VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

December 17, 2004

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

Serial No.	04-755
NLOS/GDM	R0
Docket Nos.	50-280
	50-281
License Nos.	DPR-32
	DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2
PROPOSED TECHNICAL SPECIFICATIONS CHANGE REQUEST FOR REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITS, LTOPS SETPOINT, AND LTOPS ENABLE TEMPERATURE WITH EXEMPTION REQUEST FOR ALTERNATE MATERIAL PROPERTIES BASIS PER 10 CFR 50.60(b)

Pursuant to 10 CFR 50.90, Virginia Electric and Power Company (Dominion) requests amendments, in the form of changes to the Technical Specifications (TS) to Facility Operating Licenses Numbers DPR-32 and DPR-37 for Surry Power Station Units 1 and 2, respectively. The existing Reactor Coolant System (RCS) Pressure/ Temperature (P/T) operating limits, Low Temperature Overpressure Protection System (LTOPS) setpoint, and LTOPS enable temperature (T_{enable}) basis included in the Surry Unit 1 and Unit 2 TS are valid to cumulative core burnups of 28.8 Effective Full Power Years (EFPY) (approximately year 2012) and 29.4 EFPY (approximately year 2013) for Surry Units 1 and 2, respectively. The proposed TS change revises RCS P/T operating limits, LTOPS setpoint, and LTOPS T_{enable} basis for cumulative core burnups up to 47.6 EFPY and 48.1 EFPY (corresponding to the period of the renewed licenses) for Surry Units 1 and 2, respectively. In addition, changes to the TS Bases reflecting the proposed changes are included for information only. An update to the NRC Reactor Vessel Integrity Database (RVID) is also provided.

A discussion of the proposed TS change is provided in Attachment 1. The marked-up and proposed TS pages reflecting the proposed change are provided in Attachments 2 and 3, respectively.

We have evaluated the proposed TS change and have determined that it does not involve a significant hazards consideration as defined in 10 CFR 50.92. 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, and no significant increase in individual or cumulative occupational radiation exposure will occur. Therefore, the proposed amendment is eligible for categorical exclusion as set forth in

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10 CFR 51.22(c)(9), and, 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 bases for these two determinations are provided in Attachment 1.

The proposed TS change has been reviewed and approved by the Station Nuclear Safety and Operating Committee and the Management Review Committee. After NRC approval of the proposed change, Dominion requests a six month implementation period to perform the changes necessary to implement the revised limits. Since the existing TS P/T limits, LTOPS setpoint, and LTOPS T_{enable} basis are valid to cumulative core burnups of 28.8 EFPY (approximately year 2012) and 29.4 EFPY (approximately year 2013) for Surry Units 1 and 2, respectively, the extended implementation time will have no impact on safe operation of Surry Unit 1 or 2.

Exemption Request

Finally, a request for exemption pursuant to 10 CFR 50.12 and 50.60(b) from the requirements of 10 CFR 50.61 and 10 CFR 50 Appendix G is included in Attachment 4 to allow Dominion to revise the Surry reactor vessel material initial properties basis using BAW-2308, Revision 1. It should be noted that approval of the exemption is not required for approval of the proposed change to the Surry Units 1 and 2 Technical Specifications. Dominion requests the exemption to provide margin for possible future improvements in plant safety (e.g., reduced probability of undesired PORV lifts during reactor coolant pump startups).

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

Very truly yours,

Leslie N. Hartz
Vice President – Nuclear Engineering

Attachments

Commitments made in this letter: None

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Serial No. 04-755
Docket Nos. 50-280, 281

Subject: Technical Specification Change for Revised P/T Limits, LTOPS Setpoints, LTOPS T_{Enable}

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Leslie N. Hartz who is Vice President – Nuclear Engineering of Virginia Electric and Power Company. She has affirmed before me that she 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 her knowledge and belief.

Acknowledged before me this 17TH day of December, 2004.

My Commission Expires: May 31, 2006.

Vicki L. Hull
Notary Public

(SEAL)

ATTACHMENT 1

Discussion of Change

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

Discussion of Changes

1.0 Introduction

Virginia Electric and Power Company (Dominion) proposes a change to the Surry Units 1 and 2 Technical Specifications pursuant to 10CFR50.90. The proposed change is requested to provide Reactor Coolant System (RCS) pressure/temperature (P/T) operating limits, Low Temperature Overpressure Protection System (LTOPS) setpoints, and LTOPS enable temperature (T_{enable}) values that are valid for cumulative core burnups up to 47.6 Effective Full Power Years (EFPY) and 48.1 EFPY (corresponding to the period of the renewed licenses) for Surry Units 1 and 2, respectively. The currently licensed set of unadjusted RCS P/T limit curves (i.e., the set submitted in Reference 1, and approved in References 2 and 3) are being replaced with a new set (Reference 4, attached as Appendix D). In accordance with the ASME Code Section XI, the higher cumulative core burnup applicability limits in this submittal are achieved through margins obtained by using K_{1C} stress intensity factors in the development of the unadjusted RCS P/T limit curves (Reference 4, attached as Appendix D) instead of K_{1A} stress intensity factors that represent the current licensing basis. In addition, Technical Specifications bases changes reflecting the proposed change discussed above are included for your information. The proposed TS change qualifies for categorical exclusion for an environmental assessment as set forth in 10CFR51.22(c)(9). Therefore, no environmental impact statement or environmental assessment is needed in connection with approval of the proposed Technical Specifications change.

2.0 Background

10 CFR 50 Appendix G specifies the fracture toughness requirements for ferritic materials of pressure retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime. The ASME Boiler and Pressure Vessel Code (BPVC) forms the basis for the requirements of Appendix G to 10 CFR 50. Appendix G references the requirements of ASME BPVC Section III, Division 1, "Rules for Construction of Nuclear Power Plant Components," and ASME BPVC Section XI which presents the "Rules for Inservice Inspection of Nuclear Power Plant Components."

10 CFR 50 Appendix H defines the requirements for reactor vessel materials surveillance programs. Dominion compliance with the requirements of Appendix H is documented for Surry Units 1 and 2 in References 5 and 6, respectively. Appendix H states that the purpose of the materials surveillance program is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors resulting from exposure of these materials to neutron irradiation and the thermal environment. Fracture toughness data are obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the reactor vessel. These data are used as described in Appendix G

to 10 CFR 50. The current Surry Units 1 and 2 surveillance capsule withdrawal schedules are documented in Reference 7. The withdrawal schedules contained in the Surry UFSAR were approved in Reference 8.

A method for performing analyses to guard against brittle fracture in reactor pressure vessels is presented in "Protection Against Non-ductile Failure," Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code. The method utilizes fracture mechanics concepts and is based on the reference nil-ductility temperature (RT_{NDT}). RT_{NDT} is defined as the greater of either the drop weight nil-ductility transition temperature (NDTT per ASTM E-208) or the temperature 60°F less than the 50 ft-lb (and 35 mil lateral expansion) temperature as determined from Charpy specimens oriented normal (transverse) to the major working direction of the material. The RT_{NDT} of a given material is used to index that material to a reference stress intensity factor curve (K_{1A}), which appears in Appendix G of the ASME Section XI. The K_{1A} curve is a lower bound of dynamic, crack arrest, and static fracture toughness results obtained from several heats of pressure vessel steel. When a given material is indexed to the K_{1A} curve, allowable stress intensity factors can be obtained for this material as a function of temperature. Allowable operating limits can then be determined utilizing these allowable stress intensity factors.

RT_{NDT} and the operating limits of nuclear power plants can be adjusted to account for the effects of radiation on the reactor vessel material properties. The radiation embrittlement or changes in mechanical properties of a given reactor pressure vessel steel can be monitored by a reactor vessel materials surveillance program. A surveillance capsule is periodically removed from the operating nuclear reactor and the encapsulated specimens are tested. The increase in the average Charpy V-notch 30 ft-lb temperature (ΔRT_{NDT}) due to irradiation is added to the original RT_{NDT} to adjust the RT_{NDT} for radiation embrittlement. This adjusted RT_{NDT} (RT_{NDT} initial plus ΔRT_{NDT}) is used to index the material to the K_{1A} curve and to set operating limits for the nuclear power plant, which reflect the effects of irradiation on the reactor vessel materials.

