VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

May 6, 2010

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

Serial No.	10-199
NL&OS/GDM	R1
Docket Nos.	50-280, 50-281
License Nos.	DPR-32, DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)
SURRY POWER STATION UNITS 1 AND 2
LICENSE AMENDMENT REQUEST
REVISED CUMULATIVE CORE BURNUP APPLICABILITY LIMIT FOR HEATUP AND COOLDOWN CURVES, LOW TEMPERATURE OVERPRESSURE PROTECTION SYSTEM (LTOPS) SETPOINT, AND LTOPS ENABLE TEMPERATURE

Dominion proposes a change to the Surry Power Station (Surry) Units 1 and 2 Technical Specifications (TS) to update the cumulative core burnup applicability limit (Effective Full Power Years; EFPY) for Reactor Coolant System (RCS) Heatup and Cooldown Pressure/Temperature (P/T) Limits, Low Temperature Overpressure Protection System (LTOPS) Setpoint, and LTOPS Enabling Temperature (T-enable). The evaluations described herein support extension of the cumulative core burnup applicability limit from 28.8 EFPY and 29.4 EFPY for Surry Units 1 and 2, respectively, to 48 EFPY for both units. Supporting calculations utilize the revised initial material properties and margins of NRC-approved Topical Report BAW-2308 Revision 2-A, "Initial RT_{NDT} of Linde 80 Weld Materials." The revised initial Reference Temperature Nil Ductility Transition (RT_{NDT}) values and initial margin terms for Linde 80 weld materials in BAW-2308 Revision 2-A differ slightly from those in the previously implemented NRC-approved BAW-2308 Revision 1-A, as BAW-2308 Revision 2-A incorporates a loading rate correction term not included in BAW-2308 Revision 1-A.

The evaluations discussed in Attachment 1 affirm the conservatism of existing RCS P/T limits, LTOPS setpoint, LTOPS T-enable value, and Charpy Upper Shelf Energy (CvUSE) values; provide revised 10 CFR 50.61 Pressurized Thermal Shock (PTS) screening calculation results; affirm continued compliance with 10 CFR 50 Appendix H, which governs reactor vessel materials surveillance program requirements; and update the NRC's Reactor Vessel Integrity Database (RVID).

A request for exemption from the requirements of 10 CFR 50 Appendix G and 10 CFR 50.61 was previously submitted by letter dated June 13, 2006 (Serial No. 06-434) and approved by letter dated June 27, 2007, to permit utilization of the methodology of Framatome ANP Topical Report BAW-2308, Rev. 1-A to establish revised initial (unirradiated) RT_{NDT} values and margins at Surry Units 1 and 2. As discussed herein, no additional exemption request is required to apply the more recently NRC-approved version of Topical Report BAW-2308 Revision 2-A.

Existing Surry Units 1 and 2 RCS P/T limit curves, LTOPS setpoints, and T-enable are valid to cumulative core burnups of 28.8 and 29.4 EFPY, respectively. Based on current operating cycle projections, it is estimated that a cumulative core burnup of 28.8 EFPY for Surry Unit 1 will be reached in approximately June 2011. A cumulative core burnup of 29.4 EFPY for Surry

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Unit 2 will be reached in approximately March 2012. Therefore, we respectfully request NRC review and approval of the proposed change presented herein by May 13, 2011. The existing Surry Units 1 and 2 RCS P/T limits, LTOPS setpoints, and T-enable will remain valid and conservative until approval of the proposed revised cumulative core burnup applicability limit. Attachment 1 presents a discussion of the changes to reactor vessel materials evaluations for Surry Units 1 and 2, as well as the basis for the proposed TS change. The marked-up and typed proposed TS pages are provided in Attachments 2 and 3, respectively. An update to the NRC Reactor Vessel Integrity Database (RVID) for Surry is provided in Attachment 4. TS Basis changes reflecting the proposed change have also been included for the NRC's information.

We have evaluated the proposed amendment and have determined that it does not involve a significant hazards consideration as defined in 10 CFR 50.92. The basis for this determination is included in Attachment 1. We have also determined that operation with the proposed change will not result in any significant increase in the amount of effluents that may be released offsite or any significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed amendment is eligible for categorical exclusion from an environmental assessment as set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment is needed in connection with the approval of the proposed change. The proposed TS change has been reviewed and approved by the Facility Safety Review Committee.

If you have any questions or require additional information regarding this submittal, please contact Mr. Gary D. Miller at (804) 273-2771.

Sincerely,

J. A. Price
Vice President – Nuclear Engineering

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by J. Alan Price, who is Vice President – Nuclear Engineering, of Virginia Electric and Power Company. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 6th day of May, 2010.

My Commission Expires: 4/30/13.

Ginger L Melton
Notary Public

GINGER LYNN MELTON
Notary Public
Commonwealth of Virginia
310847
My Commission Expires Apr 30, 2013

I was commissioned a notary public as Ginger L. Alligood.

Attachments:
1. Discussion of Change
2. Proposed Technical Specifications Pages (Mark-Up)
3. Proposed Technical Specifications Pages (Typed)
4. Reactor Vessel Integrity Database Update

Commitment made in this letter:

1. Sections of the Surry Units 1 and 2 UFSAR will be revised to reflect implementation of the revised design basis analyses described herein. Following NRC approval of the License Amendment Request associated with this submittal, a UFSAR revision will be made in accordance with the requirements of 10 CFR 50.71(e).

cc: U.S. Nuclear Regulatory Commission
Region II
245 Peachtree Center Avenue NE
Suite 1200
Atlanta, Georgia 30303-1257

NRC Senior Resident Inspector
Surry Power Station

State Health Commissioner
Virginia Department of Health
James Madison Building – 7th Floor
109 Governor Street
Suite 730
Richmond, Virginia 23219

Ms. K. R. Cotton
NRC Project Manager
U. S. Nuclear Regulatory Commission
One White Flint North
Mail Stop 08 G-9A
11555 Rockville Pike
Rockville, Maryland 20852-2738

Dr. V. Sreenivas
NRC Project Manager
U. S. Nuclear Regulatory Commission
One White Flint North
Mail Stop 08 G-9A
11555 Rockville Pike
Rockville, Maryland 20852-2738

Attachment 1

DISCUSSION OF CHANGE

**Virginia Electric and Power Company
(Dominion)
Surry Power Station Units 1 and 2**

DISCUSSION OF CHANGE

1.0 Introduction

Dominion proposes changes to the Surry Power Station (Surry) Units 1 and 2 Technical Specifications (TS) to update the cumulative core burnup applicability limit (Effective Full Power Years; EFPY) for Reactor Coolant System (RCS) Heatup and Cooldown Pressure/Temperature (P/T) Limits. The cumulative core burnup applicability limit is also updated for the Low Temperature Overpressure Protection System (LTOPS) Setpoint, and LTOPS Enabling Temperature (T-enable). The evaluations described herein support extension of the cumulative core burnup applicability limit from 28.8 EFPY and 29.4 EFPY for Surry Units 1 and 2, respectively, to 48 EFPY for both units. Supporting calculations utilize the revised initial material properties and margins of NRC-approved Topical Report BAW-2308 Revision 2-A [11]. BAW-2308 Revision 2 was approved by NRC in Reference [2]. The revised initial RT_{NDT} values and initial margin terms for Linde 80 weld materials in BAW-2308 Revision 2-A [11] differ slightly from those in the previously implemented NRC-approved BAW-2308 Revision 1-A [1], as BAW-2308 Revision 2-A [11] incorporates a loading rate correction term not included in BAW-2308 Revision 1-A [1].

The discussion presented herein affirms the conservatism of existing RCS P/T Limits, LTOPS Setpoint, LTOPS T-enable value, and Charpy Upper Shelf Energy (CvUSE) values for Surry Units 1 and 2 cumulative core burnups up to 48 EFPY. Revised 10 CFR 50.61 Pressurized Thermal Shock (PTS) screening calculations are presented utilizing BAW-2308 Revision 2-A [11]. A discussion of ongoing compliance with 10 CFR 50 Appendix H, which governs reactor vessel materials surveillance program requirements, is also provided. An update to the NRC Reactor Vessel Integrity Database (RVID) for Surry is provided in Attachment 4, and the marked-up and typed proposed TS pages are provided in Attachments 2 and 3, respectively.

