	Yes
	Y	132.5	2.72	1.267	178.32	178.32	167.88	10.44	Yes
Surveillance Weld Material (Heat # 72445)	CR3-LG1	146.7	0.510	0.812	139	154.50	119.12	35.38	No
	CR3-LG2	146.7	1.67	1.141	164	180.25	167.43	12.82	Yes
	W1	146.7	0.780	0.930	138	142.14	136.47	5.67	Yes
	PB-1: V	146.7	0.634	0.872	107	102.96	127.97	25.01	Yes
	PB-1: S	146.7	0.829	0.947	165	160.38	138.98	21.40	Yes
	PB-1: R	146.7	2.19	1.213	155	150.08	177.90	27.81	Yes
	PB-1: T	146.7	2.23	1.217	181	167.71	178.58	10.87	Yes

Notes:

(a) FF = fluence factor = $f^{(0.28 - 0.10*\log f)}$.

(b) Adjusted to mean temperature and chemistry, as applicable.

(c) Scatter ΔRT_{NDT} = Absolute Value [Predicted ΔRT_{NDT} – Adjusted ΔRT_{NDT}].

The data is deemed credible if all points in a data set fall within a +/- 1σ scatter band. Statistically, +/- 1σ would be expected to encompass 68% of the data. Tables G-7 and G-8 indicate that plate C4415-1, weld Heat # 299L44, weld Heat # 0227, and weld Heat # 72445 surveillance data falls inside the +/- 1σ scatter band, and plate C4339-1 surveillance data does not fall within the +/- 1σ scatter band. Therefore, the plate C4415-1, weld Heat # 299L44, weld Heat # 0227 data, and weld Heat # 72445 are deemed "credible", and C4339-1 is deemed "non-credible" per the third criterion.

Criterion 4: The irradiation temperature of the Charpy specimens in the capsule should match the vessel wall temperature at the cladding/base metal interface within +/- 25°F.

The capsule specimens are located in the reactor between the thermal shield and the vessel wall and are positioned opposite to the center of the core. The test capsules are contained in baskets attached to the thermal shield [Refs. G-3 and G-4]. The location of the specimens with respect to the reactor vessel beltline provides assurance that the reactor vessel wall and the specimens experience equivalent operating conditions such that the temperatures will not differ by more than 25°F.

Hence, Criterion 4 is met for the Surry Units 1 and 2 surveillance programs.

Criterion 5: The surveillance data for the correlation monitor material in the capsule should fall within the scatter band of the database for that material.

The Surry Units 1 and 2 surveillance programs contain Standard Reference Material (SRM). The material was obtained from an A533 Grade B, Class 1 plate (HSST Plate 02). NUREG/CR-6413, ORNL/TM-13133 [Ref. G-9] contains a plot of Residual vs. Fast Fluence for the SRM (Figure 11 in the report). This Figure shows a 2σ uncertainty of 50°F. The data used for this plot is contained in Table 14 in the NUREG report. However, the NUREG report does not consider the most up-to-date fluence and ΔRT_{NDT} values for Surry surveillance capsules. Thus, Table G-9 contains an updated calculation of Residual vs. Fast Fluence, considering the updated capsule fluence and ΔRT_{NDT} values for the Surry surveillance capsules.

Table G-9 Calculation of Residual vs. Fast Fluence for Surry Units 1 and 2

Capsule	Capsule fluence (x 10^{19} n/cm^2, E > 1.0 MeV)	FF	Measured Shift[(a)] (°F)	RG 1.99, Rev. 2[(b)] Shift (°F)	Residual[(c)] (°F)
Surry Unit 1 Capsule T	0.271	0.644	72	78.54	6.54
Surry Unit 1 Capsule V	1.80	1.161	142	141.57	0.43
Surry Unit 1 Capsule X	2.11	1.203	142	146.65	4.65
Surry Unit 2 Capsule X	0.297	0.668	62.19	81.39	19.20
Surry Unit 2 Capsule V	1.89	1.174	116.55	143.14	26.59
Surry Unit 2 Capsule Y	2.72	1.267	148.02	154.45	6.43

Notes:

(a) Measured ΔT_{30} values for the SRM were taken from Table 7-6 of BAW-2324 [Ref. G-5] for Surry Unit 1 and Table 5-12 of WCAP-16001 [Ref. G-6] for Surry Unit 2.

(b) Per NUREG/CR-6413, ORNL/TM-13133, the Cu and Ni values for the SRM (HSST Plate 02) are 0.17 and 0.64, respectively. This equates to a chemistry factor value of 121.9°F based on Regulatory Guide 1.99, Revision 2, Position 1.1. The calculated shift is thus equal to CF * FF.

(c) Residual = Absolute Value [Measured Shift – RG 1.99 Shift].

The residual is less than 50°F (the allowable scatter in NUREG/CR-4613, ORNL/TM-13133) for all capsules.

Hence, Criterion 5 is met for the Surry Units 1 and 2 surveillance programs.

G.3 CONCLUSION

Based on the preceding responses to all five criteria of Regulatory Guide 1.99, Revision 2, Section B:

- The Surry Unit 1 surveillance plate data are deemed "credible"
- The Surry Unit 2 surveillance plate data are deemed "non-credible"
- The Weld Heat # 0227 data are deemed "credible"
- The Weld Heat # 299L44 data are deemed "credible"
- The Weld Heat # 72445 data are deemed "credible"

G.4 REFERENCES

G-1 U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.

G-2 Code of Federal Regulations, 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Federal Register, Volume 60, No. 243, dated December 19, 1995.

G-3 Westinghouse Report WCAP-7723, Revision 0, "Virginia Electric and Power Co. Surry Unit No. 1 Reactor Vessel Radiation Surveillance Program," July 1971.

G-4 Westinghouse Report WCAP-8085, Revision 0, "Virginia Electric & Power Co. Surry Unit No. 2 Reactor Vessel Radiation Surveillance Program," June 1973.

G-5 Framatome ANP Report BAW-2324, Revision 0, "Analysis of Capsule X, Virginia Power Surry Unit No. 1, Reactor Vessel Material Surveillance Program," April 1998.

G-6 Westinghouse Report WCAP-16001, Revision 0, "Analysis of Capsule Y from Dominion Surry Unit 2 Reactor Vessel Radiation Surveillance Program," February 2003.

G-7 ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," American Society for Testing and Materials, 1982.

G-8 K. Wichman, M. Mitchell, and A. Hiser, US NRC, Generic Letter 92-01 and RPV Integrity Workshop Handouts, "NRC/Industry Workshop on RPV Integrity Issues," February 12, 1998. *[ADAMS Accession Number ML110070570]*

G-9 NUREG/CR-6413; ORNL/TM-13133, "Analysis of the Irradiation Data for A302B and A533B Correlation Monitor Materials," April 1996.

APPENDIX H COMPARISON OF AXIAL FLAW AND CIRCUMFERENTIAL FLAW P-T LIMIT CURVES

Per Table 5-7, the limiting Surry Units 1 and 2 1/4T ART value at 68 EFPY corresponds to an "Axial Flaw" material, while the limiting 3/4T ART value corresponds to a "Circumferential Flaw" material. The following comparison is completed to confirm that the "Axial Flaw" methodology based heatup and cooldown limit curves bound heatup and cooldown limit curves based on the "Circumferential Flaw" methodology.

Figure 6-1 presents the limiting heatup curves without margins for possible instrumentation errors using heatup rates of 20, 40, and 60°F/hr applicable for 68 EFPY, with the flange requirements and using the "Axial Flaw" methodology and the limiting "Axial Flaw" ART values summarized in Table 5-7. Figure 6-2 presents the limiting cooldown curves without margins for possible instrumentation errors using cooldown rates of 0 (steady-state), 20, 40, 60, and 100°F/hr applicable for 68 EFPY, with the flange requirements and using the "Axial Flaw" methodology and the limiting "Axial Flaw" ART values summarized in Table 5-7. The heatup and cooldown curves were generated using the 1998 through the 2000 Addenda ASME Code Section XI, Appendix G [Ref. H-1].

Figure H-1 of the Appendix presents the limiting heatup curves without margins for possible instrumentation errors using heatup rates of 20, 40, and 60°F/hr applicable for 68 EFPY, with the flange requirements and using the "Circumferential Flaw" methodology and the limiting "Circumferential Flaw" ART values summarized in Table 5-7. Figure H-2 presents the limiting cooldown curves without margins for possible instrumentation errors using cooldown rates of 0 (steady-state), 20, 40, 60, and 100°F/hr applicable for 68 EFPY, with the flange requirements and using the "Circumferential Flaw" methodology and the limiting "Circumferential Flaw" ART values summarized in Table 5-7. The heatup and cooldown curves were generated using the 1998 through the 2000 Addenda ASME Code Section XI, Appendix G. <u>Note that the "Circumferential Flaw" based heatup and cooldown limitations should not be used in plant operation based the following paragraph.</u>

Figure H-3 shows a comparison of the heatup limit curves developed using the "Axial Flaw" methodology and the "Circumferential Flaw" methodology. Similarly, Figure H-4 shows a comparison of the cooldown limit curves using the "Axial Flaw" methodology and the "Circumferential Flaw" methodology. Figures 6-5 and 6-6 indicate that the curves based on the "Axial Flaw" methodology and the "Axial Flaw" ART values represent the most limiting heatup and cooldown limitations. Therefore, the "Axial Flaw" based heatup and cooldown limit curves, summarized in Figure 6-1, Figure 6-2, Table 6-1, and Table 6-2 are considered the limiting Surry Units 1 and 2 heatup and cooldown limits generated using the 1998 through the 2000 Addenda ASME Code Section XI, Appendix G [Ref. H-1].