The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and nickel) present in reactor vessel steels. The NRC has published a method for predicting radiation embrittlement in Regulatory Guide 1.99, Revision 2 (Radiation Embrittlement of Reactor Vessel Materials, Reference 13). This methodology permits the use of credible surveillance data, such as that obtained from the capsule analysis, if it is available, in place of the calculational methodology based on a curve fit of an irradiated materials properties database. The current Surry Units 1 and 2 design and licensing basis reflects the regulatory requirements described above by the imposition of restrictions on allowable pressure and temperature (RCS P/T limits) and on heatup and cooldown rates. The Low Temperature Overpressure Protection System (LTOPS) ensures that material integrity limits are not exceeded during design basis accidents. The replacement of the RCS P/T limit curves and the revision of the LTOPS setpoint and LTOPS T_{enable} value proposed in this license amendment are performed in accordance with ASME Section XI (i.e., use of the K_{1C} stress intensity formulation as allowed in Code Case N-641, Reference 10) and the regulatory requirements described above as well.

3.0 Discussion

3.1 Licensing and Design Basis

The current Surry Units 1 and 2 Technical Specification RCS P/T limits (using a K_{1A} stress intensity formulation), LTOPS setpoint and LTOPS T_{enable} value were provided to the NRC for approval in Reference 1. The NRC approved the Technical Specification change in References 2 and 3. The cumulative core burnup applicability limit for the current limits are 28.8 EFPY for Surry Unit 1 and 29.4 EFPY for Surry Unit 2. This corresponds to a ¼-thickness (¼-T) RT_{NDT} of 228.4°F, which conservatively represents the limiting materials for both Surry Units 1 and 2. The submittals supporting license renewal for Surry Units 1 and 2 (References 7 and 9) were approved by the NRC (Reference 8). These submittals documented that the limiting ¼-T RT_{NDT} value of 238.2°F was predicted to bound the end-of-license-renewal limiting material for both Surry Units 1 and 2 (Unit 1 Longitudinal Weld L2, SA-1526). Reviews of the Surry Units 1 and 2 reactor vessel integrity data continue to confirm the conclusions from the license renewal effort.

To support the end-of-license-renewal limiting material for both Surry Units 1 and 2 (Unit 1 Longitudinal Weld L2, SA-1526), new Surry Units 1 and 2 RCS P/T limit curves were prepared using the K_{1C} stress intensity formulation and a limiting ¼-T RTNDT of 238.2°F. This RT_{NDT} value corresponds to operation for 47.6 EFPY for Surry Unit 1 and 48.1 EFPY for Surry Unit 2 (Reference 4, attached as Appendix D). The K_{1C} curve is a lower bound of dynamic, crack initiation, and static fracture toughness results obtained from several heats of pressure vessel steel. Use of the K_{1C} stress intensity formulation is allowed by 10 CFR 50 Appendix G that, in turn, endorses the use of ASME Section XI. To extend the cumulative core burnup applicability limit for the Surry Units 1 and 2 Technical Specification RCS P/T limits, LTOPS setpoint, and LTOPS T_{enable} value, the Surry Units 1 and 2 Technical Specifications governing these values are being revised to be consistent with a ¼-T RT_{NDT} value of 238.2°F. These changes include:

1. The revised RCS PTT limits in Appendix A include modifications for pressure and temperature measurement uncertainty, as well as the pressure difference between the point of measurement (RCS hot leg) and point of interest (reactor vessel beltline).
2. The cumulative core burnup applicability limits will be extended to 47.6 EFPY for Surry Unit 1 and 48.1 EFPY for Surry Unit 2.
3. Technical Specification LTOPS setpoint and LTOPS T_{enable} value have been prepared to reflect the extended cumulative core burnup applicability limits using methodologies that comply with the applicable regulations (e.g., 10 CFR 50 Appendix G), industry codes (e.g., ASME Section XI), and previously approved methods.

4. The changes to the Technical Specification P/T limits, LTOPS setpoint, and T_{enable} value will be common for Surry Units 1 and 2 to provide more consistent operational requirements for the two units.

In addition to the Technical Specification changes, Dominion is increasing the administrative cooldown rate limit from 50°F/hr to 75°F/hr. The justification for this increase is described in Section 3.4.4.

3.2 Design Inputs

3.2.1 Unadjusted Pressure/Temperature Limit Curves

The current Surry Reactor Coolant System (RCS) pressure/temperature (P/T) limit curves (designed for 40 calendar years of operation) were developed by Westinghouse and were transmitted to the NRC in Reference 1. The design limit for the current Surry LTOPS setpoints is 110% of the isothermal RCS P/T curve based upon K_{1A} stress intensity factors (Reference 1), as allowed by 10 CFR 50 Appendix G that, in turn, endorses the use of Section XI of the ASME Code (specifically ASME Code Case N-514, References 1, 2, and 3). The K_{1A} curve is a lower bound of dynamic, crack arrest, and static fracture toughness results obtained from several heats of pressure vessel steel.

Surry RCS P/T limit curves (designed for 60 years of operation) were also developed by Westinghouse, who documented their technical bases in Reference 4 (Appendix D). Westinghouse developed the P/T limit curves for a ¼-T RT_{NDT} of 238.2°F. This RT_{NDT} value is predicted to bound the end-of-license-renewal limiting material for both Surry Units 1 and 2 (Unit 1 Longitudinal Weld L2, SA-1526) as documented in Reference 7. ASME Code Case N-641, Reference 10, supports the use of 100% of the isothermal P/T curve using K_{1C} stress intensity factors as the design limit for LTOPS setpoints. The K_{1C} curve is a lower bound of dynamic, crack initiation, and static fracture toughness results obtained from several heats of pressure vessel steel. The use of K_{1C} stress intensity factors provides greater margin in the development of P/T limit curves relative to the use of K_{1A}. However, the use of the K_{1C} stress intensity formulation requires that pressure and temperature measurement uncertainties be applied, and that 100% of the isothermal P/T limit curve be used instead of 110% of the isothermal P/T curve as allowed when using K_{1A}. The P/T limit curves provided in Reference 4 (Appendix D) are valid up to 47.6 EFPY for Surry Unit 1 and 48.1 EFPY for Surry Unit 2.

The revised Surry Units 1 and 2 design basis RCS P/T limit curves (Reference 4) do not include margins for pressure and temperature measurement uncertainty, or for the pressure difference between the point of measurement (RCS hot leg) and the point of interest (reactor vessel beltline). Curves that have been modified to include pressure and temperature measurement uncertainty, and the pressure difference between the point of measurement (RCS hot leg) and the point of interest (reactor vessel beltline) are presented in Appendix A.

3.2.2 Reactor Vessel Fluence (E > 1 MeV), RT_{NDT}, and Cumulative Core Burnup

The NRC was provided (in Reference 7) information regarding reactor vessel fluence (E > 1 MeV) versus burnup and RT_{NDT} versus reactor vessel fluence in support of 60-year operation. The information in Reference 7 demonstrated that the limiting ¼-T RTNDT value of 238.2°F conservatively represents the end-of-license-renewal limiting material for both Surry Units 1 and 2. The limiting material, Unit 1 Longitudinal Weld L2, SA- 1526, is predicted to have an inner surface fluence of (0.79 X 10^{19} n/cm2) that corresponds to a cumulative core burnup of 47.6 EFPY for Surry Unit 1 and 48.1 EFPY for Surry Unit 2. Currently available material surveillance data has not changed this conclusion. The methodology used for the fluence values given above has been shown to meet the requirements of Regulatory Guide 1.190 verbatim, or by equivalent demonstration, as documented in Reference 14. Although Reference 14 represents an RAI response for the North Anna Units, the methodology used for calculating the Surry fluences is the same as that used for North Anna.

3.2.3 RCS Pressure Measurement Uncertainty for LTOPS

The Channel Statistical Accuracy (CSA) for the RCS pressure measurement uncertainty to be used for the establishment of the LTOPS setpoint has been calculated to be +2.05% of a 0 psig to 1000 psig instrument span, for a total CSA of 20.5 psi. For conservatism, a value of 25 psi is used in the development of the LTOPS setpoint. This uncertainty reflects the Narrow Range RCS pressure uncertainty for the actuation of the PORV bistables. Narrow Range RCS pressure is measured in the RCS hot leg for the PORV with the limiting setpoint.