2.0 Background

By letter dated June 13, 2006 [3], Dominion submitted an update to the NRC Reactor Vessel Integrity Database (RVID), and requested exemption from the requirements of 10 CFR 50 Appendix G and 10 CFR 50.61 to permit utilization of the methodology of Framatome ANP Topical Report BAW-2308, Rev. 1-A [1] to establish revised initial (unirradiated) RT_{NDT} values and margins at Surry Units 1 and 2. The submittal was made in anticipation of utilizing the analytical margins provided by the revised initial RT_{NDT} values to establish revised plant operating limitations valid for the 60-year renewed license period. The requested exemption was granted by letter dated June 27, 2007 [4]. As demonstrated by Page 8 of Reference [4], the NRC intended for BAW-2308 to be a living, updated document. Specifically, Page 8 of Reference [4], which establishes the conditions under which the exemption applies, states:

> The exemptions are granted for the licensee to utilize the most recent staff approved version of BAW-2308 (currently BAW-2308, Revision 1). Future revisions of BAW-2308 could affect fracture toughness data and analyses for Surry 1 and 2. Therefore, the licensee must review any future staff-approved revisions of BAW-2308 and update the units' fracture toughness assessments, based on the information in any staff-approved revision of BAW-2308.

Based on the conditions of the NRC SER [4], it is concluded that Dominion must consider and apply the results of the NRC-approved BAW-2308 Revision 2 [11] in licensing applications, but that a revised exemption request is not required.

Although Topical Report BAW-2308, Rev. 2-A [11] has been approved for use, elements of BAW-2308 Rev. 1-A [1] remain in effect. Specifically, the NRC's approval of BAW-2308, Rev. 1 requires the use of a minimum chemistry factor of 167°F and an uncertainty of σ_I = 28 °F for the Linde 80 weld heat materials, when the irradiation shift model is employed per NRC Regulatory Guide 1.99, Rev. 2 [12].

3.0 Discussion of Changes

3.1 RPV Neutron Fluence Projections

The reactor vessel beltline fluence values used for material properties calculations are unchanged from those utilized in the Reference [3] RVID update and exemption request submittal approved by the NRC in Reference [4]. The reactor vessel fast neutron fluence calculations were performed by Framatome ANP, and are applicable to the license renewal period (60-year operation) at Surry Power Station. The fluence calculations assumed removal of Flux Suppression Inserts (FSIs) from peripheral locations in Surry Unit 1, use of Integral Fuel Burnable Absorber (IFBA), rather than discreet burnable poison rods, and a 95% capacity factor. The fast neutron fluence predictions were prepared in accordance with the requirements of US NRC Regulatory Guide 1.190 [10], as described in Framatome ANP Topical Report BAW-2241-P-A [9]. BAW-2241 was approved by NRC in Reference [5].

More recently, Westinghouse performed revised reactor vessel fast neutron fluence analyses that considered planned implementation of a Measurement Uncertainty Recapture (MUR) core power uprating at Surry Units 1 and 2 [18]. The acceptance criteria for a Light Water Reactor (LWR) exposure evaluation is derived from Regulatory Guide 1.190, Section 1.4.3 [10]. This section of the regulatory guide states that "a vessel fluence uncertainty of 20% (1σ) is acceptable for RT_{PTS} and RT_{NDT} determination." The NRC approved methodology used for the Surry Units 1 and 2 fluence evaluations has been demonstrated to satisfy this criterion. The particular requirements of the regulatory guide that are incorporated in this methodology are as follows:

1. The calculations use neutron transport cross-sections from the latest version of the Evaluated Nuclear Data File (ENDF/B-VI).

2. A P5 expansion of the scattering cross-sections is used in the discrete ordinates calculations. This exceeds the minimum requirement of Regulatory Guide 1.190.

3. An S16 order of angular quadrature is used in the discrete ordinates calculations. This exceeds the minimum requirement of Regulatory Guide 1.190.

4. An uncertainty analysis that includes comparisons of calculations with test and power reactor benchmarks and an analytical uncertainty study has been completed and documented in NRC approved topical reports [6 and 7]. The overall uncertainty in the transport calculations was demonstrated to be 13% (1σ). This level of uncertainty meets the 20% (1σ) requirement of Regulatory Guide 1.190.

The reactor vessel fast neutron fluence values utilized in the Reference [3] RVID update and exemption request submittal approved by the NRC in Reference [4] conservatively bound those calculated by Westinghouse, even after consideration of the MUR uprate. Therefore, the more conservative Framatome ANP fluence analysis results continue to be used herein.

For convenience, the previously submitted [3] fluence values prepared by Framatome ANP are presented in Tables 3.1 and 3.2 for Surry Units 1 and 2, respectively, below.

Table 3.1
RPV Fluence Projections for Surry Unit 1

Surry Unit 1		**Neutron Fluence (E > 1.0 MeV)**	
Location	**Material**	**Fluence at 32 EFPY (n/cm^2)**	**Fluence at 48 EFPY (n/cm^2)**
Vessel Wall Inner Surface (0°)	Intermediate and Lower Plates	3.80E+19	5.66E+19
Lower Shell Longitudinal Weld, L1 & L2	SA-1494/8T1554 SA-1526/299L44	6.40E+18	1.04E+19
Intermediate Shell Longitudinal Weld, L3 & L4	SA-1494/8T1554	6.78E+18	1.08E+19
Intermediate to Lower Shell Circumferential Weld, W05	SA-1585/72445 SA-1650/72445	3.74E+19	5.61E+19
Nozzle to Intermediate Shell Circumferential Weld, W06	J726/25017	5.27E+18	7.75E+18

Table 3.2
RPV Fluence Projections for Surry Unit 2

Surry Unit 2		**Neutron Fluence (E > 1.0 MeV)**	
Location	**Material**	**Fluence at 32 EFPY (n/cm^2)**	**Fluence at 48 EFPY (n/cm^2)**
Vessel Wall Inner Surface (0°)	Intermediate and Lower Plates	3.64E+19	5.38E+19
Lower Shell Longitudinal Weld, L1 & L2	WF-4/8T1762 WF-8/8T1762	7.62E+18	1.14E+19
Intermediate Shell Longitudinal Weld, L3 & L4	SA-1585/72445 WF-4/8T1762	7.63E+18	1.14E+19
Intermediate to Lower Shell Circumferential Weld, W05	R3008/0227	3.62E+19	5.37E+19
Nozzle to Intermediate Shell Circumferential Weld, W06	J737/4275	4.00E+18	6.32E+18

3.2 Linde 80 Weld Material Properties

Material properties calculations utilized the revised initial material properties and margins of NRC-approved Topical Report BAW-2308, Rev. 2-A [11]. BAW-2308 Rev. 2 was approved by the NRC in Reference [2]. The revised initial RT_{NDT} values and initial margin terms for Linde 80 weld materials in BAW-2308 Revision 2-A [11] differ slightly from those in the previously implemented NRC-approved BAW-2308 Rev. 1-A [1], as BAW-2308 Revision 2-A [11] incorporates a loading rate correction term not included in BAW-2308 Revision 1-A [1].

Table 9 of the Reference [11] contains the revised initial RT_{T0} and initial margin (σ_I) values for Linde 80 weld materials that are approved by NRC staff for the purpose of RPV material property determination. The approved values from Reference [11] are shown in Table 3.3 below.

Table 3.3
NRC Staff-Accepted Initial RT_{T0} and σ_I Values for Linde 80 Weld Materials

Linde 80 Weld Wire Heat	Initial RT_{T0} (°F)	Initial Margin σ_I (°F)
406L44	-98.0	11.6
71249	-53.5	12.8
72105	-31.1	13.7
821T44	-84.2	9.6
299L44 (Surry Unit 1)	-74.3	12.8
72442	-33.2	12.2
72445 (Surry Units 1 &2)	-72.5	12.0
61782	-58.5	15.4
All Heats (Generic Value) (Surry Units 1 & 2)	-48.6	18.0

The following Linde 80 weld materials are contained in the Surry Unit 1 reactor vessel: 8T1554, 299L44, and 72445.

The following Linde 80 weld materials are contained in the Surry Unit 2 reactor vessel: 8T1762 and 72445.

Note that for any Linde 80 material not given specifically in the table above, the inputs for "All Heats (Generic Value)" are used.

A summary of the material property data relevant to this submittal is included as Attachment 4, which contains data that can be used to update the NRC's Reactor Vessel Integrity Database (RVID).

3.3 Pressurized Thermal Shock Screening Calculations

For Surry Unit 1, the limiting materials in terms of absolute value of RT_{PTS} are Intermediate to Lower Shell Circumferential Welds SA-1585/72445 and SA-1650/72445. For these materials, the value of RT_{PTS} is 226.3 °F versus the PTS screening criterion of 300 °F for circumferential welds. For these materials, there is 73.7 °F margin to the applicable PTS screening criterion.

For Surry Unit 1, the limiting material in terms of margin to the applicable PTS screening criterion is Lower Shell Longitudinal Weld SA-1526/299L44. For this material, the value of RT_{PTS} is 210.3°F versus the PTS screening criterion of 270 °F for plates, forgings, and axial welds. For this material, there is 59.7 °F margin to the applicable PTS screening criterion.