MATERIAL PROPERTY BASIS

LIMITING MATERIALS: Surry Unit 1 Intermediate to Lower Shell Circumferential Weld (Heat # 72445) and Surry Unit 2 Intermediate to Lower Shell Circumferential Weld (Heat # 0227, Position 2.1)

LIMITING ART VALUES AT 68 EFPY: 1/4T, 213.9°F (Circumferential Flaw)
3/4T, 179.8°F (Circumferential Flaw)

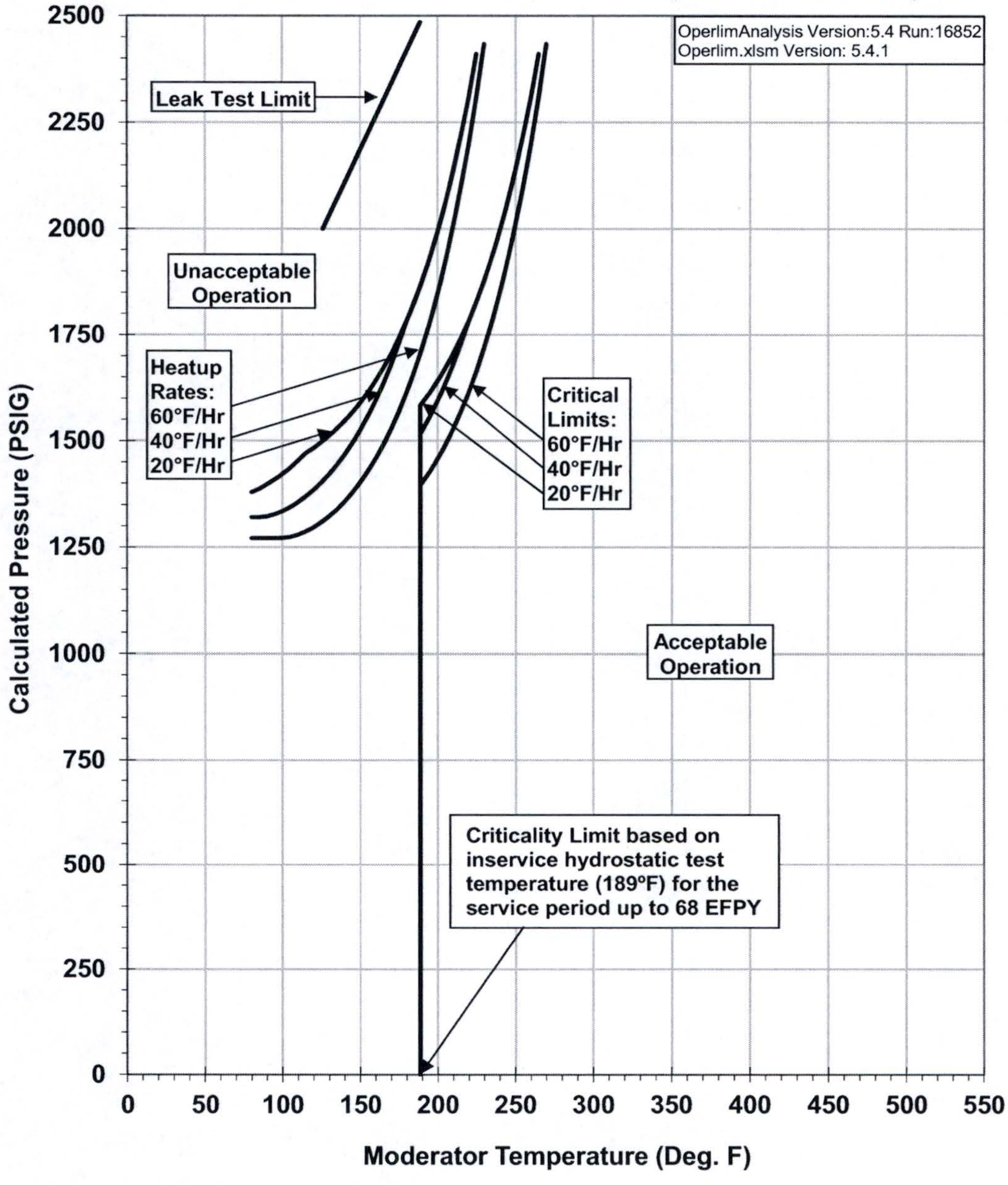


Figure H-1 Surry Units 1 and 2 Reactor Coolant System Heatup Limitations (Heatup Rates of 20, 40, and 60°F/hr) Applicable for 68 EFPY (with Flange Requirements and without Margins for Instrumentation Errors) using the 1998 Edition through the 2000 Addenda App. G "Circumferential Flaw" Methodology (w/ K_{Ic})

Note: Curves generated for informational and comparison purposes only.

MATERIAL PROPERTY BASIS

LIMITING MATERIALS: Surry Unit 1 Intermediate to Lower Shell Circumferential Weld (Heat # 72445) and Surry Unit 2 Intermediate to Lower Shell Circumferential Weld (Heat # 0227, Position 2.1)

LIMITING ART VALUES AT 68 EFPY: 1/4T, 213.9°F (Circumferential Flaw)
3/4T, 179.8°F (Circumferential Flaw)

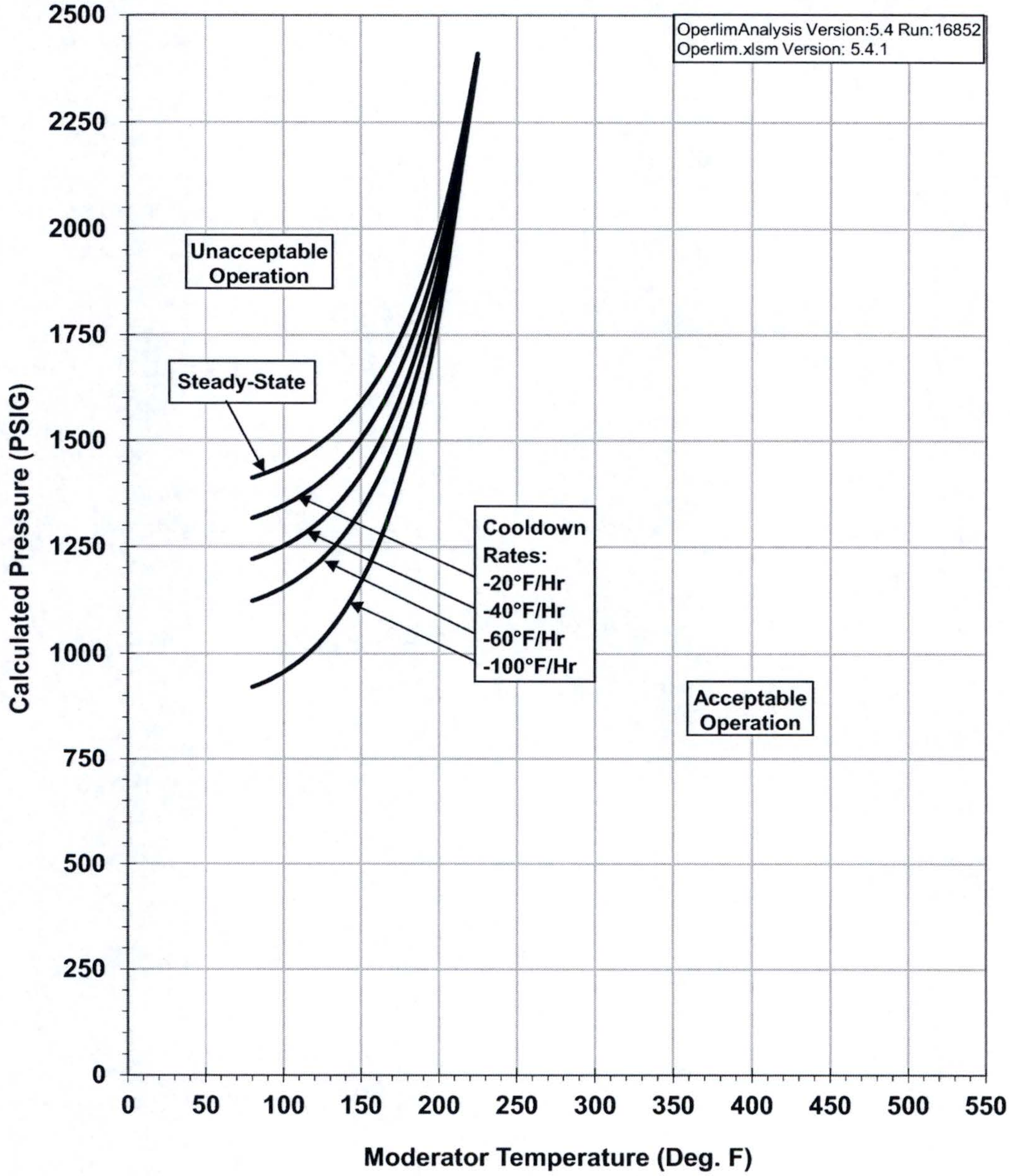


Figure H-2 Surry Units 1 and 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 20, 40, 60, and 100°F/hr) Applicable for 68 EFPY (with Flange Requirements and without Margins for Instrumentation Errors) using the 1998 Edition through the 2000 Addenda App. G "Circumferential Flaw" Methodology (w/ K_{Ic})

Note: Curves generated for informational and comparison purposes only.