3.2.4 Wide Range RCS Pressure Measurement Uncertainty for P/T Limits

The CSA for the Wide Range RCS pressure measurement uncertainty has been calculated to be 2.213% of a 0 psig to 3000 psig instrument span (indication uncertainty included), for a total CSA of 67 psi. Wide Range RCS pressure is measured in the RCS hot leg. The Wide Range RCS pressure measurement channel is used for confirming RCS pressure during normal operation heatup and cooldown.

3.2.5 Wide Range RCS Temperature Measurement Uncertainty

The CSA for the Wide Range RCS temperature measurement uncertainty has been calculated to be 2.0% of a 0°F to 700°F instrument span, for a total CSA of 14°F. For conservatism, a value of 20°F is used in the development of the LTOPS enabling temperature. Wide Range RCS temperature is measured in the RCS cold leg. The Wide Range RCS temperature measurement channel is used for confirming RCS temperature during normal operation heatup and cooldown, and as input for the LTOPS enabling temperature.

3.2.6 Pressure Difference between Hot Leg and Reactor Vessel Beltline

The pressure difference between the point of measurement (Narrow Range or Wide Range RCS pressure measured in the RCS hot leg) and the point of interest (reactor vessel beltline) has been determined to be 57 psi. This value was developed in consideration of one reactor coolant pump (RCP), two RCP, and three RCP operation. This difference is applied as a bias to measured RCS pressure, to simulate pressure measurement at the reactor vessel beltline.

3.2.7 LTOPS PORV Lift Setpoint "Overshoot" Values from Mass Addition Accident Analysis

The mass addition and heat addition accident analyses that support the proposed Surry Units 1 and 2 Technical Specification LTOPS setpoint are unchanged from those used for the current licensing analysis (References 1 through 3). The PORV lift setpoint "overshoot" values determined in the accident analysis are presented in Appendix B. (See column labeled "PORV Setpoint Overshoot".) The maximum PORV lift setpoint overshoot is a function of the PORV lift setpoint and RCS temperature. Note that a pressure measurement location bias of approximately 9 psi, originally applied to the values contained in the "PORV Setpoint Overshoot" column to account for the static head difference between the RCS hot leg and the reactor vessel beltline, has been removed because it is redundant. The treatment of static head pressure measurement bias is now contained in the 57 psi value described in Item 3.2.6.

3.2.8 Margin Term for ASME Code Case N-641

In the plant specific determination of LTOPS T_{enable}, ASME Code Section XI (specifically Code Case N-641, Reference 10) requires that a margin term value be determined and applied to ensure that LTOPS provides adequate protection against brittle failure at low temperatures. This value represents a margin term that accommodates the specific geometry and design pressure of the reactor vessel considered. The following is the plant specific determination of the ASME Code Section XI margin term:

ASME Section XI Margin Term = 50 ln [((1.1 • M_m (p R_i /t)) - 33.2)/20.734] (from ASME Code Section XI)

M_m= 0.926 $t^{1/2}$ for IS axial flaw, $2 \leq t^{1/2} \leq 3.464$
p= vessel design pressure = 2.5 ksia
R_i= 78.95 in.
t= 8.08 in.
The Surry specific values for p, R_i, and t are from Reference 4 (Appendix D), Section 6.

Section XI Margin Term = 50 ln [(1.1 • -0.926•$8.08^{1/2}$ • (2.5•78.95/8.08) - 33.2)/20.734]

Section XI Margin Term = 29.7°F

(Note: The ASME Code Section XI formulation for the membrane stress correction factor, M, is valid since $t^{1/2} = (8.08)^{1/2} = 2.84$, which satisfies the inequality $2 \leq t^{1/2} \leq 3.464$.)

3.3 Method of Analysis

To develop the proposed Surry Units 1 and 2 Technical Specification P/T limit curves, LTOPS setpoint, and LTOPS T_{enable} value, the unadjusted P/T limit curves (Reference 4, attached as Appendix D)) were modified to account for pressure and temperature measurement uncertainty, and for the pressure difference between the point of measurement (RCS hot leg) and the point of interest (the reactor vessel beltline). The resulting proposed Surry Units 1 and 2 Technical Specification P/T limit curves are presented in Appendix A.

To determine the allowable TS LTOPS setpoint value, the temperature-dependent pressurizer PORV lift setpoint pressure "overshoot" values determined in the design basis mass addition and heat addition accident analysis were subtracted from the revised LTOPS design basis (i.e., isothermal) P/T limit curve. The margin between the proposed TS PORV lift setpoint allowable value pressure and the temperature-dependent LTOPS setpoint pressure allowable value is verified to be positive at each temperature. The results of the LTOPS margin assessment performed using this methodology are presented in Appendix B. Note in Appendix B that the variable MULT represents a multiplier on the design basis isothermal curve. MULT has been set to a value of 1.0 since the K_{1C} stress intensity formulation requires that 100% of the isothermal P/T limit curve is used instead of 110% of the isothermal P/T curve as allowed when using K_{1A}.

LTOPS T_{enable} is the temperature below which LTOPS must be enabled. The LTOPS T_{enable} value was determined using ASME Section XI (specifically, the features of ASME Code Case N-641, Reference 10, included in ASME Section XI) and is calculated as the sum of the following:

- 238.2°F, the ¼-T RT_{NDT},
- 29.7°F, the margin required by ASME Code Section XI for plant specific applications,
- 15°F, the margin for the temperature lag between the quarter-thickness vessel location and the coolant temperature during a 60°F/hr heatup (i.e., 60°F/hr heatup data from Reference 11, attached as Appendix E) provided a value of 14.2°F at 320°F; conservatively rounded to 15°F to account for a higher LTOPS T_{enable}),
- 20°F, the temperature measurement instrument uncertainty, and
- 25°F, the estimated temperature difference between the cold leg and the belt-line materials during a cooldown with natural circulation.

3.4 Results

3.4.1 Revised P/T Limit Curves

As described in Section 3.2.1, the unadjusted P/T limit curves, including the LTOPS design basis P/T limit curve (i.e., the "steady state", "isothermal", or "0°F/hr cooldown" curve), were modified to account for pressure and temperature measurement uncertainty, and for the pressure difference between the point of measurement (RCS hot leg) and the point of interest (the reactor vessel beltline). The resulting proposed revised Surry Units 1 and 2 Technical Specification P/T limit curves are presented in Appendix A.

3.4.2 Revised LTOPS Setpoint Allowable Values

The Narrow Range RCS pressure measurement channel feeds the logic for opening and closing the pressurizer PORV at conditions during which the LTOPS system is enabled (i.e., temperatures that are below the LTOPS T_{enable}).

The Surry TS LTOPS setpoint consists of the following variables:

- LTOPS PORV Setpoint
- LTOPS Tenable (Derived in Section 3.4.3)

LTOPS T_{enable} is the temperature below which LTOPS must be enabled. For temperatures above T_{enable}, adequate overpressure protection is provided by the pressurizer safety valves (PSVs). In addition to the setpoint described above, the LTOPS PORV bistables are set in a staggered fashion for each of the two pressurizer PORVs. The staggered bistable control setpoints for each PORV avoids simultaneous PORV lift. In addition, for Surry Units 1 and 2, the stagger is also required as the RCS pressure measurement for one PORV is received from the RCS hot leg and the other PORV receives RCS pressure information from the pressurizer. The pressure difference between the RCS hot leg and the pressurizer is approximately 10 psig. Note that the PORV bistable that takes pressure information from the pressurizer is set 20 psig lower than the PORV bistable that obtains pressure information from the RCS hot leg. The value of 20 psig provides sufficient margin for the pressure measurement difference between the hot leg and the pressurizer and sufficient margin to provide the necessary staggering to prevent simultaneous opening of both PORVs. For ease of comparison with Reference 1, only the higher and hence lower margin PORV setpoint (i.e., that controlled by the RCS hot leg pressure measurement) is described below.

The results of calculations performed for the LTOPS margin assessment outlined in Section 3.3 are presented in Appendix B. As Appendix B demonstrates, the revised Surry Units 1 and 2 TS LTOPS setpoint provides bounding protection for 100% of the proposed revised design basis isothermal curve under postulated mass addition and heat addition accident conditions. The analysis includes consideration of pressure and temperature uncertainties, as well as the pressure difference between the point of measurement (RCS hot leg) and the point of interest (reactor vessel beltline). The design basis P/T limit curves are based on a ¼-T RT_{NDT} of 238.2°F, which conservatively bounds the most limiting ¼-T RT_{NDT} values at cumulative core burnups of 47.6 EFPY and 48.1 EFPY for Surry Units 1 and 2, respectively. Therefore, the revised Surry Units 1 and 2 LTOPS setpoint is concluded to be conservative for cumulative core burnups up to 47.6 EFPY and 48.1 EFPY for Surry Units 1 and 2 respectively.