For Surry Unit 2, the limiting material in terms of absolute value of RT_{PTS} is Intermediate to Lower Shell Circumferential Weld R3008/0227. For this material, the value of RT_{PTS} is 236.4 °F versus the

PTS screening criterion of 300 °F for circumferential welds. For this material, there is 63.6 °F margin to the applicable PTS screening criterion. This material is also limiting in terms of margin to the applicable PTS screening criterion.

In summary, when 60-year fluence projections are considered in conjunction with the revised Linde 80 weld material properties per BAW-2308 Revision 2-A, the Surry Units 1 and 2 reactor vessel beltline materials meet the 10 CFR 50.61 PTS screening criteria at the end of the current 60-year operating license period.

3.4 10 CFR 50 Appendix G Charpy Upper Shelf Energy Calculations

Charpy Upper Shelf Energy (CvUSE) calculations for Surry Units 1 and 2 are unchanged from those previously provided in Reference [3]. Percentage drops in CvUSE values at the 1/4-T location within the reactor vessel wall were calculated using the Regulatory Guide 1.99 Rev. 2 Position 1.2 methodology. Equivalent Margin Analyses (EMAs) are required for the Surry Units 1 and 2 reactor vessel beltline materials for which either (1) initial, unirradiated USE values were not known, or (2) initial unirradiated USE values were available and the beltline materials USE at the end of the licensed period of operation were projected to fall below the 50 ft-lb criterion specified in Section IV.A.1 of 10 CFR Part 50, Appendix G. For those weld materials which meet either of these criteria, the attached RVID Update "50-Year CvUSE Summary" table displays a value of "EMA" in place of the calculated values for USE at 1/4-T, Unirradiated USE, % Drop in USE at 1/4-T, and % Drop in USE Method.

It is noted that CvUSE values are provided in the attached RVID update summary table for Surry Unit 2 weld materials R3008 and SA-1585 even though the EMA documented in Reference [8] covers these materials. The EMA is not cited in the summary table because 10 CFR Part 50 Appendix G calls for using the EMA if the USE value does not meet the 50 ft-lb criterion. Since Surry Unit 2 weld materials R3008 and SA-1585 continue to be above 50 ft-lbs, the USE value is reported instead of the EMA.

In summary, when the 60-year fluence projections are considered, the Surry Units 1 and 2 reactor vessel beltline materials meet the 10 CFR Part 50 Appendix G requirements by satisfying the 50 ft-lb Upper Shelf Energy limit, or an Equivalent Margin Analyses has demonstrated acceptable results relative to Appendix K to Section XI of the ASME Code for projected low upper-shelf Charpy impact energy levels at 48 EFPY.

3.5 Technical Specification Pressure/Temperature Limits

The current Surry Units 1 and 2 TS RCS P/T Limits and LTOPS Setpoint are based on a limiting 1/4-thickness (1/4-T) RT_{NDT} of 228.4 °F, and a limiting 3/4-thickness (3/4-T) RT_{NDT} of 189.5 °F [15][16]. When the current P/T Limits and LTOPS Setpoint were developed, these values of RT_{NDT} were determined to bound the Surry Units 1 and 2 reactor vessel beltline materials at end-of-original 40-year license fluences corresponding to 28.8 EFPY and 29.4 EFPY for Surry Units 1 and 2, respectively [13][14].

The attached RVID update summary table shows the results for the 1/4-T RT_{NDT} and the 3/4-T RT_{NDT} values considering the 60-year fluence projections corresponding to 48 EFPY, and the Linde 80 weld material properties per Topical Report BAW-2308-2-A. At fluence projections corresponding to 48 EFPY, the limiting 1/4-T RT_{NDT} value is 222.5°F for Surry Unit 2 Intermediate to Lower Shell Circumferential Weld material R3008/0227. At fluence projections corresponding to

48 EFPY, the limiting 3/4-T RT_{NDT} value is 188.6°F for Surry Unit 2 Intermediate to Lower Shell Circumferential Weld material R3008/0227. These values are less limiting than those assumed in the development of the existing Technical Specification heatup and cooldown limit curves [13] [14]. Therefore, the existing Technical Specification heatup and cooldown limit curves remain valid and conservative for cumulative core burnups up to 48 EFPY at Surry Units 1 and 2.

3.6 Low Temperature Overpressure Protection System (LTOPS) Protection

The Surry Units 1 and 2 TS LTOPS Setpoint requirement of ≤390 psig and minimum LTOPS T-enable requirement of ≤300°F were established in References [13] and [14]. The setpoint was established to provide protection against 110% of the ASME Section XI Appendix G isothermal limit curve based on K_{1R} (lower bound curve of crack initiation and crack arrest data). T-enable was established at the limiting RT_{NDT} plus (a) the maximum temperature difference between the water and metal at the 1/4-T and 3/4-T locations during heatup or cooldown at the maximum allowable rate), (b) temperature measurement uncertainty, and (c) 50°F. Because the limiting value of RT_{NDT} remains less than that established in References [13] and [14], and other elements of the design and licensing basis are unchanged, LTOPS protection provided by the existing LTOPS PORV lift setpoint and T-enable remain valid and conservative for cumulative core burnups up to 48 EFPY at Surry Units 1 and 2.

3.7 RPV Material Surveillance Program per 10 CFR 50, Appendix H

Current reactor vessel material surveillance monitoring requirements for Surry are based on the predicted shift in Charpy V-notch 30 ft-lb energy (ΔT_{30}). The alternate methodology described in Topical Report BAW-2308, Rev. 1-A does not rely on obtaining direct fracture toughness measurements (i.e. in accordance with ASTM-1921) in the irradiated condition for the purposes of monitoring changes due to irradiation in the Linde 80 weld materials. Topical Report BAW-2308, Rev. 1-A also confirmed that the irradiation-induced shift in Charpy V-notch 30 ft-lb energy (ΔT_{30}) conservatively overpredicted the Master CurveΔT_0 test data for Linde 80 weld materials. Therefore, the current reactor vessel material surveillance program at Surry Power Station is not affected (i.e. current monitoring requirements are based on predicted shift in Charpy V-notch 30 ft-lb energy (ΔT_{30})).

Revised reactor vessel materials surveillance capsule withdrawal schedules to accommodate the 60-year license period have been developed and submitted to the NRC for review and approval in accordance with 10 CFR 50 Appendix H, Section III.B.3 [15]. The proposed schedules satisfy the requirements and guidance of ASTM E-185-82 [16] and the GALL Report (NUREG-1801) [17] for surveillance capsule withdrawal and testing. With the approval of that submittal, Surry Units 1 and 2 will remain in compliance with the requirements of 10 CFR 50 Appendix H.

4.0 Proposed Technical Specifications Change

When the revised 60-year fluence projections at 48 EFPY are used with the revised Linde 80 weld material properties, the limiting 1/4-T and 3/4-T RT_{NDT} values remain less than those assumed in the development of the existing TS P/T Limits, LTOPS Setpoint, and T-enable values for Surry Units 1 and 2. Therefore, the existing Surry Technical Specification RCS P/T Limits, LTOPS Setpoint, and T-enable value remain valid and conservative for cumulative core burnups up to 48 EFPY for Surry Units 1 and 2. On this basis, changes to the Surry Units 1 and 2 TS are proposed to extend the burnup applicability limit for P/T limits, LTOPS Setpoint, and T-enable value from the current values of

28.8 EFPY and 29.4 EFPY for Surry Units 1 and 2, respectively, to 48 EFPY for both Surry Units 1 and 2. The specific TS change is as follows:

- Modify Surry Units 1 and 2 Reactor Coolant System Heatup and Cooldown Limitations, Figures 3.1-1 and 3.1-2, respectively, to reflect a cumulative core burnup applicability limit of 48 EFPY.

Supporting TS basis changes are also being incorporated as follows:

- Modify Bases for TS 3.1.B to reflect a cumulative core burnup applicability limit of 48 EFPY for Heatup and Cooldown Limit Curves.
- Modify Bases for TS 3.1.B to reflect occurrence of a limiting RT_{NDT} of 222.5°F at the 1/4-T, 0° location of the Unit 2 intermediate-to-lower shell circumferential weld.