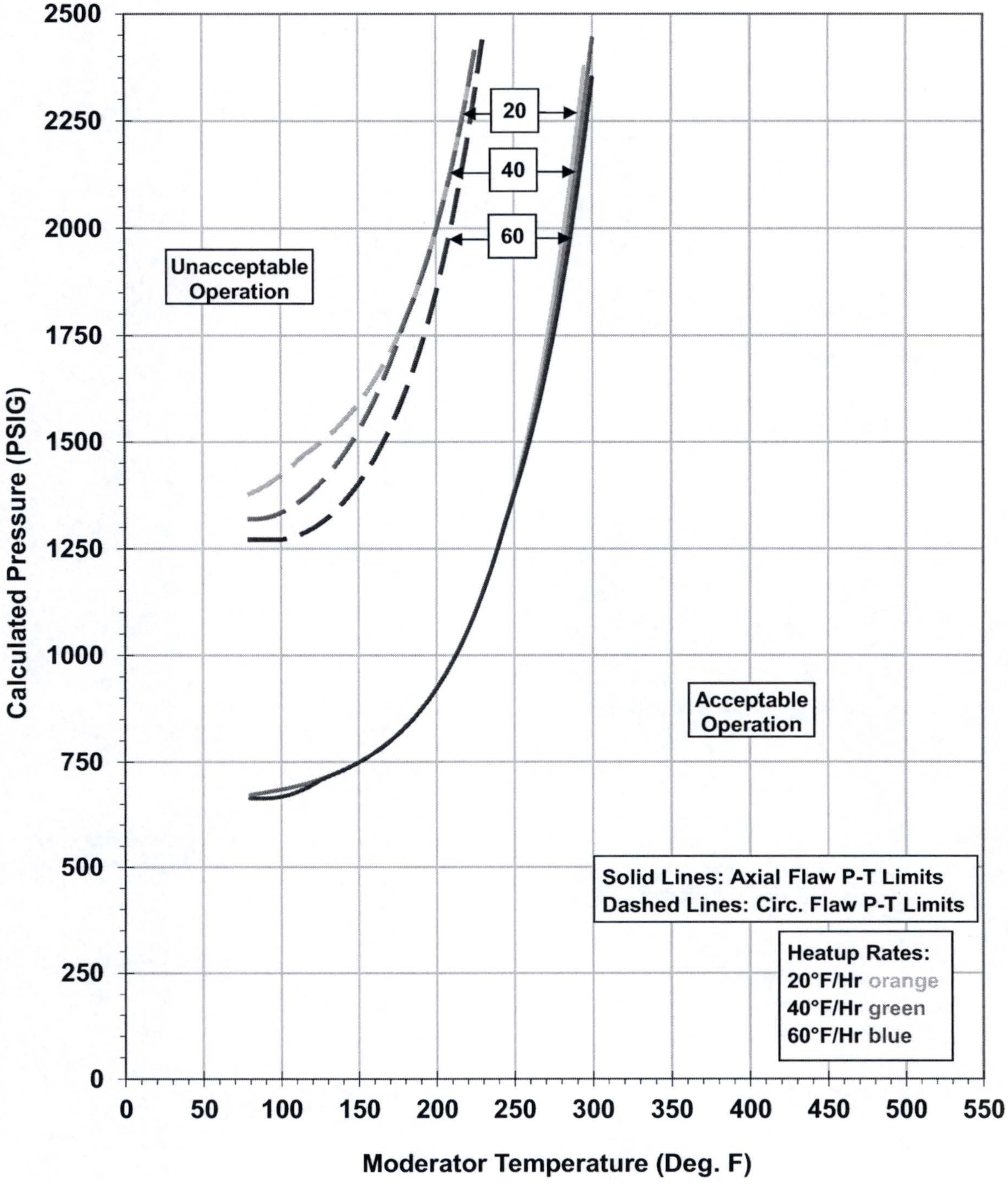


Figure H-3 Surry Units 1 and 2 Heatup P-T Limit Curve Comparison between Limiting "Axial Flaw" Based Curves and "Circumferential Flaw" Based Curves

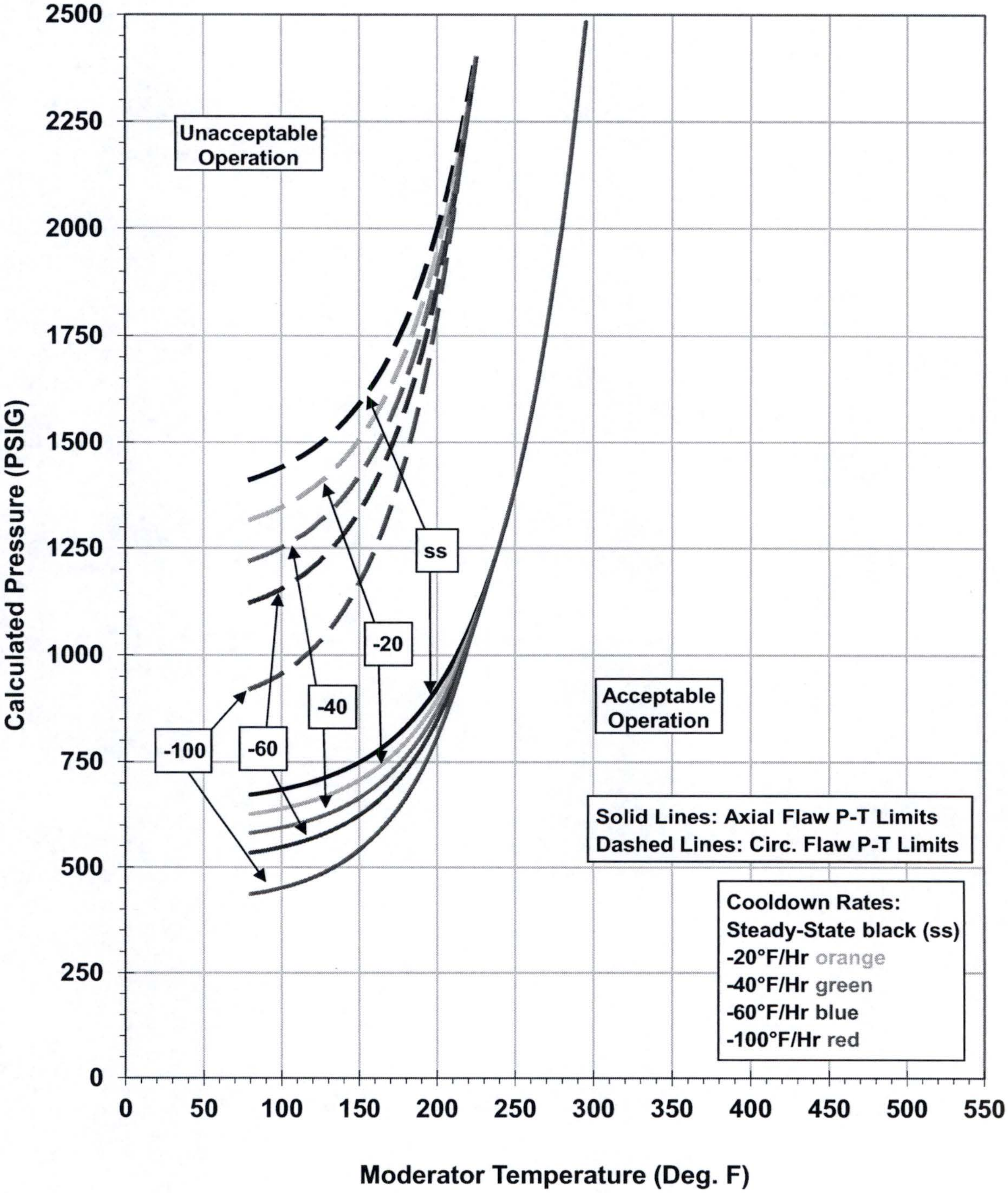


Figure H-4 Surry Units 1 and 2 Cooldown P-T Limit Curve Comparison between Limiting "Axial Flaw" Based Curves and "Circumferential Flaw" Based Curves

H.1 REFERENCES

H-1 Appendix G to the 1998 Edition through 2000 Addenda of ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Division 1, "Fracture Toughness Criteria for Protection Against Failure."

APPENDIX I SURRY UNITS 1 AND 2 UPPER-SHELF ENERGY EVALUATION AT 68 EFPY

I.1 INTRODUCTION

The decrease in Charpy upper-shelf energy (USE) is associated with the determination of acceptable RPV toughness during the license renewal period when the vessel is exposed to additional irradiation.