The proposed revised TS LTOPS setpoint (valid for both Surry Units 1 and 2) is shown below. The development of the bistable control setpoints will be performed in a manner to provide for additional margin to this value to accommodate postulated setpoint drift between periodic calibrations.

Surry Unit 1 and 2 Technical Specification LTOPS Setpoint	≤395 psig @ ≤350°F

3.4.3 Revised LTOPS T_{enable}

As described in Section 3.3, the temperature below which LTOPS must be enabled is calculated as the summation of various components. Below is the equation in algebraic form:

LTOPS T_{enable}(°F) = RT_{NDT}(1/4-T) + 29.7°F + 15°F[ΔT(1/4-T)] + 20°F(Temperature Measurement Uncertainty) + 25°F (natural circulation bias)

Using a limiting ¼-T RT_{NDT} of 238.2°F, which corresponds to operation for 47.6 EFPY for Surry Unit 1 and 48.1 EFPY for Surry Unit 2, a minimum value for LTOPS T_{enable} of 327.9°F is required. To provide additional conservatism, and to avoid unnecessary setpoint and procedure changes, the proposed value for LTOPS T_{enable} will remain at 350°F for Surry Units 1 and 2.

3.4.4 Administrative RCS Cooldown Rate Limit for Surry Units 1 and 2

While the maximum allowable RCS cooldown rate assumed in the development of the P/T limit curves is 100°F/hr, a 50°F/hr administrative RCS cooldown rate is currently in effect as described in Reference 1. The administrative limit was established to ensure the adequacy of the P/T limits for non-linear cooldown ramp rates (i.e., short duration temperature changes of limited magnitude that may occur during normal operation, but which may result in calculated cooldown rates in excess of the limits prescribed in the Technical Specifications). This section addresses use of a 75°F/hr administrative cooldown rate limit.

The concern presented by an increase in the allowable administrative RCS cooldown rate is related to the operator's ability to control cooldown rate. It can be reasonably estimated that cooldown rate can be controlled to within 25°F/hr. While it is possible to have short duration changes of limited magnitude that can exceed this rate, such changes only produce small changes in overall metal temperature, and have no significant effect on the applied stress intensity at the assumed crack tip. Studies of the effects of "step changes" in cooldown rate suggest that more restrictive limits may be appropriate if cooldown rates are not held constant at rates less than analyzed. However, on the basis of engineering judgement, small step changes (e.g., < 25°F) do not present a significant concern since the reactor vessel material would experience insignificant temperature change at the assumed ¼-T flaw location (i.e., limited contribution to reactor vessel stress). Increasing the administrative cooldown rate from 50°F/hr to 75°F/hr continues to provide adequate margin to the analyzed rate to accommodate small unanticipated changes in cooldown rate due to short duration temperature changes of limited magnitude.

3.4.5 RT_{PTS} Screening

Reference 7 stated that the limiting material with respect to PTS screening was the Surry Unit 1 Longitudinal Weld L2, SA-1526. The information in Reference 7 demonstrated that the limiting RTPTS value of 268.5°F represents the end-of-license-renewal limiting material for both Surry Units 1 and 2 (Unit 1 Longitudinal Weld L2, SA-1526) corresponding to a cumulative core burnup of 47.6 EFPY for Surry Unit 1 and 48.1 EFPY for Surry Unit 2. Currently available material surveillance data has not changed this conclusion.

4.0 Changes to Surry Units 1 and 2 Technical Specifications

The following specific changes to the Surry Units 1 and 2 Technical Specifications are proposed:

- **Technical Specification 3.1.B:** Figures 3.1-1 and 3.1-2 and the cumulative core burnup limits are being replaced by revised Figures 3.1-1 and 3.1-2, Reactor Coolant System Heatup and Cooldown Limitations. The proposed curves (provided in Appendix A) will be valid for both Surry Units 1 and 2. Note also that the axis labels have been clarified to identify the source of the parameter indication used (i.e., indicated wide range instrumentation). Basis changes reflect both the K_{1C} stress intensity formulation and the extended cumulative core burnup applicability limits.

- **Technical Specification 3.1.G.1.c(4):** This specification has been revised to reflect the proposed LTOPS setpoint value of 395 psig.

5.0 Significant Hazards Consideration Determination

Virginia Electric and Power Company (Dominion) has reviewed the requirements of 10 CFR 50.92, relative to the proposed change to the Surry Units 1 and 2 Technical Specifications, and determined that a Significant Hazards Consideration is not involved. The proposed change to the Surry Units 1 and 2 Technical Specifications modifies the

Reactor Coolant System (RCS) pressure/temperature (P/T) limit curves, LTOPS setpoint, and LTOPS T_{enable} value, and extends the cumulative core burnup applicability limits for these parameters. The proposed P/T limit curves, LTOPS setpoint, and LTOPS T_{enable} value are valid to cumulative core burnups of 47.6 EFPY and 48.1 EFPY for Surry Units 1 and 2, respectively.

The following is provided to support this conclusion that the proposed change does not create a significant hazards consideration.

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

   The proposed change modifies the Surry Units 1 and 2 RCS P/T limit curves, LTOPS setpoint, and LTOPS T_{enable} value and extends the cumulative core burnup applicability limits for these parameters. The allowable operating pressures and temperatures under the proposed RCS P/T limit curves are not significantly different from those allowed under the existing Technical Specification P/T limits. The revisions in the values for the LTOPS setpoint and LTOPS T_{enable} do not significantly change the plant operating space. No changes to plant systems, structures or components are proposed, and no new operating modes are established. The P/T limits, LTOPS setpoint, and T_{enable} value do not contribute to the probability of occurrence or consequences of accidents previously analyzed. The revised licensing basis analyses utilize acceptable analytical methods, and continue to demonstrate that established accident analysis acceptance criteria are met. Therefore, there is no increase in the probability or consequences of any accident previously evaluated.

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

   The proposed change modifies the Surry Units 1 and 2 RCS P/T limit curves, LTOPS setpoint, and LTOPS T_{enable} value and extends the cumulative core burnup applicability limits for these parameters. The allowable operating pressures and temperatures under the proposed RCS P/T limit curves are not significantly different from those allowed under the existing Technical Specification P/T limits. No changes to plant systems, structures or components are proposed, and no new operating modes are established. Therefore, the proposed changes do not create the possibility of any accident or malfunction of a different type previously evaluated.

3. Does the change involve a significant reduction in the margin of safety?

   The proposed revised RCS P/T limit curves, LTOPS setpoint, and LTOPS T_{enable} value analysis bases do not involve a significant reduction in the margin of safety for these parameters. The proposed revised RCS P/T limit curves are valid to cumulative core burnups of 47.6 EFPY and 48.1 EFPY for Surry Units 1 and 2, respectively. The proposed revised LTOPS setpoint and T_{enable} analyses support these same cumulative core burnup limits. The proposed revised RCS P/T limit curves utilize ASME Code Section XI, which supports use of a conservative but less restrictive stress intensity formulation (K_{IC}). The proposed extension of the cumulative core burnup applicability limits along with a small increase in the LTOPS PORV setpoint is accommodated by the margin provided by ASME Code Section XI. The analyses demonstrate that established analysis acceptance criteria continue to be met. Specifically, the proposed 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. Therefore, the proposed change does not result in a significant reduction in margin of safety.

6.0 Environmental Assessment

The proposed Technical Specification (TS) change to the Reactor Coolant System (RCS) pressure/temperature (P/T) limit curves, LTOPS setpoint, and LTOPS enable temperature (T_{enable}) value and the extended cumulative core burnup applicability limits for these parameters meet the eligibility criteria for categorical exclusion from an environmental assessment set forth in 10 CFR 51.22(c)(9), as discussed below:

(i) The license condition involves no Significant Hazards Consideration.