Based on current operating cycle projections, it is estimated that a cumulative core exposure of 28.8 EFPY for Surry Unit 1 will be reached in approximately June 2011. A cumulative core exposure of 29.4 EFPY for Surry Unit 2 will be reached in approximately March 2012. Therefore, modification of the cumulative core burnup applicability limit for Surry RCS P/T Limit curves, LTOPS Setpoint, and T-enable must be implemented prior to these dates. Prior to approval of the TS change proposed herein, the existing RCS P/T Limits, LTOPS Setpoints, and LTOPS T-enable presently in the Surry Units 1 and 2 TS continue to remain valid and conservative through their period of applicability (28.8 EFPY and 29.4 EFPY for Surry Units 1 and 2, respectively).

5.0 Determination of No Significant Hazards Consideration

Virginia Electric and Power Company (Dominion) proposes a change to the Surry Units 1 and 2 Technical Specifications (TS) to modify the Surry Units 1 and 2 Reactor Coolant System (RCS) Heatup and Cooldown Limitations, Figures 3.1-1 and 3.1-2, to reflect a cumulative core burnup applicability limit of 48 EFPY for RCS Pressure/Temperature (P/T) Limits. The cumulative core burnup limit is also increased to 48 EFPY for the Low Temperature Overpressure Protection System (LTOPS) Setpoints and LTOPS Enabling Temperature (T-enable). In accordance with the criteria set forth in 10 CFR 50.92, Dominion has evaluated the proposed TS change and determined that the change does not represent a significant hazards consideration. The following is provided in support of this conclusion:

1. <u>Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?</u>

   Response: No.

   The proposed change revises the Surry Units 1 and 2 TS RCS Heatup and Cooldown Limitations figures to reflect an increase in the cumulative core burnup applicability limit to 48 EFPY. The existing Surry TS RCS P/T Limits, LTOPS Setpoint, and T-enable value remain valid and conservative for cumulative core burnups up to 48 EFPY, thus increasing the cumulative core burnup applicability limit for RCS P/T Limits, LTOPS Setpoints and LTOPS T-enable to 48 EFPY has no bearing on the probability or consequences of an accident previously evaluated. These evaluations address the LTOPS design basis mass addition accident (inadvertent charging pump start), heat addition accident (Reactor Coolant Pump

(RCP) start with a secondary–to-primary temperature difference of 50°F) and Pressurized Thermal Shock (PTS) events, the analysis of which is covered by 10 CFR 50.61

The increased cumulative core burnup applicability limit is accomplished through application of improved analytical margins provided by Topical Report BAW-2308, Revision 2-A, "Initial RT_{NDT} of Linde 80 Weld Materials," which was approved by the NRC in March 2008 for use in plant-specific applications. Dominion assessed the effect of the use of the analytical margins and determined that the existing TS limits (RCS P/T Limits, LTOPS Setpoints and LTOPS T-enable) governing reactor vessel integrity remain valid and conservative for cumulative core burnups up to 48 EFPY. No changes to plant systems, structures or components are proposed, and no new operating modes are established. Furthermore, plant operating limits and setpoints are not being changed. Therefore, there is no increase in the probability or consequences of any accident previously evaluated.

2. <u>Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?</u>

Response: No.

No changes to plant operating conditions, operating limits or setpoints are being proposed and no changes to plant systems, structures or components are being implemented. The existing Surry TS RCS P/T Limits, LTOPS Setpoint, and T-enable value remain valid and conservative for cumulative core burnups up to 48 EFPY. Analyses supporting the increased cumulative core burnup applicability limit were performed in accordance with applicable regulatory guidance and confirm that design functions (i.e., ensuring that combined pressure and thermal stresses under normal operation heatup and cooldown conditions and under design basis accident conditions at low temperature) are maintained. Therefore, the proposed change does not create the possibility of any accident or malfunction of a different type previously evaluated.

3. <u>Does the change involve a significant reduction in the margin of safety?</u>

Response: No.

The increased cumulative core burnup applicability limit is accomplished through application of improved analytical margins provided by Topical Report BAW-2308, Revision 2-A, which was approved by the NRC in March 2008 for use in plant-specific applications. Dominion assessed the effect of the use of the analytical margins and determined that the existing TS P/T Limits, LTOPS Setpoints, and LTOPS T-enable governing reactor vessel integrity remain valid and conservative for cumulative core burnups up to 48 EFPY. No changes to plant systems, structures or components are proposed, and no new operating modes are established. Furthermore, plant operating limits and setpoints are not being changed. Consequently, the existing TS P/T Limit curves, LTOPS Setpoint, and LTOPS T-enable value provide acceptable margin to vessel fracture under both normal operation and LTOPS design basis (mass addition and heat addition) accident conditions for cumulative core burnups up to 48 EFPY. Therefore, the proposed change does not result in a significant reduction in the margin of safety.

Based on the above, Dominion concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10CFR50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

6.0 Environmental Assessment

The proposed license amendment meets the eligibility criterion for categorical exclusion set forth in 10CFR51.22(c)(10) as follows:

(i) The proposed change involves no significant hazards consideration.

As described in Section 5.0 above, the proposed change involves no significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

The proposed change does not involve the installation of any new equipment or the modification of any equipment that may affect the types or amounts of effluents that may be released offsite. In addition, the proposed change does not introduce any new modes of plant operation. Therefore, there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

(iii) There is no significant increase in individual or cumulative occupation radiation exposure.

The proposed change does not involve physical plant changes or introduce any new modes of plant operation. Therefore, there is no significant increase in individual or cumulative occupational radiation exposure.

Based on the above, Dominion concludes that, pursuant to 10CFR51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 Affected UFSAR Sections

Sections 4.1, 4.2, and 4.3 of the Surry Units 1 and 2 UFSAR will be revised to reflect implementation of the revised design basis analyses described herein. Following NRC approval of the License Amendment Request associated with this submittal, a UFSAR revision will be made in accordance with the requirements of 10 CFR 50.71(e).

8.0 Reactor Vessel Integrity Database

Attachment 4 of this submittal contains a Reactor Vessel Integrity Database update based on the alternate material properties basis for Linde 80 weld materials as provided in Topical Report BAW-2308, Revision 2-A. An update to the NRC's Reactor Vessel Integrity Database has been prepared to document the results of Dominion's most recent 10 CFR 50.61 Pressurized Thermal shock (PTS) screening calculations, Nil Ductility Transition Reference Temperature (RT_{NDT}) values, and Upper Shelf Energy (USE) values. The calculations utilize revised initial (unirradiated) RT_{NDT} values for Linde 80 weld materials based on Topical Report BAW-2308, Revision 2-A [11]. The calculations assume fluence values applicable to the current 60-year license period, as described above. 10 CFR 50.61 PTS screening criteria and 10 CFR 50 Appendix G Upper Shelf Energy criteria continue to be met for the Surry Units 1 and 2 reactor vessel materials.

9.0 Conclusions

Revised Initial RT_{NDT} Values: Calculations have been performed to determine the effect of changes in the Surry Units 1 and 2 reactor vessel integrity analyses due to revised material properties for the Linde 80 welds as described in Topical Report BAW-2308, Revision 2-A. Reactor vessel fast neutron fluence analyses applicable to the license renewal period (60-year operation), and which considered Flux Suppression Inserts removal and Integral Fuel Burnable Absorber fuel, rather than discreet burnable poison rods, continue to be utilized. The reactor vessel fluence values have been determined to conservatively represent fluence values applicable under Measurement Uncertainty Recapture uprated power conditions, as determined by Westinghouse using their approved fluence analysis methodology that complies with the requirements of RG 1.190.

RVID Update: A Reactor Vessel Integrity Database (RVID) update for Surry Units 1 and 2 is provided in Attachment 4. Shaded values represent a change from values previously provided in Reference [3].

Pressurized Thermal Shock Screening Calculations: The Surry Units 1 and 2 reactor vessel beltline materials continue to meet the 10 CFR 50.61 PTS screening criteria at the end of the current 60-year operating license period. The use of the revised Linde 80 weld material properties per Topical Report BAW-2308, Rev. 2-A does not require submission of an exemption request to the NRC, since the previously submitted exemption request remains valid and applicable to use of BAW-2308, Rev. 2-A.

Charpy Upper Shelf Energy Calculations: The Surry Units 1 and 2 reactor vessel beltline materials continue to meet the 10 CFR Part 50 Appendix G Charpy Upper Shelf Energy requirements by satisfying the 50 ft-lb limit at the end of the current 60-year operating license, or an Equivalent Margin Analyses has demonstrated acceptable results relative to Appendix K to Section XI of the ASME Code for projected low upper-shelf Charpy impact energy levels at 48 EFPY. CvUSE results are unchanged from those provided in the most recent, previous RVID update.