The requirements on USE are included in 10 CFR 50, Appendix G [Ref. I-1]. 10 CFR 50, Appendix G requires utilities to submit an analysis at least three years prior to the time that the USE of any RPV material is predicted to drop below 50 ft-lb, as measured by Charpy V-notch specimen testing.

There are two methods that can be used to predict the decrease in USE with irradiation, depending on the availability of credible surveillance capsule data as defined in Regulatory Guide 1.99, Revision 2 [Ref. I-2]. For vessel beltline materials that are not in the surveillance program or have non-credible data, the Charpy USE (Position 1.2) is assumed to decrease as a function of fluence and copper content, as indicated in Regulatory Guide 1.99, Revision 2. When two or more credible surveillance sets become available from the reactor, they may be used to determine the Charpy USE of the surveillance material. The surveillance data are then used in conjunction with the Regulatory Guide to predict the change in USE (Position 2.2) of the RPV material due to irradiation.

The 68 EFPY (SLR) Position 1.2 USE values of the vessel materials can be predicted using the corresponding 1/4T fluence projections, the copper content of the materials, and Figure 2 in Regulatory Guide 1.99, Revision 2.

The predicted Position 2.2 USE values are determined for the reactor vessel materials that are contained in the surveillance program by using the reduced plant surveillance data along with the corresponding 1/4T fluence projection. The reduced plant surveillance data was obtained from Table 7-6 of BAW-2324 [Ref. I-3] for Surry Unit 1. The reduced plant surveillance data was obtained from Table 5-12 of WCAP-16001, Revision 0 [Ref. I-4] for Surry Unit 2. The surveillance data was plotted in Regulatory Guide 1.99, Revision 2, Figure 2 (see Figures I-1 and I-2 of this report) using the surveillance capsule fluence values documented in Table 2-1 of this report for Surry Unit 1 and Table 2-2 of this report for Surry Unit 2. Bounding material fluence values, only, are shown in Figures I-1 and I-2 for some materials. This data was fitted by drawing a line parallel to the existing lines as the upper bound of all the surveillance data. These reduced lines were used instead of the existing lines to determine the Position 2.2 SLR USE values.

The projected USE values were calculated to determine if the Surry Units 1 and 2 beltline and extended beltline materials remain above the 50 ft-lb criterion at 68 EFPY. These calculations are summarized in Tables I-1 and I-2. Fluence values corresponding to the lowest extent of the nozzle welds at the surface were used to conservatively calculate the projected USE values for the nozzle forgings.

I.2 CONCLUSION

For Surry Unit 1, the limiting USE value at 68 EFPY is 32 ft-lb (see Table I-1); this value corresponds to the Intermediate to Lower Shell Circumferential Weld (Heat # 72445) using Position 1.2. For Surry Unit 2, the limiting USE value at 68 EFPY is 41 ft-lb (see Table I-2); this value corresponds to the Upper to Intermediate Shell Circumferential Weld (Heat # 4275) using Position 1.2.

The NRC has previously approved the use of the equivalent margins analysis (EMA) BAW-2494, Revision 1 [Ref. I- 5] to qualify all of the materials currently projected to drop below 50 ft-lb USE at 68 EFPY. These materials are identified by the notes in Tables 3-1, 3-3, 5-1 and 5-2 herein and are summarized below. The EMAs for these materials are updated for SLR under PWROG PA-MSC-1481. An EMA should be submitted 3 years before a material is projected to drop below 50 ft-lbs; however, no additional materials are projected to drop below 50 ft-lb USE during the SLR period of operation.

The following Surry Units 1 and 2 materials are addressed by EMAs in PA-MSC-1481 for SLR.

Surry Unit 1:

- Upper to Intermediate Shell Circumferential Weld, Heat # 25017
- Intermediate Shell Longitudinal Welds L3 and L4, Heat # 8T1554
- Intermediate to Lower Shell Circumferential Weld, Heat # 72445
- Lower Shell Longitudinal Weld L1, Heat # 8T1554
- Lower Shell Longitudinal Weld L2, Heat # 299L44
- Inlet Nozzle to Shell Welds, Heat # 299L44 and # 8T1762
  (Projected USE > 50 ft-lbs at 68 EFPY)
- Outlet Nozzle to Shell Welds, Heat # 8T1762 and # 8T1554B
  (Projected USE > 50 ft-lbs at 68 EFPY)

Surry Unit 2:

- Upper to Intermediate Shell Circumferential Weld, Heat # 4275
- Intermediate Shell Longitudinal Welds L3 and L4, Heat # 72445
- Intermediate Shell Longitudinal Weld L4, Heat # 8T1762
- Intermediate to Lower Shell Circumferential Weld, Heat # 0227
- Lower Shell Longitudinal Weld L1 and L2, Heat # 8T1762
- Inlet Nozzle to Shell Welds, Heat # 8T1762
  (Projected USE > 50 ft-lbs at 68 EFPY)
- Outlet Nozzle to Shell Welds, Rotterdam Weld
  (Projected USE > 50 ft-lbs at 68 EFPY)

Note that Dominion has conservatively elected to complete an EMA for the Surry Units 1 and 2 Inlet and Outlet Nozzle to Shell Welds even though these materials are not projected to drop below 50 ft-lbs through 68 EFPY using the methods herein. The inlet and outlet nozzle welds are the only materials included in PA-MSC-1481 that were not previously addressed by EMA. The EMA would be applicable to the Surry Units 1 and 2 nozzle to shell welds which exceed the fluence criterion of 1 x 10^{17} n/cm^2 before 68 EFPY. These materials include those listed on the following page.

- Surry Unit 1 Outlet Nozzle 1 to Upper Shell Weld
- Surry Unit 1 Inlet Nozzle 1 to Upper Shell Weld
- Surry Unit 1 Inlet Nozzle 3 to Upper Shell Weld
- Surry Unit 2 Outlet Nozzle 1 to Upper Shell Weld
- Surry Unit 2 Inlet Nozzle 1 to Upper Shell Weld
- Surry Unit 2 Inlet Nozzle 3 to Upper Shell Weld

For Surry Unit 1, the limiting USE value for materials not requiring an EMA at 68 EFPY is 54 ft-lb (see Table I-1); this value corresponds to the Inlet Nozzle to Upper Shell Welds (Heat # 299L44) using Position 2.2. For Surry Unit 2, the limiting USE value for materials not requiring an EMA at 68 EFPY is also 54 ft-lb (see Table I-2); this value corresponds to the Outlet Nozzle to Upper Shell Welds using Position 2.1. Except for the materials listed above, all of the beltline and extended beltline materials in the Surry Units 1 and 2 reactor vessels are projected to remain above the USE screening criterion value of 50 ft-lb (per 10 CFR 50, Appendix G) through SLR (68 EFPY).

Table I-1 Predicted USE Values at 68 EFPY for Surry Unit 1

RPV Material	Wt. % Cu[(a)]	SLR 1/4T Fluence[(b)] (x 10^{19} n/cm^2)	Initial USE[(a)] (ft-lb)	Projected USE Decrease[(c)] (%)	SLR USE (ft-lb)
		Position 1.2			
Upper Shell Forging 122V109VA1	0.11	0.465	114	17	95
Upper to Intermediate Shell Circumferential Weld[(e)] (Heat # 25017)	0.33	0.465	64	39	39[(e)]
Intermediate Shell Plate C4326-1	0.11	3.88	115	28	83
Intermediate Shell Plate C4326-2	0.11	3.88	94	28	68
Intermediate Shell Longitudinal Welds L3 and L4[(e)] (Heat # 8T1554)	0.16	0.771	64	29	45[(e)]
Intermediate to Lower Shell Circumferential Weld[(e)] (Heat # 72445)	0.22	3.89	64	50	32[(e)]
Lower Shell Plate C4415-1	0.102	3.92	103	27	75
Lower Shell Plate C4415-2	0.11	3.92	82	28.5	59
Lower Shell Longitudinal Weld L1[(e)] (Heat # 8T1554)	0.16	0.777	64	29	45[(e)]
Lower Shell Longitudinal Weld L2[(e)] (Heat # 299L44)	0.34	0.777	64	41	38[(e)]
Inlet Nozzle 1 to Upper Shell Weld (Heat # 299L44)	0.34	0.0188	64	24	49
Inlet Nozzle 2 to Upper Shell Weld (Heat # 299L44)	0.34	0.00484	64	24	49
Inlet Nozzle 3 to Upper Shell Weld (Heat # 299L44)	0.34	0.00672	64	24	49
Inlet Nozzle 1 to Upper Shell Weld (Heat # 8T1762)	0.19	0.0188	64	13	56
Inlet Nozzle 2 to Upper Shell Weld (Heat # 8T1762)	0.19	0.00484	64	13	56
Inlet Nozzle 3 to Upper Shell Weld (Heat # 8T1762)	0.19	0.00672	64	13	56
Outlet Nozzle 1 to Upper Shell Weld (Heat # 8T1762)	0.19	0.00502	64	13	56
Outlet Nozzle 2 to Upper Shell Weld (Heat # 8T1762)	0.19	0.00362	64	13	56
Outlet Nozzle 3 to Upper Shell Weld (Heat # 8T1762)	0.19	0.0140	64	13	56
Outlet Nozzle 1 to Upper Shell Weld (Heat # 8T1554B)	0.16	0.00502	64	12	56
Outlet Nozzle 2 to Upper Shell Weld (Heat # 8T1554B)	0.16	0.00362	64	12	56