As discussed in the evaluation of the Significant Hazards Consideration above, the proposed change to the RCS P/T limit curves, LTOPS setpoint, and LTOPS T_{enable} value for Surry will not involve a significant increase in the probability or consequences of an accident previously evaluated. The possibility of a new or different kind of accident from any accident previously evaluated is also not created, and the proposed change does not involve a significant reduction in a margin of safety. Therefore, the proposed change to the RCS P/T limit curves, LTOPS setpoint, and LTOPS T_{enable} value meet the requirements of 10 CFR 50.92(c) and do not involve a 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 allowable operating pressures and temperatures under the proposed RCS P/T limit curves are not significantly different from those allowed under the existing Technical Specification P/T limits. No changes to plant systems, structures or components are proposed, and no new operating modes are established. Therefore, the proposed change to the RCS P/T limit curves, LTOPS setpoint, and LTOPS T_{enable} will not significantly change the types, or significantly increase the amounts, of effluents that may be released offsite.

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

The proposed change modifies the Surry Units 1 and 2 RCS P/T limit curves, LTOPS setpoint, and LTOPS T_{enable} value, and extends the cumulative core burnup applicability limits for these parameters. The allowable operating pressures and temperatures under the proposed RCS P/T limit curves are not significantly different from those allowed under the existing Technical Specification P/T limits. No changes to plant systems, structures or components are proposed, and no new operating modes are established. In addition, the supporting analyses for the proposed changes continue to provide acceptable margin to vessel fracture under both normal operation and LTOPS design basis (mass addition and heat addition) accident conditions. Therefore, the proposed changes will not increase radiation levels compared to the existing Technical Specification P/T limits, LTOPS setpoint, and LTOPS T_{enable} value, so individual and cumulative occupational exposures are unchanged.

Based on the above, the proposed changes do not have a significant effect on the environment, and meet the criteria of 10 CFR 51.22(c)(9). Therefore, the proposed Technical Specification change qualifies for a categorical exclusion from a specific environmental review by the Commission, as described in 10 CFR 51.22.

7.0 Updates to the Reactor Vessel Integrity Database (RVID)

Table 1 of Appendix C of this submittal contains the RVID update based on current material properties basis (i.e., supporting operation for 47.6 EFPY and 48.1 EFPY for Surry Units 1 and 2, respectively).

In addition, Table 2 of Appendix C contains proposed alternate material properties basis that is developed from BAW-2308, Revision 1 (Reference 12). Reference 12 establishes revised (i.e., reduced) initial RT_{NDT} values for the Linde 80 weld heat materials. Revised initial RT_{NDT} values could be used in the future by Dominion to make various plant safety improvements (i.e., reduced probability of undesired PORV lifts during reactor coolant pump startups). An exemption request is provided in Attachment 4. Following NRC approval of this exemption request, Table 2 of Appendix C would serve as the RVID update. Note that NRC approval of the exemption is not required for approval of the proposed TS change as the TS change is supported by the current material properties basis (Table 1 of Appendix C).

8.0 Conclusions

A change to the Surry Units 1 and 2 Technical Specifications is proposed to extend the cumulative core burnup applicability limit for the Surry Units 1 and 2 Technical Specification RCS P/T limits, LTOPS setpoint, and LTOPS T_{enable} value. This change has been developed using methodologies that comply with the applicable regulations (e.g., 10 CFR 50 Appendix G), industry codes (e.g., ASME Section XI), and previously approved methods. The proposed change to the Surry Units 1 and 2 Technical Specifications will continue to provide acceptable margin with respect to the prevention of reactor vessel brittle fracture. Therefore, the proposed change will not adversely impact safe operation of Surry Units 1 and 2.

9.0 References

1. Letter from J. P. O'Hanlon 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 Setpoints," dated June 8, 1995.

2. Letter from USNRC to J. P. O'Hanlon, "Exemption from Requirements of 10CFR50.60, Acceptance Criteria for Fracture Prevention for Light-Water Nuclear Power Reactors for Normal Operation, Surry Power Station, Units 1 and 2, (TAC NOS. M92537 and M92538)," dated October 31, 1995.

3. Letter from USNRC to J. P. O'Hanlon, "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)," dated December 28,1995.

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

5. WCAP-7723, "Surry Unit 1 Reactor Vessel Radiation Surveillance Program," dated July 1971.

6. WCAP-8085, "Surry Unit 2 Reactor Vessel Radiation Surveillance Program," dated June 1973.

7. Letter from L. N. Hartz to USNRC, "Virginia Electric and Power Company (Dominion), Surry and North Anna Power Stations Units 1 and 2, Response to Request for Supplemental Information License Renewal Applications," Serial No. 02-601, dated October 15,2002.

8. Letter from USNRC to D. A Christian, "License Renewal Safety Evaluation Report for North Anna, Units 1 and 2, and Surry, Units 1 And 2", Serial No. 02-709, November 5,2002.

9. Letter from D. A. Christian to USNRC, 'Virginia Electric and Power Company, Surry and North Anna Power Stations Units 1 and 2, License Renewal Applications - Submittal," Serial No. 01 -282, dated May 29,2001.

10. ASME Code Section XI, Code Case N-641, "Alternate Pressure-Temperature Relationship and Low Temperature Overpressure Protection System Requirements", dated January 17, 2000.

11. Letter VPA-03-193 from Westinghouse, "Thermal Stress Intensity Factors and Vessel Wall Temperatures for PT Curves from WCAP-15130, Revision 1," dated October 9, 2003.

12. BAW-2308, "Initial RTNDT of Linde 80 Weld Materials," Revision 1, dated August 2003.

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

14. Letter from L. N. Hartz to USNRC, 'Virginia Electric and Power Company (Dominion), North Anna Power Station Units 1 and 2, Request for Additional Information, Proposed Technical Specification Change Request, Reactor Coolant System Pressure/Temperature Limits, LTOPS Setpoints and LTOPS Enable Temperatures," Serial No. 04-380A, dated October 28,2004.

APPENDIX A

Pressure/Temperature Limit Curves
Surry Units 1 and 2

Table 1	Surry Units 1 and 2 Heatup Data with Margins of 20 Degrees F and 67 psi for Instrumentation Errors (WCAP-15130 Rev. 1) and 57 psi for Pressure Measurement Bias

Heatup Rate = 20 Deg. F/hr			Heatup Rate = 40 Deg. F/hr			Heatup Rate = 60 Deg. F/hr		
	Indicated Temperature (Deg. F)	Indicated Pressure (psig)		Indicated Temperature (Deg. F)	Indicated Pressure (psig)		Indicated Temperature (Deg. F)	Indicated Pressure (psig)
1	100	497	1	100	497	1	100	497
2	105	497	2	105	497	2	105	497
3	110	497	3	110	497	3	110	497
4	115	497	4	115	497	4	115	497
5	120	497	5	120	497	5	120	497
6	125	497	6	125	497	6	125	497
7	130	497	7	130	497	7	130	497
8	135	497	8	135	497	8	135	497
9	140	497	9	140	497	9	140	497
10	145	497	10	145	497	10	145	497
11	150	497	11	150	497	11	150	497
12	150	566	12	150	566	12	150	525
13	155	571	13	155	571	13	155	532
14	160	576	14	160	576	14	160	539
15	165	582	15	165	582	15	165	548
16	170	589	16	170	589	16	170	558
17	175	596	17	175	596	17	175	570
18	180	604	18	180	604	18	180	583
19	185	613	19	185	613	19	185	597
20	190	623	20	190	623	20	190	614
21	195	634	21	195	634	21	195	632
22	200	646	22	200	646	22	200	646
23	205	659	23	205	659	23	205	659
24	210	673	24	210	673	24	210	673
25	215	690	25	215	690	25	215	690
26	220	707	26	220	707	26	220	707
27	225	727	27	225	727	27	225	727
28	230	749	28	230	749	28	230	749
29	235	773	29	235	773	29	235	773
30	240	800	30	240	800	30	240	800
31	245	829	31	245	829	31	245	829
32	250	861	32	250	861	32	250	861
33	255	897	33	255	897	33	255	897
34	260	937	34	260	937	34	260	937
35	265	981	35	265	981	35	265	981
36	270	1029	36	270	1029	36	270	1029
37	275	1083	37	275	1083	37	275	1083
38	280	1142	38	280	1142	38	280	1142
39	285	1202	39	285	1201	39	285	1205
40	290	1268	40	290	1262	40	290	1260
41	295	1341	41	295	1328	41	295	1321
42	300	1421	42	300	1401	42	300	1388
43	305	1510	43	305	1482	43	305	1462
44	310	1608	44	310	1572	44	310	1544
45	315	1716	45	315	1671	45	315	1634
46	320	1836	46	320	1780	46	320	1734
47	325	1968	47	325	1900	47	325	1843
48	330	2115	48	330	2033	48	330	1965
49	335	2276	49	335	2180	49	335	2098
			50	340	2342	50	340	2246