Proposed Technical Specifications Changes: The existing Surry Technical Specifications RCS P/T Limits, LTOPS Setpoint, and T-enable value have been demonstrated herein to remain valid and conservative for cumulative core burnups up to 48 EFPY for Surry Units 1 and 2. Therefore, changes to the Surry Units 1 and 2 Technical Specifications are proposed to extend the cumulative core burnup applicability limit for P/T Limits, as well as the LTOPS Setpoint and T-enable value, from the current values of 28.8 EFPY and 29.4 EFPY for Surry Units 1 and 2, respectively, to 48 EFPY for both Surry Units 1 and 2.

Based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by implementation of the proposed TS change, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

10.0 References

1. Framatome ANP Topical Report BAW-2308, Revision 1-A, "Initial RT_{NDT} of Linde 80 Weld Materials," approved August 2005. (See NRC Letter from H. N. Berkow to J. S. Holm (Framatome ANP), "Final Safety Evaluation for Topical Report BAW-2308, Revision 1, "Initial RT_{NDT} of Linde 80 Weld Materials" (TAC No. MB6636)," dated August 4, 2005.)

2. Letter from H. K. Nieh (USNRC) to Gordon Bischoff (PWROG), "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," (TAC No. MD4241, dated March 24, 2008

3. Letter from E. S. Grecheck (VEPCO) to USNRC, "Virginia Electric and Power Company, Surry Power Station Units 1 and 2, Update to NRC Reactor Vessel Integrity Database and Exemption Request for Alternate Material Properties Basis Per 10 CFR 50.60(b), Serial No 06-434, dated June 13, 2006.

4. Letter from Siva P. Lingam (USNRC) to David A. Christian (Dominion), "Surry Power Station, Unit Nos. 1 and 2, Exemption from the Requirements of 10 CFR Part 50, Appendix G and 10 CFR Part 60, Section 50.61 (TAC Nos. MD2337 and MD2338), Dominion Serial No. 07-0498, dated June 27, 2007.

5. NRC Letter from S. A. Richards to J. J. Kelly (B&WOG), "Acceptance for Referencing of Licensing Topical Report BAW-2241P, Revision 1, Fluence and Uncertainty Methodologies (TAC No. M98962)," dated April 5, 2000.

6. WCAP-14040-NP-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," J. D. Andrachek, et al., May 2004.

7. WCAP-16083-NP-A, Revision 0, "Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry," S. L. Anderson, May 2006.

8. 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.

9. Framatome ANP Topical Report BAW-2241P-A, Rev. 1, "Fluence and Uncertainty Methodologies," December 1999.

10. NRC Regulatory Guide 1.190, Revision 0, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence, dated March 2001.

11. Areva Topical Report BAW-2308, Revision 2-A, "Initial RT_{NDT} of Linde 80 Weld Materials," approved March 2008.

12. NRC Regulatory Guide 1.99 Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," dated May 1988.

10.0 References (cont'd)

13. Letter from R. F. Saunders (VEPCO) to USNRC, "Virginia Electric and Power Company, Surry Power Station Units 1 and 2, Request for Exemption - ASME Code Case N-514, Proposed Technical Specifications Change, Revised Pressure/Temperature Limits and LTOPS Setpoint," Serial No. 95-197, June 8, 1995.

14. Letter from B. C. Buckley (USNRC) to J. P. O'Hanlon (VEPCO), "Surry Units 1 and 2 - Issuance of Amendments Re: Surry Units 1 and 2 Reactor Vessel Heatup and Cooldown Curves (TAC Nos. M92537 and M92538)," Serial No. 96-020, dated December 28, 1995.

15. Letter from J. Alan Price (VEPCO) to USNRC, ""Virginia Electric and Power Company (Dominion) Surry Power Station Units 1 and 2 Revised Reactor Vessel Materials Surveillance Capsule Withdrawal Schedules, October 26, 2009.

16. ASTM E-185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," dated July 1, 1982.

17. NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," dated July 2001.

18. Letter from L.N. Hartz (VEPCO) to USNRC, ""Virginia Electric and Power Company (Dominion) Surry Power Station Units 1 and 2 License Amendment Request Measurement Uncertainty Recapture Power Uprate, January 27, 2010.

Attachment 2

PROPOSED TECHNICAL SPECIFICATIONS PAGES (MARK-UP)

**Virginia Electric and Power Company
(Dominion)
Surry Power Station Units 1 and 2**

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of ~~28.8~~ 48 Effective Full Power Years (EFPY) ~~and 29.4 EFPY~~ for Units 1 and 2~~, respectively~~. The most limiting value of RT_{NDT} (~~228.4~~ 222.5°F) occurs at the 1/4-T, 0° azimuthal location in the Unit ~~1~~ 2 intermediate-to-lower shell circumferential weld. The limiting RT_{NDT} at the 1/4-T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. This ensures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results are presented in UFSAR Section 4.1. Reactor operation and resultant fast neutron (E greater than 1 MEV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the copper and nickel content of the material and the fluence was calculated in accordance with the recommendations of Regulatory Guide 1.99, Revision 2 "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curves of Figures 3.1-1 and 3.1-2 include predicted adjustments for this shift in RT_{NDT} at the end of ~~28.8~~ 48 EFPY ~~and 29.4 EFPY~~ for Units 1 and 2~~, respectively~~ (as well as adjustments for location of the pressure sensing instrument).

Surveillance capsules will be removed in accordance with the requirements of ASTM E185-82 and 10 CFR 50, Appendix H. The surveillance specimen withdrawal schedule is shown in the UFSAR. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure, or when the service period exceeds ~~28.8~~ 48 EFPY ~~or 29.4 EFPY~~ for Units 1 and 2~~, respectively,~~ prior to a scheduled refueling outage.

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

K_{IR} is provided by the code as a function of temperature relative to the RT_{NDT} of the material.

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{It}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

The heatup limit curve, Figure 3.1-1, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour. The cooldown limit curves of Figure 3.1-2 are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The cooldown limit curves are valid for cooldown rates up to 100°F/hr. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of 48 ~~28.8~~ EFPY ~~and 29.4 EFPY~~ for Units 1 and 2~~, respectively~~. The adjusted reference temperature was calculated using materials properties data from the B&W Owners Group Master Integrated Reactor Vessel Surveillance Program (MIRVSP) documented in the most recent revision to BAW-1543 and reactor vessel neutron fluence data obtained from plant-specific analyses.