Table I-1 Predicted USE Values at 68 EFPY for Surry Unit 1

RPV Material	Wt. % Cu[(a)]	SLR 1/4T Fluence[(b)] (x 10^{19} n/cm^2)	Initial USE[(a)] (ft-lb)	Projected USE Decrease[(c)] (%)	SLR USE (ft-lb)
Outlet Nozzle 3 to Upper Shell Weld (Heat # 8T1554B)	0.16	0.0140	64	12	56
Inlet Nozzle 1 (Heat # 9-4787)	0.159	0.0304	63	11	56
Inlet Nozzle 2 (Heat # 9-5078)	0.159	0.00784	64	10	58
Inlet Nozzle 3 (Heat # 9-4819)	0.159	0.0109	68	10	61
Outlet Nozzle 1 (Heat # 9-4825-1)	0.159	0.00813	68	10	61
Outlet Nozzle 2 (Heat # 9-4762)	0.159	0.00586	82	10	74
Outlet Nozzle 3 (Heat # 9-4788)	0.159	0.0227	71	10.5	64
Position 2.2[(d)]					
Lower Shell Plate C4415-1	0.102	3.92	103	28	74
Lower Shell Plate C4415-2	0.11	3.92	82	28	59
Lower Shell Longitudinal Weld L2[(e)] (Heat # 299L44)	0.34	0.777	64	35	42[(e)]
Inlet Nozzle 1 to Upper Shell Weld (Heat # 299L44)	0.34	0.0188	64	15	54
Inlet Nozzle 2 to Upper Shell Weld (Heat # 299L44)	0.34	0.00484	64	15	54
Inlet Nozzle 3 to Upper Shell Weld (Heat # 299L44)	0.34	0.00672	64	15	54

Notes:

(a) Material data is from Tables 3-1 and 3-2 of this report.

(b) The 1/4T fluence was calculated using the fluence data in Table 2-3, the Regulatory Guide 1.99, Revision 2 [Ref. I-2] correlation, and the Surry Units 1 and 2 reactor vessel wall thickness of 8.05 inches. The surface fluence at the lowest extent of the nozzle weld was used to represent the inlet and outlet nozzle forgings; this approach is conservative. Bounding material fluence values, only, are shown in Figure I-1 for the nozzle materials.

(c) The Position 1.2 USE decrease values were calculated by plotting the 1/4T fluence values on Figure 2 of Regulatory Guide 1.99, Revision 2 and using the material-specific Cu wt. % values.

(d) Surveillance data (deemed credible per Appendix G) from Table 7-6 of BAW-2324 [Ref. I-3] were used in the calculation of Surry Unit 1 Position 2.2 USE projections. Regulatory Guide 1.99, Revision 2, Position 2.2 indicates that an upper-bound line drawn parallel to the existing lines (in Figure 2 of the Guide) through the surveillance data points should be used in preference to the existing graph lines for determining the decrease in USE.

(e) These weld materials were previously addressed by EMA in BAW-2494, Revision 1 [Ref. I-5], and are included herein to establish a baseline for SLR evaluation. EMAs for these materials are addressed under PA-MSC-1481.

Table I-2 Predicted USE Values at 68 EFPY for Surry Unit 2

RPV Material	Wt. % Cu(a)	SLR 1/4T Fluence(b) (x 10^{19} n/cm^2)	Initial USE(a) (ft-lb)	Projected USE Decrease(c) (%)	SLR USE (ft-lb)
Position 1.2					
Upper Shell Forging 123V303VA1	0.11	0.534	104	18	85
Upper to Intermediate Shell Circumferential Weld(e) (Heat # 4275)	0.35	0.534	68	39	41(e)
Intermediate Shell Plate C4331-2	0.12	4.44	84	30	59
Intermediate Shell Plate C4339-2	0.11	4.44	83	29	59
Intermediate Shell Longitudinal Welds L3 and L4 (OD 50%)(e) (Heat # 72445)	0.22	0.796	64	34	42(e)
Intermediate Shell Longitudinal Weld L4 (ID 50%)(e) (Heat # 8T1762)	0.19	0.796	64	32	44(e)
Intermediate to Lower Shell Circ. Weld(e) (Heat # 0227)	0.187	4.45	82	47	43(e)
Lower Shell Plate C4208-2	0.15	4.48	94	35	61
Lower Shell Plate C4339-1	0.107	4.48	101	29	72
Lower Shell Longitudinal Weld L1 and L2(e) (Heat # 8T1762)	0.19	0.802	64	33	43(e)
Inlet Nozzle 1 to Upper Shell Weld (Heat # 8T1762)	0.19	0.0210	64	14	55
Inlet Nozzle 2 to Upper Shell Weld (Heat # 8T1762)	0.19	0.00484	64	13.5	55
Inlet Nozzle 3 to Upper Shell Weld (Heat # 8T1762)	0.19	0.00660	64	13.5	55
Outlet Nozzle 1 to Upper Shell Weld (Rotterdam)	0.35	0.00491	71	24	54
Outlet Nozzle 2 to Upper Shell Weld (Rotterdam)	0.35	0.00361	71	24	54
Outlet Nozzle 3 to Upper Shell Weld (Rotterdam)	0.35	0.0156	71	24	54
Inlet Nozzle 1 (Heat # 9-5104)	0.159	0.0340	73	12.5	64
Inlet Nozzle 2 (Heat # 9-4815)	0.159	0.00784	66	10	59
Inlet Nozzle 3 (Heat # 9-5205)	0.159	0.0107	67	10	60
Outlet Nozzle 1 (Heat # 9-4825-2)	0.159	0.00796	73	10	66
Outlet Nozzle 2 (Heat # 9-5086-1)	0.159	0.00585	77	10	69
Outlet Nozzle 3 (Heat # 9-5086-2)	0.159	0.0253	71	10.5	64
Position 2.2(d)					
Lower Shell Plate C4339-1	0.107	4.48	101	19	82
Intermediate Shell Plate C4339-2	0.11	4.44	83	19	67
Intermediate to Lower Shell Circ. Weld(e) (Heat # 0227)	0.187	4.45	82	42	48(e)

Notes on the following page.

Notes:

(a) Material data is from Tables 3-3 and 3-4 of this report.

(b) The 1/4T fluence was calculated using the fluence data in Table 2-4, the Regulatory Guide 1.99, Revision 2 [Ref. I-2] correlation, and the Surry Units 1 and 2 reactor vessel wall thickness of 8.05 inches. The surface fluence at the lowest extent of the nozzle weld was used to represent the inlet and outlet nozzle forgings; this approach is conservative. Bounding material fluence values, only, are shown in Figure I-2 for the nozzle materials.

(c) The Position 1.2 USE decrease values were calculated by plotting the 1/4T fluence values on Figure 2 of Regulatory Guide 1.99, Revision 2 and using the material-specific Cu wt. % values.

(d) Surveillance data (deemed credible and non-credible per Appendix G) from Table 5-12 of WCAP-16001, Revision 0 [Ref. I-4] were used for Surry Unit 2 Position 2.2 USE projections. Regulatory Guide 1.99, Revision 2, Position 2.2 indicates that an upper-bound line drawn parallel to the existing lines (in Figure 2 of the Guide) through the surveillance data points should be used in preference to the existing graph lines for determining the decrease in USE. Credibility Criterion 3 in the Discussion section of Regulatory Guide 1.99, Revision 2, indicates that even if the surveillance data are not considered credible for determination of ΔRT_{NDT}, "they may be credible for determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E 185-82." Thus, the surveillance data may be used for Surry Unit 2 USE projections.

(e) These weld materials were previously addressed by EMA in BAW-2494, Revision 1 [Ref. I-5], and are included herein to establish a baseline for SLR evaluation. EMAs for these materials are addressed under PA-MSC-1481.