Surry Units 1 and 2
Reactor Coolant System Heatup Limitations

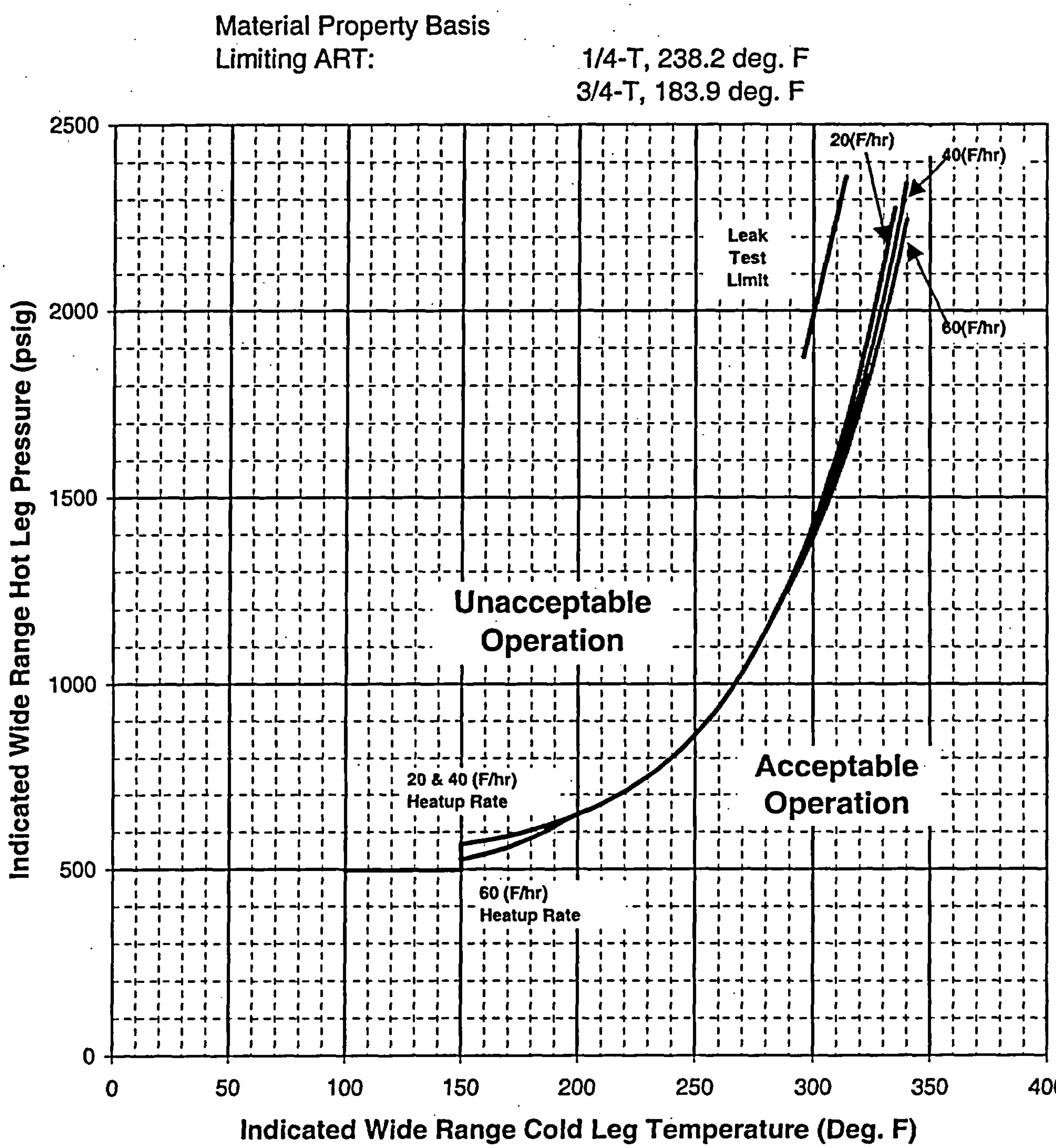

Figure 3.1-1: Surry Units 1 and 2 Reactor Coolant System Heatup Limitations (Heatup Rates up to 60 F/hr) Applicable for the first 47.6 EFPY for Unit 1 and 48.1 EFPY for Unit 2 (Including Margins for Instrumentation Errors)

Table2	Surry Units 1 and 2 Cooldown Data with Margins of 20 Degrees F and 67 psi for Instrumentation Errors (WCAP-15130 Rev. 1, Modified) and 57 psi for Pressure Measurement Bias

Cooldown Rate = 0 Deg. F/hr

	Indicated Temperature (Deg. F)	Indicated Pressure (psig)
1	100	497.00
2	105	497.00
3	110	497.00
4	115	497.00
5	120	497.00
6	125	497.00
7	130	497.00
8	135	497.00
9	140	497.00
10	145	497.00
11	150	497.00
12	150	566.22
13	155	571.08
14	160	576.45
15	165	582.38
16	170	588.94
17	175	596.19
18	180	604.20
19	185	613.06
20	190	622.84
21	195	633.66
22	200	645.61
23	205	658.82
24	210	673.42
25	215	689.55
26	220	707.38
27	225	727.09
28	230	748.87
29	235	772.94
30	240	799.54
31	245	828.94
32	250	861.43
33	255	897.34
34	260	937.02
35	265	980.88
36	270	1029.35
37	275	1082.91
38	280	1142.11
39	285	1207.54
40	290	1279.85
41	295	1359.76
42	300	1448.08
43	305	1545.68
44	310	1653.56
45	315	1772.77
46	320	1904.53
47	325	2050.14
48	330	2211.06

Cooldown Rate = 20 Deg. F/hr

	Indicated Temperature (Deg. F)	Indicated Pressure (psig)
1	100	490.72
2	105	492.47
3	110	494.41
4	115	496.59
5	120	497.00
6	125	497.00
7	130	497.00
8	135	497.00
9	140	497.00
10	145	497.00
11	150	497.00
12	150	520.51
13	155	525.62
14	160	531.27
15	165	537.56
16	170	544.51
17	175	552.25
18	180	560.79
19	185	570.29
20	190	580.79
21	195	592.44
22	200	605.31
23	205	619.60
24	210	635.39
25	215	652.89
26	220	672.23
27	225	693.67
28	230	717.36
29	235	743.60
30	240	772.61
31	245	804.73
32	250	840.23
33	255	879.53
34	260	922.97
35	265	971.04
36	270	1024.19

Cooldown Rate = 40 Deg. F/hr

	Indicated Temperature (Deg. F)	Indicated Pressure (psig)
1	100	443.45
2	105	445.16
3	110	447.08
4	115	449.27
5	120	451.71
6	125	454.47
7	130	457.56
8	135	461.03
9	140	464.89
10	145	469.23
11	150	474.05
12	150	474.05
13	155	479.45
14	160	485.44
15	165	492.13
16	170	499.56
17	175	507.83
18	180	517.01
19	185	527.23
20	190	538.55
21	195	551.14
22	200	565.08
23	205	580.57
24	210	597.72
25	215	616.76
26	220	637.83
27	225	661.21
28	230	687.08
29	235	715.76
30	240	747.50
31	245	782.67
32	250	821.59
33	255	864.69
34	260	912.39
35	265	965.20
36	270	1023.63

Table 2 (Cont'd)	Surry Units 1 and 2 Cooldown Data with Margins of 20 Degrees F and 67 psi for Instrumentation Errors (WCAP-15130 Rev. 1, Modified) and 57 psi for Pressure Measurement Bias

Cooldown Rate = 60 Deg. F/hr

	Indicated Temperature (Deg. F)	Indicated Pressure (psig)
1	100	395.16
2	105	396.84
3	110	398.77
4	115	400.97
5	120	403.47
6	125	406.30
7	130	409.50
8	135	413.12
9	140	417.17
10	145	421.74
11	150	426.85
12	150	426.85
13	155	432.58
14	160	438.97
15	165	446.13
16	170	454.10
17	175	463.01
18	180	472.91
19	185	483.95
20	190	496.22
21	195	509.87
22	200	525.03
23	205	541.89
24	210	560.59
25	215	581.36
26	220	604.40
27	225	629.97
28	230	658.31
29	235	689.75
30	240	724.58
31	245	763.20
32	250	805.98
33	255	853.39
34	260	905.89
35	265	964.06
36	270	1028.46