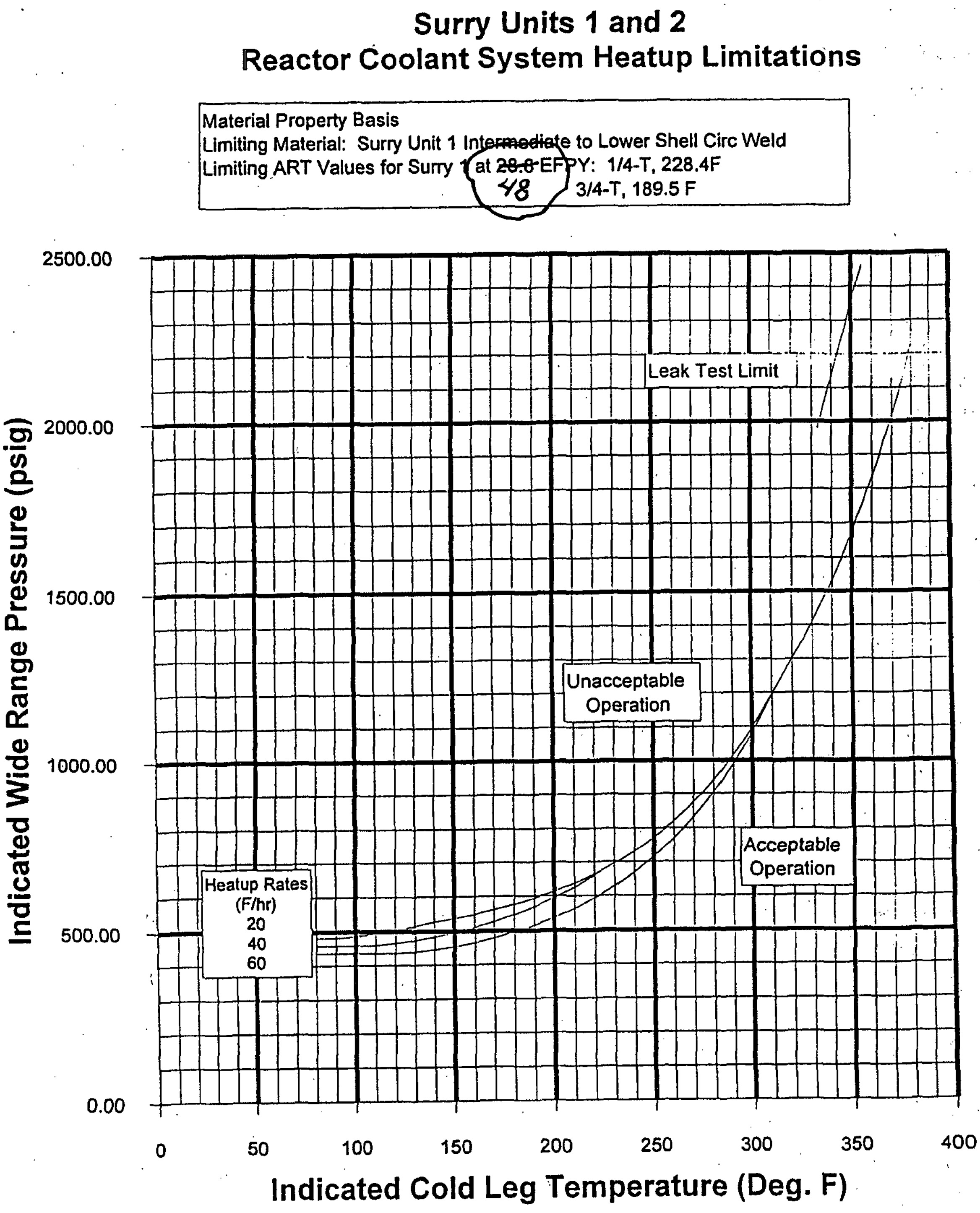

Figure 3.1-1 : Surry Units 1 and 2 Reactor Coolant System Heatup Limitations (Heatup Rates up to 60°F/hr) Applicable for ~~the first 28.8~~ 48 EFPY ~~for Surry Unit 1 and the first 29.4 EFPY for Surry Unit 2~~

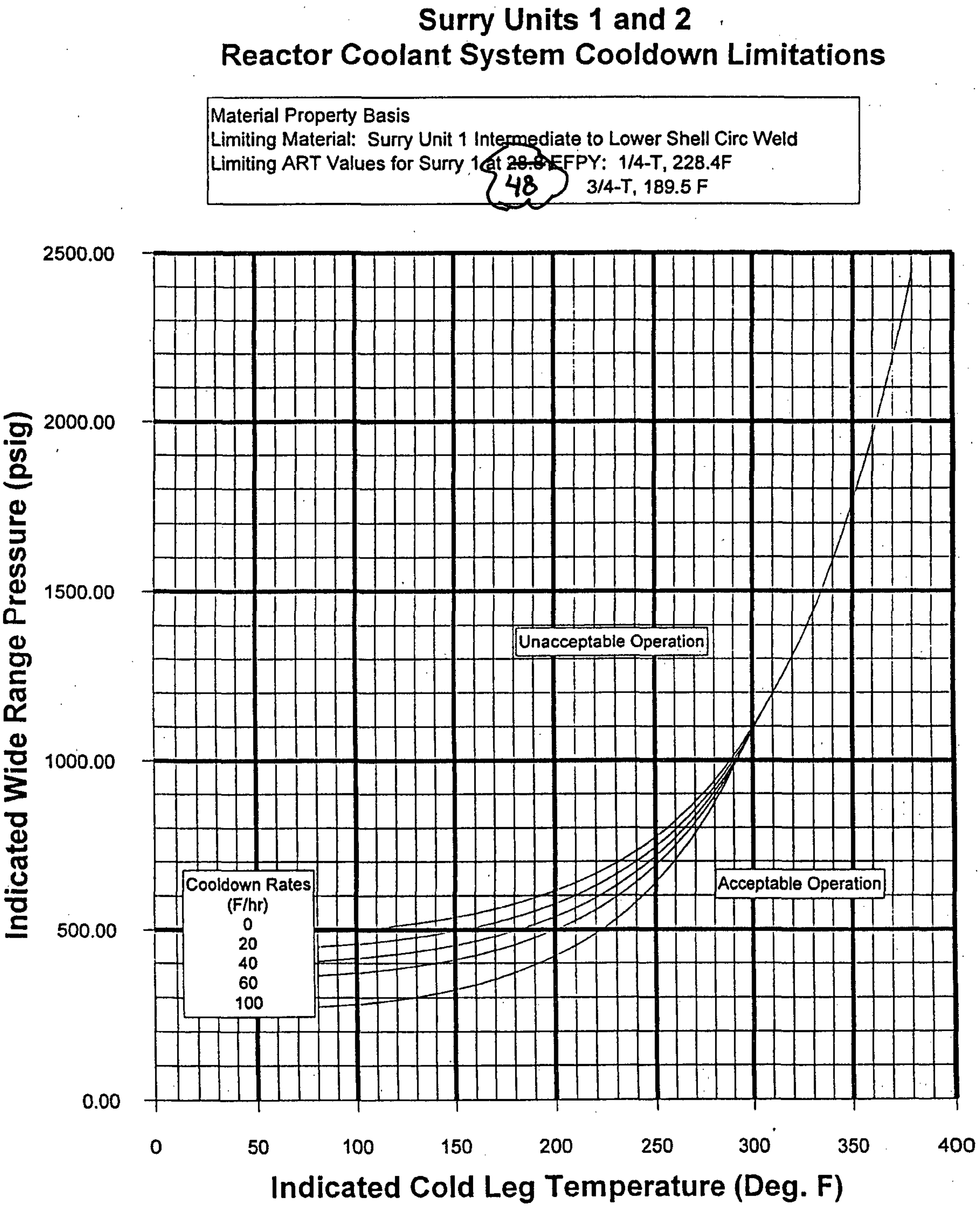

Figure 3.1-2 : Surry Units 1 and 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for ~~the first 28.8~~ 48 EFPY ~~for Surry Unit 1 and the first 29.4 EFPY for Surry Unit 2~~

Attachment 3

PROPOSED TECHNICAL SPECIFICATIONS PAGES (TYPED)

**Virginia Electric and Power Company
(Dominion)
Surry Power Station Units 1 and 2**

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 48 Effective Full Power Years (EFPY) for Units 1 and 2. The most limiting value of RT_{NDT} (222.5°F) occurs at the 1/4-T, 0° azimuthal location in the Unit 2 intermediate-to-lower shell circumferential weld. The limiting RT_{NDT} at the 1/4-T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. This ensures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results are presented in UFSAR Section 4.1. Reactor operation and resultant fast neutron (E greater than 1 MEV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the copper and nickel content of the material and the fluence was calculated in accordance with the recommendations of Regulatory Guide 1.99, Revision 2 "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curves of Figures 3.1-1 and 3.1-2 include predicted adjustments for this shift in RT_{NDT} at the end of 48 EFPY for Units 1 and 2 (as well as adjustments for location of the pressure sensing instrument).

Surveillance capsules will be removed in accordance with the requirements of ASTM E185-82 and 10 CFR 50, Appendix H. The surveillance specimen withdrawal schedule is shown in the UFSAR. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure, or when the service period exceeds 48 EFPY for Units 1 and 2 prior to a scheduled refueling outage.

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

K_{IR} is provided by the code as a function of temperature relative to the RT_{NDT} of the material.

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{It}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

The heatup limit curve, Figure 3.1-1, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour. The cooldown limit curves of Figure 3.1-2 are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The cooldown limit curves are valid for cooldown rates up to 100°F/hr. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of 48 EFPY for Units 1 and 2. The adjusted reference temperature was calculated using materials properties data from the B&W Owners Group Master Integrated Reactor Vessel Surveillance Program (MIRVSP) documented in the most recent revision to BAW-1543 and reactor vessel neutron fluence data obtained from plant-specific analyses.