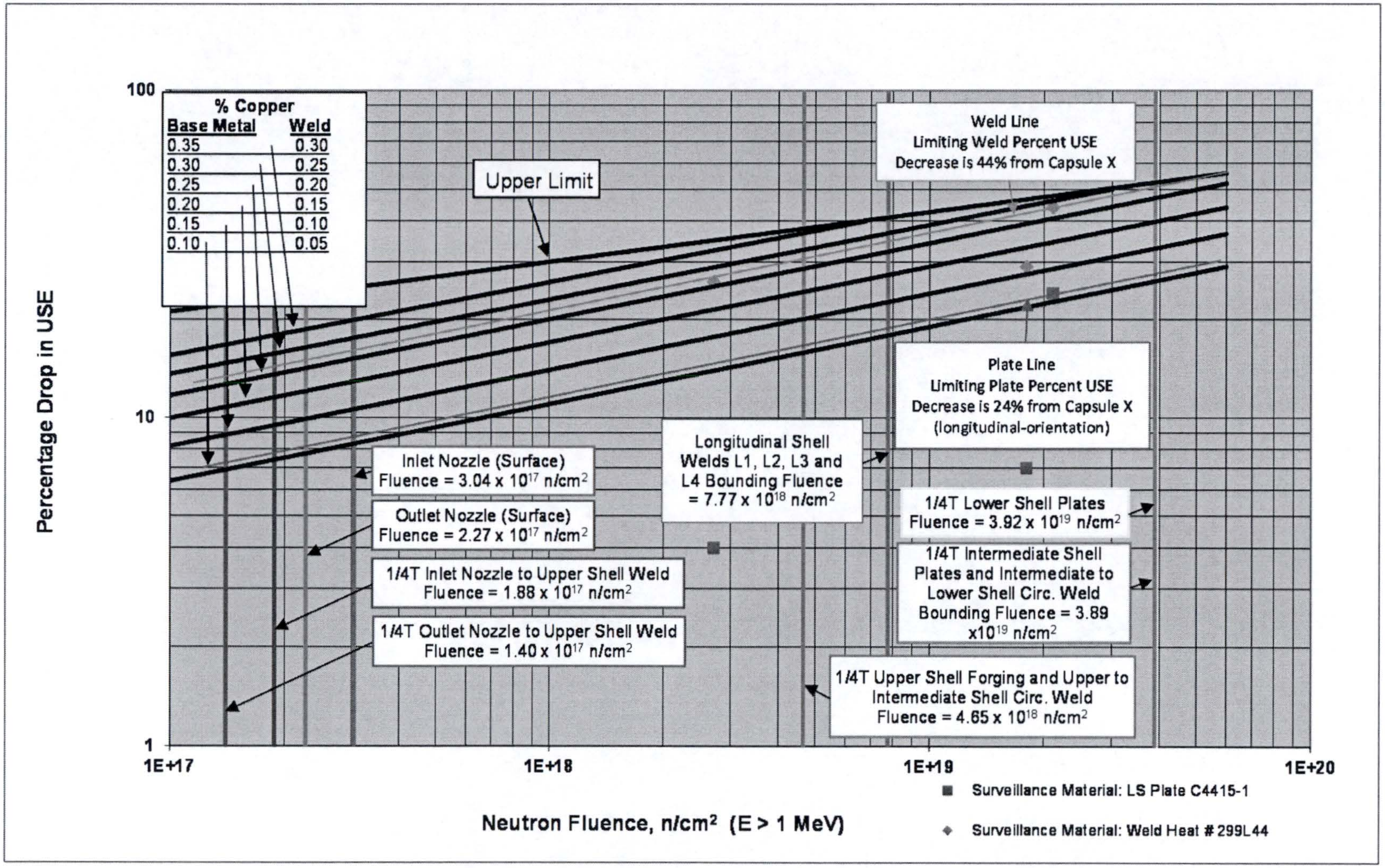


Figure I-1 **Regulatory Guide 1.99, Revision 2 Predicted Decrease in Upper-Shelf Energy as a Function of Copper and Fluence for Surry Unit 1 at 68 EFPY**

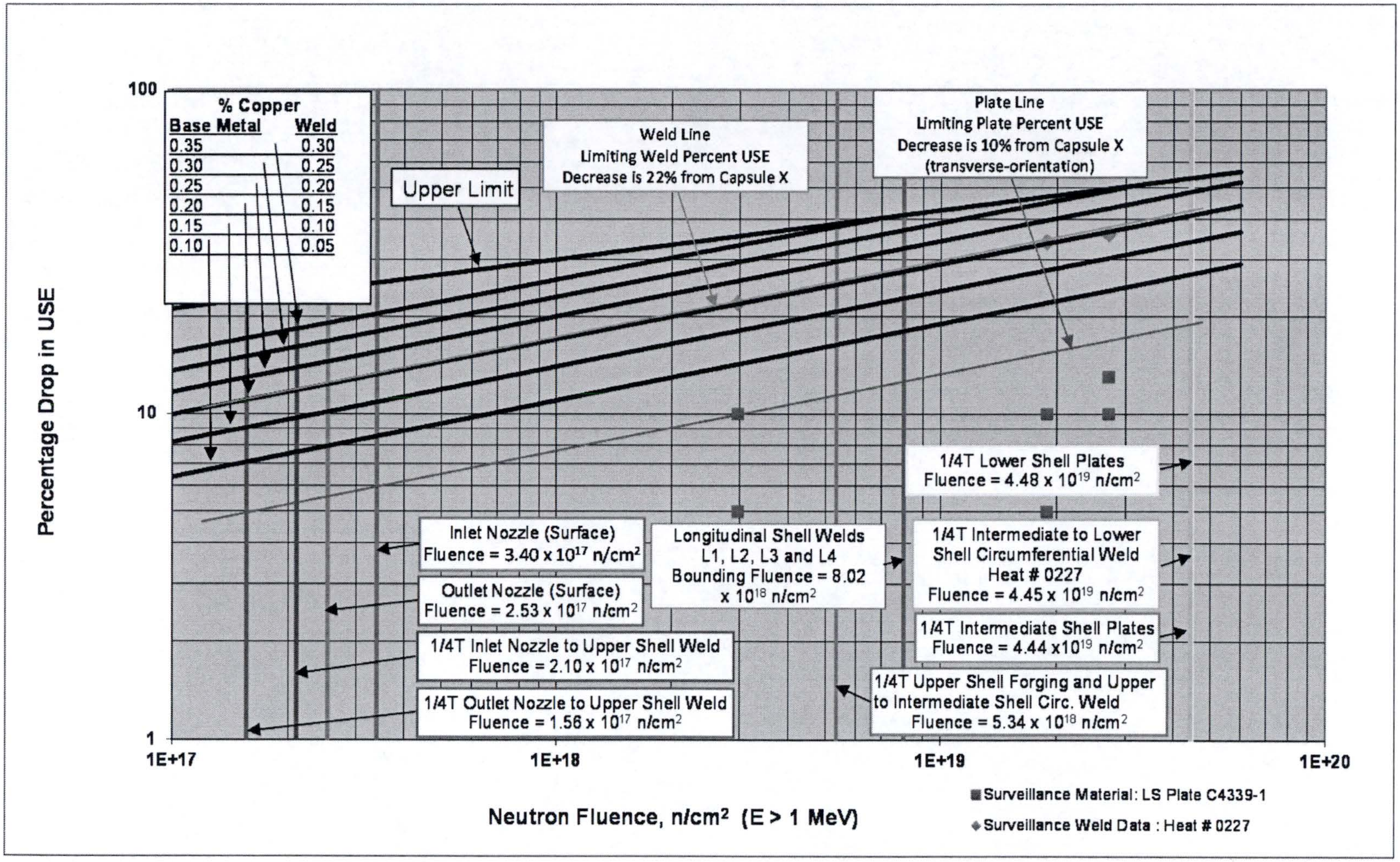


Figure I-2 **Regulatory Guide 1.99, Revision 2 Predicted Decrease in Upper-Shelf Energy as a Function of Copper and Fluence for Surry Unit 2 at 68 EFPY**

I.3 REFERENCES

I-1 Code of Federal Regulations, 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Federal Register, Volume 60, No. 243, dated December 19, 1995.

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

I-3 Framatome ANP Report BAW-2324, Revision 0, "Analysis of Capsule X, Virginia Power Surry Unit No. 1, Reactor Vessel Material Surveillance Program," April 1998.

I-4 Westinghouse Report WCAP-16001, Revision 0, "Analysis of Capsule Y from Dominion Surry Unit 2 Reactor Vessel Radiation Surveillance Program," February 2003.

I-5 Framatome ANP Report BAW-2494, Revision 1, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessel of Surry Units 1 and 2 for Extended Life through 48 Effective Full Power Years," September 2005.

APPENDIX J MATERIAL PROPERTY INPUT COMPARISON

This appendix provides tables which compare the material property input values utilized in this report, taken primarily from PWROG-16045-NP [Ref. J-1], with those utilized in Dominion calculation SM-1008, Addendum 00M [Ref. J-2] and the Surry Power Station Updated Final Safety Analysis Report (UFSAR) [Ref. J-3], as applicable.

Table J-1 Comparison of Previous and Current Initial RT_{NDT} Values for Surry Unit 1

Material Identification	Previous Initial RT_{NDT}(a) (°F)	Current Initial RT_{NDT}(b) (°F)
Replacement Reactor Vessel Closure Head Flange E4381/E4382	-67	-67
Reactor Vessel Flange FV-1870	10	-114.6
Inlet Nozzle 1 (Heat # 9-4787)	60	10.3
Inlet Nozzle 2 (Heat # 9-5078)	60	11.6
Inlet Nozzle 3 (Heat # 9-4819)	60	-47.2
Outlet Nozzle 1 (Heat # 9-4825-1)	60	-44.9
Outlet Nozzle 2 (Heat # 9-4762)	60	-87.5
Outlet Nozzle 3 (Heat # 9-4788)	60	-50.2
Inlet Nozzle to Upper Shell Welds (Heat # 299L44)	- - -	-7.0
Inlet Nozzle to Upper Shell Welds (Heat # 8T1762)	- - -	-4.9
Outlet Nozzle to Upper Shell Welds (Heat # 8T1762)	- - -	-4.9
Outlet Nozzle to Upper Shell Welds (Heat # 8T1554B)	- - -	-4.9
Upper Shell Forging 122V109VA1	40	40
Upper to Intermediate Shell Circumferential Weld (Heat # 25017)	0	0
Intermediate Shell Plate C4326-1	10	10
Intermediate Shell Plate C4326-2	0	11.4
Intermediate Shell Longitudinal Welds L3 and L4 (Heat # 8T1554)	-48.6	-48.6
Intermediate to Lower Shell Circumferential Weld (Heat # 72445)	-72.5	-72.5
Lower Shell Plate C4415-1	20	20
Lower Shell Plate C4415-2	0	4.6
Lower Shell Longitudinal Weld L1 (Heat # 8T1554)	-48.6	-48.6
Lower Shell Longitudinal Weld L2 (Heat # 299L44)	-74.3	-74.3

Notes:

(a) The previous initial RT_{NDT} values were taken from the Surry Power Station UFSAR, Table 4.1-14 [Ref. J-3]. These values are consistent with those documented in Dominion Calculation SM-1008, Addendum 00M [Ref. J-2]; however, some initial RT_{NDT} values are only listed in the UFSAR.