Cooldown Rate = 100 Deg. F/hr

	Indicated Temperature (Deg. F)	Indicated Pressure (psig)
1	100	295.37
2	105	297.06
3	110	299.04
4	115	301.35
5	120	304.02
6	125	307.11
7	130	310.63
8	135	314.65
9	140	319.22
10	145	324.40
11	150	330.25
12	150	330.25
13	155	336.85
14	160	344.26
15	165	352.60
16	170	361.94
17	175	372.40
18	180	384.10
19	185	397.18
20	190	411.77
21	195	428.05
22	200	446.18
23	205	466.38
24	210	488.86
25	215	513.87
26	220	541.66
27	225	572.57
28	230	606.88
29	235	645.00
30	240	687.31
31	245	734.27
32	250	786.36
33	255	844.15
34	260	908.23
35	265	979.29

Surry Units 1 and 2
Reactor Coolant System Cooldown Limitations

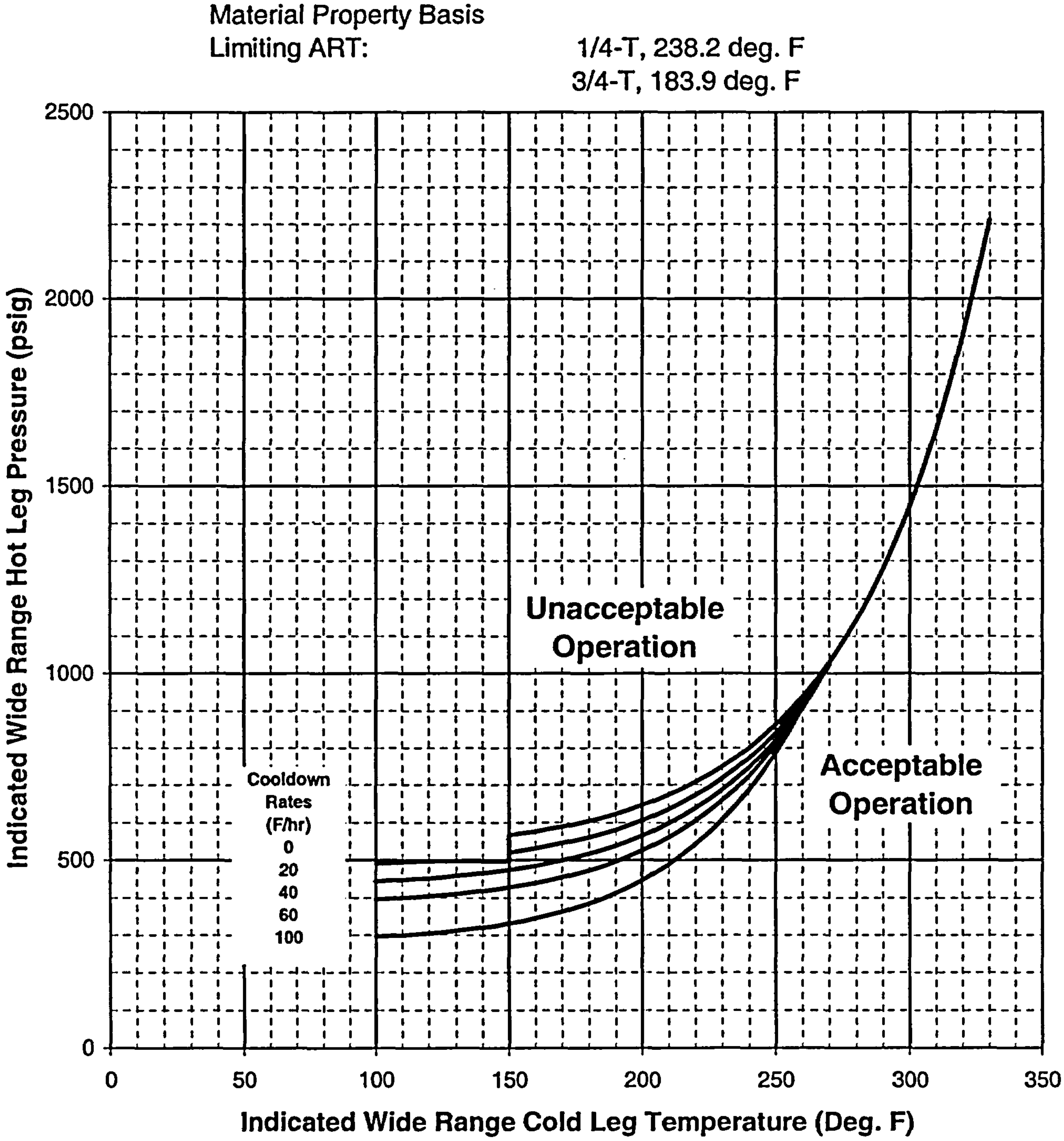

Figure 3.1-2: Surry Units 1 and 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100 F/hr) Applicable for the first 47.6 EFPY for Unit 1 and 48.1 EFPY for Unit 2 (Including Margins for Instrumentation Errors)

APPENDIX B

Surry Units 1 and 2
LTOPS Margin Assessment

Surry Units 1 and 2 LTOPS Margin Assessment

	Cooldown Rate = 0 Deg. F/hr WCAP-15130, R1 (w/o Uncs & Bias)			Cooldown Rate = 0 Deg. F/hr WCAP-15130, R1 (Uncs, Bias, and Mult)			PORV Setpoint Overshoot		0 deg. F/hr Curve minus PORV Setpoint Overshoot	
	Indicated Temperature (Deg. F)	Indicated Pressure (psig)		Indicated Temperature (Deg. F)	Indicated Pressure (psig)		Indicated Temperature (Deg. F)	Pressure (psi)	Indicated Temperature (Deg. F)	Indicated Pressure (psig)
1	80	621.00	1	100	539.00	1	100	142.00	100	397.00
2	85	621.00	2	105	539.00	2	105	142.00	105	397.00
3	90	621.00	3	110	539.00	3	110	142.00	110	397.00
4	95	621.00	4	115	539.00	4	115	142.00	115	397.00
5	100	621.00	5	120	539.00	5	120	142.00	120	397.00
6	105	621.00	6	125	539.00	6	125	142.00	125	397.00
7	110	621.00	7	130	539.00	7	130	142.00	130	397.00
8	115	621.00	8	135	539.00	8	135	142.00	135	397.00
9	120	621.00	9	140	539.00	9	140	142.00	140	397.00
10	125	621.00	10	145	539.00	10	145	142.00	145	397.00
11	130	621.00	11	150	539.00	11	150	142.00	150	397.00
13	135	695.08	13	155	613.08	13	155	130.90	155	482.18
14	140	700.45	14	160	618.45	14	160	130.90	160	487.55
15	145	706.38	15	165	624.38	15	165	130.90	165	493.48
16	150	712.94	16	170	630.94	16	170	130.90	170	500.04
17	155	720.19	17	175	638.19	17	175	130.90	175	507.29
18	160	728.20	18	180	646.20	18	180	130.90	180	515.30
19	165	737.06	19	185	655.06	19	185	130.90	185	524.16
20	170	746.84	20	190	664.84	20	190	130.90	190	533.94
21	175	757.66	21	195	675.66	21	195	130.90	195	544.76
22	180	769.61	22	200	687.61	22	200	130.90	200	556.71
23	185	782.82	23	205	700.82	23	205	101.80	205	599.02
24	190	797.42	24	210	715.42	24	210	101.80	210	613.62
25	195	813.55	25	215	731.55	25	215	101.80	215	629.75
26	200	831.38	26	220	749.38	26	220	101.80	220	647.58
27	205	851.09	27	225	769.09	27	225	101.80	225	667.29
28	210	872.87	28	230	790.87	28	230	101.80	230	689.07
29	215	896.94	29	235	814.94	29	235	101.80	235	713.14
30	220	923.54	30	240	841.54	30	240	101.80	240	739.74
31	225	952.94	31	245	870.94	31	245	101.80	245	769.14
32	230	985.43	32	250	903.43	32	250	101.80	250	801.63
33	235	1021.34	33	255	939.34	33	255	72.40	255	866.94
34	240	1061.02	34	260	979.02	34	260	72.40	260	906.62
35	245	1104.88	35	265	1022.88	35	265	72.40	265	950.48
36	250	1153.35	36	270	1071.35	36	270	72.40	270	998.95
37	255	1206.91	37	275	1124.91	37	275	72.40	275	1052.51
38	260	1266.11	38	280	1184.11	38	280	72.40	280	1111.71
39	265	1331.54	39	285	1249.54	39	285	72.40	285	1177.14
40	270	1403.85	40	290	1321.85	40	290	72.40	290	1249.45
41	275	1483.76	41	295	1401.76	41	295	72.40	295	1329.36
42	280	1572.08	42	300	1490.08	42	300	72.40	300	1417.68
43	285	1669.68	43	305	1587.68	43	305	58.30	305	1529.38
44	290	1777.56	44	310	1695.56	44	310	58.30	310	1637.26
45	295	1896.77	45	315	1814.77	45	315	58.30	315	1756.47
46	300	2028.53	46	320	1946.53	46	320	58.30	320	1888.23
47	305	2174.14	47	325	2092.14	47	325	58.30	325	2033.84
48	310	2335.06	48	330	2253.06	48	330	58.30	330	2194.76