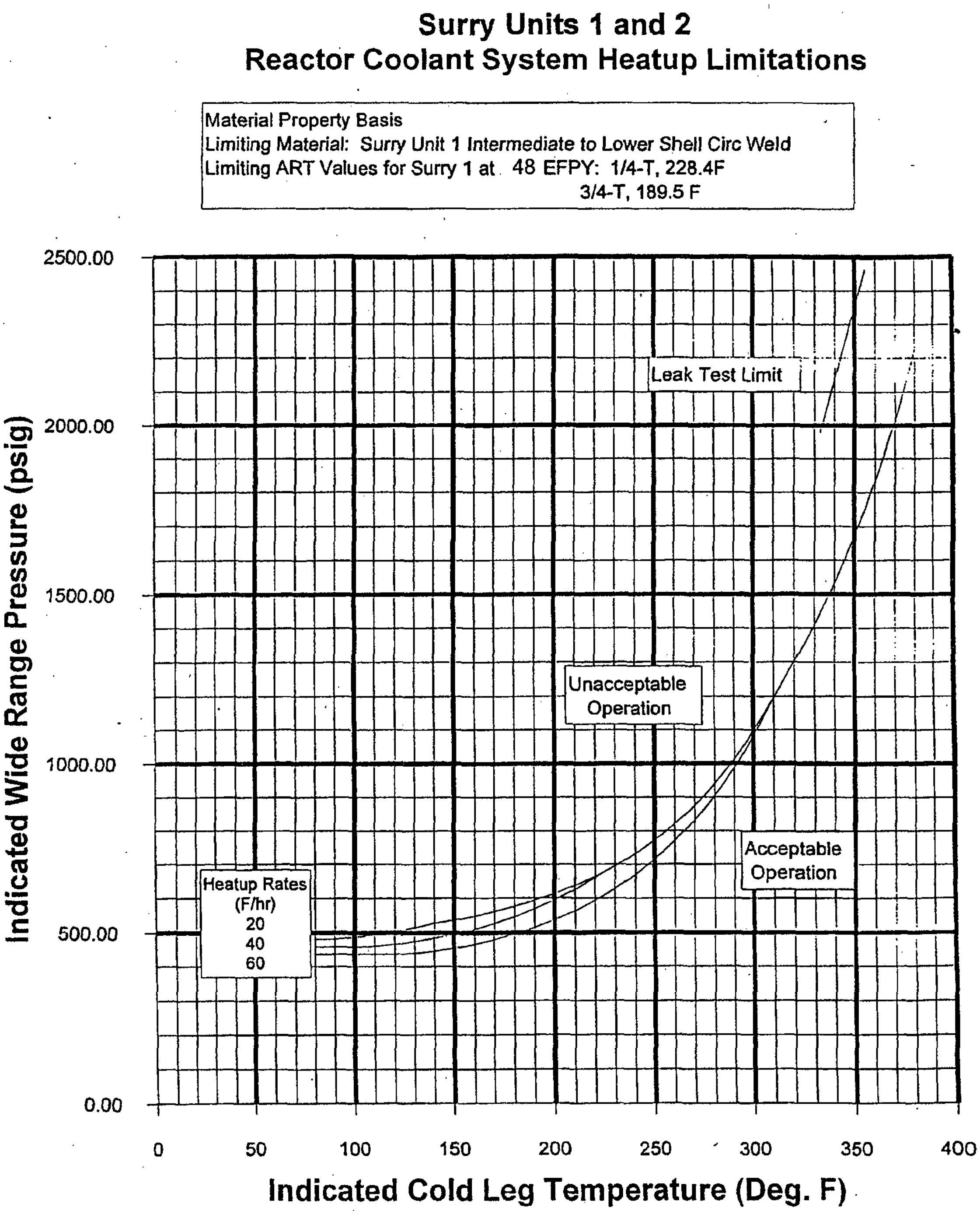

Figure 3.1-1 : Surry Units 1 and 2 Reactor Coolant System Heatup Limitations (Heatup Rates up to 60°F/hr) Applicable for 48 EFPY

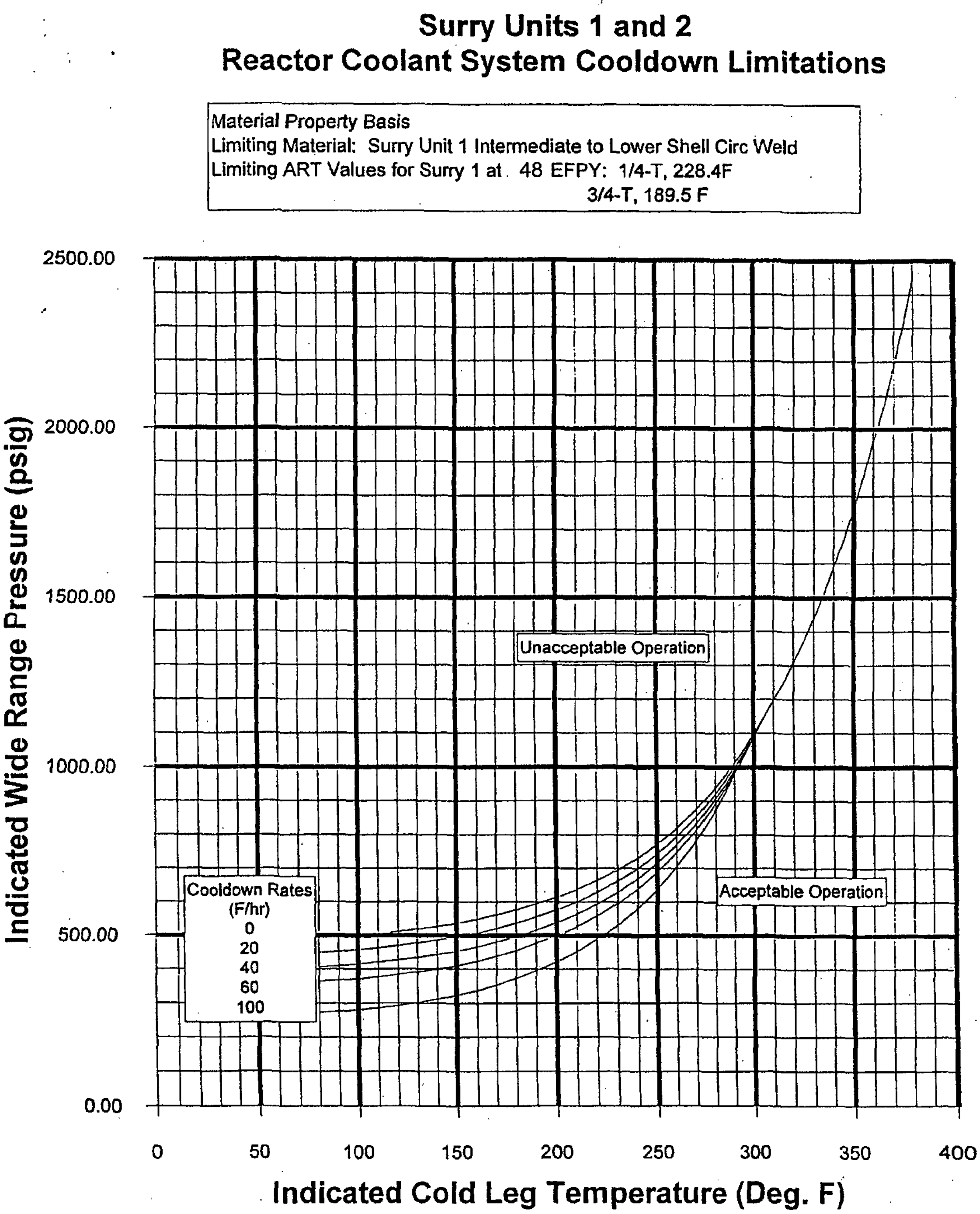

Figure 3.1-2 : Surry Units 1 and 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for 48 EFPY

Attachment 4

REACTOR VESSEL INTEGRITY DATABASE UPDATE

Virginia Electric and Power Company
(Dominion)
Surry Power Station Units 1 and 2

Facility: Surry Unit 1
Vessel Manufacturer: B&W and Rotterdam Dockyard

RPV Weld Wire Heat or Material ID	Location	Best-Estimate Copper (wt%)	Best-Estimate Nickel (wt%)	ID Fluence (x1E19)	Assigned Material Chemistry Factor	Method of Determining CF	Initial RT(NDT)	Shift ΔRT(NDT)	Sigma(I)	Sigma(delta)	Margin	Inner Surf. ART or RT(PTS)	1/4-T ART*	3/4-T ART**
122V109VA1	Nozzle Shell Forging	0.110	0.740	0.775	76.1	Tables	40	70.7	0.0	17.0	34.0	144.7	134.2	115.0
C4326-1	Intermediate Shell	0.110	0.550	5.660	73.5	Tables	10	104.8	0.0	17.0	34.0	148.8	141.3	122.6
C4326-2	Intermediate Shell	0.110	0.550	5.660	73.5	Tables	0	104.8	0.0	17.0	34.0	138.8	131.3	112.6
4415-1	Lower Shell	0.102	0.493	5.660	85.0	Surv. Data	20	121.2	0.0	8.5	17.0	158.2	149.5	127.9
4415-2	Lower Shell	0.110	0.500	5.660	73.0	Tables	0	104.1	0.0	17.0	34.0	138.1	130.6	112.1
J726/25017	Nozzle to Int Shell Circ Weld	0.330	0.100	0.775	152.0	Tables	0	141.1	20.0	28.0	68.8	209.9	189.1	150.8
SA-1585/72445	Int. to Low Sh. Circ (ID 40%)	0.220	0.540	5.610	167.0	BAW-2308-1-A	-72.5	237.9	12.0	28.0	60.9	226.3	209.1	166.7
SA-1650/72445	Int. to Low Sh. Circ (OD 60%)	0.220	0.540	5.610	167.0	BAW-2308-1-A	-72.5	237.9	12.0	28.0	60.9	226.3	209.1	166.7
SA-1494/8T1554	Int Shell Long. Welds L3 & L4	0.160	0.570	1.080	167.0	BAW-2308-1-A	-48.6	170.6	18.0	28.0	66.6	188.6	165.4	121.4
SA-1494/8T1554	Lower Shell Long. Weld L1	0.160	0.570	1.040	167.0	BAW-2308-1-A	-48.6	168.8	18.0	28.0	66.6	186.8	163.7	119.8
SA-1526/299L44	Lower Shell Long. Weld L2	0.340	0.680	1.040	220.6	Tables	-74.3	223.0	12.8	28.0	61.6	210.3	179.8	121.8

* 1/4-T ART value of 228.4 F was used in the determination of P/T limits (Approved by NRC on 12/28/95)
** 3/4-T ART value of 189.5 F was used in the determination of P/T limits (Approved by NRC on 12/28/95)
Note: Shaded cells indicate a changed value relative to Dominion's most recent update to SM-1008-M.