(b) Current initial RT_{NDT} values correspond to the values utilized herein. In some cases, these values have been updated or defined based on evaluations completed in PWROG-16045-NP [Ref. J-1].

Table J-2 Comparison of Previous and Current Initial RT_{NDT} Values for Surry Unit 2

Material Identification	Previous Initial RT_{NDT}[a] (°F)	Current Initial RT_{NDT}[b] (°F)
Replacement Reactor Vessel Closure Head 02W1-1-1-1	-60	-60
Reactor Vessel Flange FV-2542	-65	-156.3
Inlet Nozzle 1 (Heat # 9-5104)	60	-29.7
Inlet Nozzle 2 (Heat # 9-4815)	60	4.5
Inlet Nozzle 3 (Heat # 9-5205)	60	6.5
Outlet Nozzle 1 (Heat # 9-4825-2)	60	-58.1
Outlet Nozzle 2 (Heat # 9-5086-1)	60	-26.6
Outlet Nozzle 3 (Heat # 9-5086-2)	60	-33.8
Inlet Nozzle to Upper Shell Welds (Heat # 8T1762)	- - -	-4.9
Outlet Nozzle to Upper Shell Welds (Rotterdam)	- - -	30
Upper Shell Forging 123V303VA1	30	30
Upper to Intermediate Shell Circumferential Weld (Heat # 4275)	0	0
Intermediate Shell Plate C4331-2	-10	15.0
Intermediate Shell Plate C4339-2	-20	7.8
Intermediate Shell Longitudinal Welds L3 and L4 (OD 50%) (Heat # 72445)	-72.5	-72.5
Intermediate Shell Longitudinal Weld L4 (ID 50%) (Heat # 8T1762)	-48.6	-48.6
Intermediate to Lower Shell Circumferential Weld (Heat # 0227)	0	0
Lower Shell Plate C4208-2	-30	-30
Lower Shell Plate C4339-1	-10	-4.4
Lower Shell Longitudinal Weld L1 and L2 (Heat # 8T1762)	-48.6	-48.6

Notes:

(a) The previous initial RT_{NDT} values were taken from the Surry Power Station UFSAR, Table 4.1-15 [Ref. J-3]. These values are consistent with those documented in Dominion Calculation SM-1008, Addendum 00M [Ref. J-2]; however, some initial RT_{NDT} values are only listed in the UFSAR.

(b) Current initial RT_{NDT} values correspond to the values utilized herein. In some cases, these values have been updated or defined based on evaluations completed in PWROG-16045-NP [Ref. J-1] and Appendix E (for weld Heat # 0227).

Table J-3 Comparison of Previous and Current σ_I Values for Surry Unit 1

Material Identification	Previous σ_I [(a)] (°F)	Current σ_I [(b)] (°F)
Inlet Nozzle 1 (Heat # 9-4787)	---	0
Inlet Nozzle 2 (Heat # 9-5078)	---	0
Inlet Nozzle 3 (Heat # 9-4819)	---	0
Outlet Nozzle 1 (Heat # 9-4825-1)	---	0
Outlet Nozzle 2 (Heat # 9-4762)	---	0
Outlet Nozzle 3 (Heat # 9-4788)	---	0
Inlet Nozzle to Upper Shell Welds (Heat # 299L44)	---	20.6
Inlet Nozzle to Upper Shell Welds (Heat # 8T1762)	---	19.7
Outlet Nozzle to Upper Shell Welds (Heat # 8T1762)	---	19.7
Outlet Nozzle to Upper Shell Welds (Heat # 8T1554B)	---	19.7
Upper Shell Forging 122V109VA1	0	0
Upper to Intermediate Shell Circumferential Weld (Heat # 25017)	20.0	20.0
Intermediate Shell Plate C4326-1	0	0
Intermediate Shell Plate C4326-2	0	0
Intermediate Shell Longitudinal Welds L3 and L4 (Heat # 8T1554)	18.0	18.0
Intermediate to Lower Shell Circumferential Weld (Heat # 72445)	12.0	12.0
Lower Shell Plate C4415-1	0	0
Lower Shell Plate C4415-2	0	0
Lower Shell Longitudinal Weld L1 (Heat # 8T1554)	18.0	18.0
Lower Shell Longitudinal Weld L2 (Heat # 299L44)	12.8	12.8

Notes:

(a) The previous σ_I values were taken from Dominion Calculation SM-1008, Addendum 00M [Ref. J-2].

(b) Current σ_I values correspond to the values utilized herein. In some cases, these values have been confirmed or defined based on evaluations completed in PWROG-16045-NP [Ref. J-1]. σ_I is set equal to 0 when measured data is used per WCAP-14040-A, Revision 4 [Ref. J-4].

Table J-4 Comparison of Previous and Current σ_I Values for Surry Unit 2

Material Identification	Previous σ_I[(a)] (°F)	Current σ_I[(b)] (°F)
Inlet Nozzle 1 (Heat # 9-5104)	- - -	0
Inlet Nozzle 2 (Heat # 9-4815)	- - -	0
Inlet Nozzle 3 (Heat # 9-5205)	- - -	0
Outlet Nozzle 1 (Heat # 9-4825-2)	- - -	0
Outlet Nozzle 2 (Heat # 9-5086-1)	- - -	0
Outlet Nozzle 3 (Heat # 9-5086-2)	- - -	0
Inlet Nozzle to Upper Shell Welds (Heat # 8T1762)	- - -	19.7
Outlet Nozzle to Upper Shell Welds (Rotterdam)	- - -	0
Upper Shell Forging 123V303VA1	0	0
Upper to Intermediate Shell Circumferential Weld (Heat # 4275)	20.0	20.0
Intermediate Shell Plate C4331-2	0	0
Intermediate Shell Plate C4339-2	0	0
Intermediate Shell Longitudinal Welds L3 and L4 (OD 50%) (Heat # 72445)	12.0	12.0
Intermediate Shell Longitudinal Weld L4 (ID 50%) (Heat # 8T1762)	18.0	18.0
Intermediate to Lower Shell Circumferential Weld (Heat # 0227)	20.0	0
Lower Shell Plate C4208-2	0	0
Lower Shell Plate C4339-1	0	0
Lower Shell Longitudinal Weld L1 and L2 (Heat # 8T1762)	18.0	18.0

Notes:

(a) The previous σ_I values were taken from Dominion Calculation SM-1008, Addendum 00M [Ref. J-2].

(b) Current σ_I values correspond to the values utilized herein. In some cases, these values have been confirmed or defined based on evaluations completed in PWROG-16045-NP [Ref. J-1] and Appendix E (for weld Heat # 0227). σ_I is set equal to 0 when measured data is used per WCAP-14040-A, Revision 4 [Ref. J-4].

Table J-5 Comparison of Previous and Current σ_Δ Values for Surry Unit 1

Material Identification	Previous σ_Δ [(a)] (°F)	Current σ_Δ [(b)] (°F)
Inlet Nozzle 1 (Heat # 9-4787)	---	17.0
Inlet Nozzle 2 (Heat # 9-5078)	---	17.0
Inlet Nozzle 3 (Heat # 9-4819)	---	17.0
Outlet Nozzle 1 (Heat # 9-4825-1)	---	17.0
Outlet Nozzle 2 (Heat # 9-4762)	---	17.0
Outlet Nozzle 3 (Heat # 9-4788)	---	17.0
Inlet Nozzle to Upper Shell Welds (Heat # 299L44)	---	14.0/28.0[(c)]
Inlet Nozzle to Upper Shell Welds (Heat # 8T1762)	---	28.0
Outlet Nozzle to Upper Shell Welds (Heat # 8T1762)	---	28.0
Outlet Nozzle to Upper Shell Welds (Heat # 8T1554B)	---	28.0
Upper Shell Forging 122V109VA1	17.0	17.0
Upper to Intermediate Shell Circumferential Weld (Heat # 25017)	28.0	28.0
Intermediate Shell Plate C4326-1	17.0	17.0
Intermediate Shell Plate C4326-2	17.0	17.0
Intermediate Shell Longitudinal Welds L3 and L4 (Heat # 8T1554)	28.0	28.0
Intermediate to Lower Shell Circumferential Weld (Heat # 72445)	28.0	28.0
Lower Shell Plate C4415-1	8.5	8.5/17.0[(d)]
Lower Shell Plate C4415-2	17.0	8.5/17.0[(d)]
Lower Shell Longitudinal Weld L1 (Heat # 8T1554)	28.0	28.0
Lower Shell Longitudinal Weld L2 (Heat # 299L44)	28.0	28.0

Notes:

(a) The previous σ_Δ values were taken from Dominion Calculation SM-1008, Addendum 00M [Ref. J-2].

(b) Current σ_Δ values correspond to the values utilized herein; however, values reported in this table do not consider that σ_Δ need not exceed $0.5*\Delta RT_{NDT}$ per Regulatory Guide 1.99, Revision 2 [Ref. J-5]. See Section 5 and Appendix B for the actual σ_Δ values utilized in cases where $0.5*\Delta RT_{NDT}$ was limiting.

(c) For Regulatory Guide 1.99, Revision 2 [Ref. J-5] Position 2.1, 14.0°F was utilized as a result of credible surveillance data. For Regulatory Guide 1.99, Revision 2 [Ref. J-5] Position 1.1, 28.0°F was utilized

(d) For Regulatory Guide 1.99, Revision 2 [Ref. J-5] Position 2.1, 8.5°F was utilized as a result of credible surveillance data. For Regulatory Guide 1.99, Revision 2 [Ref. J-5] Position 1.1, 17.0°F was utilized.

Table J-6 Comparison of Previous and Current σ_Δ Values for Surry Unit 2

Material Identification	Previous σ_Δ [a] (°F)	Current σ_Δ [b] (°F)
Inlet Nozzle 1 (Heat # 9-5104)	---	17.0
Inlet Nozzle 2 (Heat # 9-4815)	---	17.0
Inlet Nozzle 3 (Heat # 9-5205)	---	17.0
Outlet Nozzle 1 (Heat # 9-4825-2)	---	17.0
Outlet Nozzle 2 (Heat # 9-5086-1)	---	17.0
Outlet Nozzle 3 (Heat # 9-5086-2)	---	17.0
Inlet Nozzle to Upper Shell Welds (Heat # 8T1762)	---	28.0
Outlet Nozzle to Upper Shell Welds (Rotterdam)	---	28.0
Upper Shell Forging 123V303VA1	17.0	17.0
Upper to Intermediate Shell Circumferential Weld (Heat # 4275)	28.0	28.0
Intermediate Shell Plate C4331-2	17.0	17.0
Intermediate Shell Plate C4339-2	17.0	17.0
Intermediate Shell Longitudinal Welds L3 and L4 (OD 50%) (Heat # 72445)	28.0	28.0
Intermediate Shell Longitudinal Weld L4 (ID 50%) (Heat # 8T1762)	28.0	28.0
Intermediate to Lower Shell Circumferential Weld (Heat # 0227)	14.0	14.0/28.0[c]
Lower Shell Plate C4208-2	17.0	17.0
Lower Shell Plate C4339-1	17.0	17.0
Lower Shell Longitudinal Weld L1 and L2 (Heat # 8T1762)	28.0	28.0

Notes:

(a) The previous σ_Δ values were taken from Dominion Calculation SM-1008, Addendum 00M [Ref. J-2].

(b) Current σ_Δ values correspond to the values utilized herein; however, values reported in this table do not consider that σ_Δ need not exceed $0.5*\Delta RT_{NDT}$ per Regulatory Guide 1.99, Revision 2 [Ref. J-5]. See Section 5 and Appendix B for the actual σ_Δ values utilized in cases where $0.5*\Delta RT_{NDT}$ was limiting.

(c) For Regulatory Guide 1.99, Revision 2 [Ref. J-5] Position 2.1, 14.0°F was utilized as a result of credible surveillance data. For Regulatory Guide 1.99, Revision 2 [Ref. J-5] Position 1.1, 28.0°F was utilized.

Table J-7 Comparison of Previous and Current Unirradiated USE Values for Surry Unit 1

Material Identification	Previous Unirradiated USE[(a)] (ft-lb)	Current Unirradiated USE[(b)] (ft-lb)
Inlet Nozzle 1 (Heat # 9-4787)	64	63
Inlet Nozzle 2 (Heat # 9-5078)	64	64
Inlet Nozzle 3 (Heat # 9-4819)	68	68
Outlet Nozzle 1 (Heat # 9-4825-1)	68	68
Outlet Nozzle 2 (Heat # 9-4762)	85	82
Outlet Nozzle 3 (Heat # 9-4788)	72	71
Inlet Nozzle to Upper Shell Welds (Heat # 299L44)	---	64
Inlet Nozzle to Upper Shell Welds (Heat # 8T1762)	---	64
Outlet Nozzle to Upper Shell Welds (Heat # 8T1762)	---	64
Outlet Nozzle to Upper Shell Welds (Heat # 8T1554B)	---	64
Upper Shell Forging 122V109VA1	83	114
Upper to Intermediate Shell Circumferential Weld (Heat # 25017)	EMA	$\geq$64
Intermediate Shell Plate C4326-1	115	115
Intermediate Shell Plate C4326-2	94	94
Intermediate Shell Longitudinal Welds L3 and L4 (Heat # 8T1554)	77/EMA	64
Intermediate to Lower Shell Circumferential Weld (Heat # 72445)	77/EMA	64
Lower Shell Plate C4415-1	103	103
Lower Shell Plate C4415-2	83	82
Lower Shell Longitudinal Weld L1 (Heat # 8T1554)	77/EMA	64
Lower Shell Longitudinal Weld L2 (Heat # 299L44)	70/EMA	64

Notes:

(a) The previous unirradiated USE values were taken from the Surry Power Station UFSAR, Table 4.1-14 [Ref. J-3]. These values are consistent with those documented in Dominion Calculation SM-1008, Addendum 00M [Ref. J-2]; however, some initial USE values are only listed in the UFSAR.

(b) Current unirradiated USE values correspond to the values utilized herein. In some cases, these values have been updated or defined based on evaluations completed in PWROG-16045-NP [Ref. J-1]. The current unirradiated USE values for materials previously designated with "EMA" have been updated utilizing a generic value or weld qualification data as described in Table 3-1.

Table J-8 Comparison of Previous and Current Unirradiated USE Values for Surry Unit 2

Material Identification	Previous Unirradiated USE[(a)] (ft-lb)	Current Unirradiated USE[(b)] (ft-lb)
Inlet Nozzle 1 (Heat # 9-5104)	73	73
Inlet Nozzle 2 (Heat # 9-4815)	66	66
Inlet Nozzle 3 (Heat # 9-5205)	66	67
Outlet Nozzle 1 (Heat # 9-4825-2)	74	73
Outlet Nozzle 2 (Heat # 9-5086-1)	79	77
Outlet Nozzle 3 (Heat # 9-5086-2)	73	71
Inlet Nozzle to Upper Shell Welds (Heat # 8T1762)	---	64
Outlet Nozzle to Upper Shell Welds (Rotterdam)	---	71
Upper Shell Forging 123V303VA1	104	104
Upper to Intermediate Shell Circumferential Weld (Heat # 4275)	EMA	$\geq$68
Intermediate Shell Plate C4331-2	84	84
Intermediate Shell Plate C4339-2	83	83
Intermediate Shell Longitudinal Welds L3 and L4 (OD 50%) (Heat # 72445)	77/EMA	64
Intermediate Shell Longitudinal Weld L4 (ID 50%) (Heat # 8T1762)	EMA	64
Intermediate to Lower Shell Circumferential Weld (Heat # 0227)	90/EMA	82
Lower Shell Plate C4208-2	94	94
Lower Shell Plate C4339-1	105	101
Lower Shell Longitudinal Weld L1 and L2 (Heat # 8T1762)	EMA	64

Notes:

(a) The previous unirradiated USE values were taken from the Surry Power Station UFSAR, Table 4.1-15 [Ref. J-3]. These values are consistent with those documented in Dominion Calculation SM-1008, Addendum 00M [Ref. J-2]; however, some initial USE values are only listed in the UFSAR or SM-1008, Addendum 00M.

(b) Current unirradiated USE values correspond to the values utilized herein. In some cases, these values have been updated or defined based on evaluations completed in PWROG-16045-NP [Ref. J-1] and Appendix E (for weld Heat # 0227). The current unirradiated USE values for materials previously designated with "EMA" have been updated utilizing a generic value, weld qualification data, or data in Appendix E as described in Table 3-3.

J.1 REFERENCES

J-1 Pressurized Water Reactor Owners Group (PWROG) Report PWROG-16045-NP, Revision 0, "Determination of Unirradiated RT_{NDT} and Upper-Shelf Energy Values of the Surry Units 1 and 2 Reactor Vessel Materials," March 2017.

J-2 Dominion Calculation SM-1008, Revision 0, Addendum 00M, "Reactor Vessel Integrity Calculations Supporting a Technical Specifications Change Request (TSCR) to Update the Burnup Applicability Limit for RCS Pressure/Temperature Limits, LTOPS Setpoint, and LTOPS Enabling Temperature at Surry Power Station Units 1 and 2," January 2010.

J-3 Surry Power Station Updated Final Safety Analysis Report, Revision 48, September 2016.

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

J-5 U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.