Facility: Surry Unit 2
Vessel Manufacturer: B&W and Rotterdam Dockyard

RPV Weld Wire Heat or Material ID	Location	Best-Estimate Copper (wt%)	Best-Estimate Nickel (wt%)	ID Fluence (x1E19)	Assigned Material Chemistry Factor	Method of Determining CF	Initial RT(NDT)	Shift ΔRT(NDT)	Sigma(I)	Sigma(delta)	Margin	Inner Surf. ART or RT(PTS)	1/4-T ART*	3/4-T ART**
123V303VA1	Nozzle Shell Forging	0.110	0.720	0.632	75.8	Tables	30	66.1	0.0	17.0	34.0	130.1	119.8	101.4
C4331-2	Intermediate Shell	0.120	0.600	5.380	83.0	Tables	-10	117.6	0.0	17.0	34.0	141.6	132.9	111.6
C4339-2	Intermediate Shell	0.110	0.540	5.380	73.4	Tables	-20	104.0	0.0	17.0	34.0	118.0	110.3	91.5
C4208-2	Lower Shell	0.150	0.550	5.380	107.3	Tables	-30	151.9	0.0	17.0	34.0	155.9	144.7	117.2
C4339-1	Lower Shell	0.107	0.530	5.380	70.8	Tables	-10	100.3	0.0	17.0	34.0	124.3	116.9	98.8
L737/4275	Nozzle to Int Shell Circ Weld	0.350	0.100	0.632	160.5	Tables	0	139.9	20.0	28.0	68.8	208.7	187.0	147.9
R3008/0227	Int. to Lower Shell Circ Weld	0.187	0.545	5.370	132.4	Surv. Data	0	187.6	20.0	14.0	48.8	236.4	222.5	188.6
WF-4/8T1762	Int. Shell Long. L4 (ID 50%)	0.190	0.570	1.140	167.0	BAW-2308-1-A	-48.6	173.1	18.0	28.0	66.6	191.1	168.0	123.6
SA-1585/72445	Int. Sh. L3 (100%), L4 (OD 50)	0.220	0.540	1.140	167.0	BAW-2308-1-A	-72.5	173.1	12.0	28.0	60.9	161.5	138.4	94.1
WF-4/8T1762	LS L2 (ID 63%), L1 (100)	0.190	0.570	1.140	167.0	BAW-2308-1-A	-48.6	173.1	18.0	28.0	66.6	191.1	168.0	123.6
WF-8/8T1762	LS Long. Weld L2 (OD 37%)	0.190	0.570	1.140	167.0	BAW-2308-1-A	-48.6	173.1	18.0	28.0	66.6	191.1	168.0	123.6

* 1/4-T ART value of 228.4 F was used in the determination of P/T limits (Approved by NRC on 12/28/95)
** 3/4-T ART value of 189.5 F was used in the determination of P/T limits (Approved by NRC on 12/28/95)
Note: Shaded cells indicate a changed value relative to Dominion's most recent update to SM-1008-M.

CvUSE Values

Facility: Surry Unit 1

Vessel Manufacturer: B&W and Rotterdam Dockyard

RPV Weld Wire Heat or Material ID	Location	Forging or Flux Type	USE @ 1/4 T	1/4-T Fluence (x1E19)	Unirradiated USE	Unirradiated USE Method	%Drop in USE @ EOL @ 1/4 T	%Drop in USE Method	Cu %
122V109VA1	Nozzle Shell Forging	SA508, Cl. 2	69.2	0.473	83.0	Measured/MTEB 5-2	16.6%	Pos 1.2	0.11
C4326-1	Intermediate Shell	SA533, Gr. B1	84.4	3.453	115.0	Measured	26.7%	Pos 1.2	0.11
C4326-2	Intermediate Shell	SA533, Gr. B1	68.9	3.453	94.0	Measured/MTEB 5-2	26.7%	Pos 1.2	0.11
4415-1	Lower Shell	SA533, Gr. B1	76.6	3.453	103.0	Measured	25.6%	Pos 1.2	0.10
4415-2	Lower Shell	SA533, Gr. B1	60.9	3.453	83.0	Measured/MTEB 5-2	26.7%	Pos 1.2	0.11
J726/25017	Nozzle to Int Shell Circ Weld	SAF 89	EMA	0.473	EMA	Estimate	EMA	EMA	0.33
SA-1585/72445	Int. to Low Sh. Circ (ID 40%)	Linde 80	EMA	3.423	EMA	Measured	EMA	EMA	0.22
SA-1650/72445	Int. to Low Sh. Circ (OD 60%)	Linde 80	EMA	3.423	EMA	Measured	EMA	EMA	0.22
SA-1494/8T1554	Int Shell Long. Welds L3 & L4	Linde 80	EMA	0.659	EMA	Estimate	EMA	EMA	0.16
SA-1494/8T1554	Lower Shell Long. Weld L1	Linde 80	EMA	0.634	EMA	Estimate	EMA	EMA	0.16
SA-1526/299L44	Lower Shell Long. Weld L2	Linde 80	EMA	0.634	EMA	Measured	EMA	EMA	0.34

Note: Shaded cells indicate a changed value relative to Dominion's most recent update to the NRC's Reactor Vessel Integrity Database (RVID) (Last Update by Dominion Letter Serial No. 06-434 dated 6/13/06).

CvUSE Values
Facility: Surry Unit 2
Vessel Manufacturer: B&W and Rotterdam Dockyard

RPV Weld Wire Heat or Material ID	Location	Forging or Flux Type	USE @ 1/4 T	1/4-T Fluence (x1E19)	Unirradiated USE	Unirradiated USE Method	%Drop in USE @ 1/4 T	%Drop in USE Method	Cu %
123V303VA1	Nozzle Shell Forging	SA508, Cl. 2	87.6	0.386	104.0	Measured/MTEB 5-2	15.8%	Pos 1.2	0.11
C4331-2	Intermediate Shell	SA533, Gr. B1	60.9	3.282	84.0	Measured/MTEB 5-2	27.5%	Pos 1.2	0.12
C4339-2	Intermediate Shell	SA533, Gr. B1	61.2	3.282	83.0	Measured/MTEB 5-2	26.3%	Pos 1.2	0.11
C4208-2	Lower Shell	SA533, Gr. B1	64.7	3.282	94.0	Measured/MTEB 5-2	31.2%	Pos 1.2	0.15
C4339-1	Lower Shell	SA533, Gr. B1	77.8	3.282	105.0	Measured	25.9%	Pos 1.2	0.11
L737/4275	Nozzle to Int Shell Circ Weld	SAF 89	EMA	0.386	EMA	Estimate	EMA	EMA	0.35
R3008/0227	Int. to Lower Shell Circ Weld	Grau Lo	51.5	3.276	90.0	Measured	42.7%	Pos 1.2	0.19
WF-4/8T1762	Int. Shell Long. L4 (ID 50%)	Linde 80	EMA	0.695	EMA	Estimate	EMA	EMA	0.19
SA-1585/72445	Int. Sh. L3 (100%), L4 (OD 50)	Linde 80	51.6	0.695	77.0	Measured	33.0%	Pos 1.2	0.22
WF-4/8T1762	LS L2 (ID 63%), L1 (100)	Linde 80	EMA	0.695	EMA	Estimate	EMA	EMA	0.19
WF-8/8T1762	LS Long. Weld L2 (OD 37%)	Linde 80	EMA	0.695	EMA	Estimate	EMA	EMA	0.19

Note: Shaded cells indicate a changed value relative to Dominion's most recent update to the NRC's Reactor Vessel Integrity Database (RVID) (Last Update by Dominion Letter Serial No. 06-434 dated 6/13/06).