Surry Units 1 and 2 LTOPS Margin Assessment

	PORV Setpoint Indicated Temperature (Deg. F)	Indicated Pressure (psig)		Margin (Positive = Acceptable) Indicated Temperature (Deg. F)	Pressure (psig)	Minimum Margin (psig)
1	100	395	1	100	2.00	2.00
2	105	395	2	105	2.00	
3	110	395	3	110	2.00	
4	115	395	4	115	2.00	
5	120	395	5	120	2.00	
6	125	395	6	125	2.00	
7	130	395	7	130	2.00	
8	135	395	8	135	2.00	
9	140	395	9	140	2.00	
10	145	395	10	145	2.00	
11	150	395	11	150	2.00	
13	155	395	13	155	87.18	
14	160	395	14	160	92.55	
15	165	395	15	165	98.48	
16	170	395	16	170	105.04	
17	175	395	17	175	112.29	
18	180	395	18	180	120.30	
19	185	395	19	185	129.16	
20	190	395	20	190	138.94	
21	195	395	21	195	149.76	
22	200	395	22	200	161.71	
23	205	395	23	205	204.02	
24	210	395	24	210	218.62	
25	215	395	25	215	234.75	
26	220	395	26	220	252.58	
27	225	395	27	225	272.29	
28	230	395	28	230	294.07	
29	235	395	29	235	318.14	
30	240	395	30	240	344.74	
31	245	395	31	245	374.14	
32	250	395	32	250	406.63	
33	255	395	33	255	471.94	
34	260	395	34	260	511.62	
35	265	395	35	265	555.48	
36	270	395	36	270	603.95	
37	275	395	37	275	657.51	
38	280	395	38	280	716.71	
39	285	395	39	285	782.14	
40	290	395	40	290	854.45	
41	295	395	41	295	934.36	
42	300	395	42	300	1022.68	
43	305	395	43	305	1134.38	
44	310	395	44	310	1242.26	
45	315	395	45	315	1361.47	
46	320	395	46	320	1493.23	
47	325	395	47	325	1638.84	
48	330	395	48	330	1799.76	

[redacted]

APPENDIX C

Material Properties Basis and RVID Update

Table 1 RVID Update Based on Current Material Properties

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)	Sigma(I)	Sigma(delta)	Margin	Inner Surf. ART or RT(PTS)	1/4-T ART*
122V109VA1	Nozzle Shell Forging	0.110	0.740	0.496	76.1	Tables	40	0.0	17.0	34.0	135.2	125.2
C4326-1	Intermediate Shell	0.110	0.550	5.400	73.5	Tables	10	0.0	17.0	34.0	148.2	140.5
C4326-2	Intermediate Shell	0.110	0.550	5.400	73.5	Tables	0	0.0	17.0	34.0	138.2	130.5
4415-1	Lower Shell	0.102	0.493	5.400	85.0	Surv. Data	20	0.0	8.5	17.0	157.4	148.6
4415-2	Lower Shell	0.110	0.500	5.400	73.0	Tables	0	0.0	17.0	34.0	137.5	129.8
J726/25017	Nozzle to Int Shell Circ Weld	0.330	0.100	0.496	152.0	Tables	0	20.0	28.0	68.8	191.1	171.0
SA-1585/72445	Int. to Low Sh. Circ (ID 40%)	0.220	0.540	4.700	131.4	Surv. Data	-5	19.7	28.0	68.5	246.2	231.7
SA-1650/72445	Int. to Low Sh. Circ (OD 60%)	0.220	0.540	4.700	131.4	Surv. Data	-5	19.7	28.0	68.5	246.2	231.7
SA-1494/8T1554	Int Shell Long. Welds L3 & L4	0.160	0.570	0.914	143.9	Tables	-5	19.7	28.0	68.5	203.7	183.9
SA-1494/8T1554	Lower Shell Long. Weld L1	0.160	0.570	0.790	143.9	Tables	-5	19.7	28.0	68.5	197.9	178.1
SA-1526/299L44	Lower Shell Long. Weld L2	0.340	0.680	0.790	220.6	Tables	-7	20.6	28.0	69.5	268.5	238.2

* 1/4-T ART value of 238.2 F was used in the determination of P/T limits

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 on 3/27/03).

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)	Sigma(I)	Sigma(delta)	Margin	Inner Surf. ART or RT(PTS)	1/4-T ART*
123V303VA1	Nozzle Shell Forging	0.110	0.720	0.471	75.8	Tables	30	0.0	17.0	34.0	123.9	114.0
C4331-2	Intermediate Shell	0.120	0.600	5.340	83.0	Tables	-10	0.0	17.0	34.0	141.5	132.7
C4339-2	Intermediate Shell	0.110	0.540	5.340	73.4	Tables	-20	0.0	17.0	34.0	117.9	110.2
C4208-2	Lower Shell	0.150	0.550	5.340	107.3	Tables	-30	0.0	17.0	34.0	155.8	144.5
C4339-1	Lower Shell	0.107	0.530	5.340	70.8	Tables	-10	0.0	17.0	34.0	124.2	116.8
L737/4275	Nozzle to Int Shell Circ Weld	0.350	0.100	0.471	160.5	Tables	0	20.0	28.0	68.8	195.7	174.6
R3008/0227	Int. to Lower Shell Circ Weld	0.187	0.545	5.340	132.4	Surv. Data	0	20.0	14.0	48.8	236.2	222.3
WF-4/8T1762	Int. Shell Long. L4 (ID 50%)	0.190	0.570	1.080	152.4	Tables	-5	19.7	28.0	68.5	219.1	198.0
SA-1585/72445	Int. Sh. L3 (100%), L4 (OD 50)	0.220	0.540	1.080	131.4	Surv. Data	-5	19.7	28.0	68.5	197.7	179.5
WF-4/8T1762	LS L2 (ID 63%), L1 (100)	0.190	0.570	1.080	152.4	Tables	-5	19.7	28.0	68.5	219.1	198.0
WF-8/8T1762	LS Long. Weld L2 (OD 37%)	0.190	0.570	1.080	152.4	Tables	-5	19.7	28.0	68.5	219.1	198.0

* 1/4-T ART value of 238.2 F was used in the determination of P/T limits

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 on 3/27/03).

Table 2 RVID Update Based on Material Properties as Modified by BAW-2308, Rev. 1

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)	Sigma(I)	Sigma(delta)	Margin	Inner Surf. ART or RT(PTS)	1/4-T ART*
122V109VA1	Nozzle Shell Forging	0.110	0.740	0.496	76.1	Tables	40	0.0	17.0	34.0	135.2	125.2
C4326-1	Intermediate Shell	0.110	0.550	5.400	73.5	Tables	10	0.0	17.0	34.0	148.2	140.5
C4326-2	Intermediate Shell	0.110	0.550	5.400	73.5	Tables	0	0.0	17.0	34.0	138.2	130.5
4415-1	Lower Shell	0.102	0.493	5.400	85.0	Surv. Data	20	0.0	8.5	17.0	157.4	148.6
4415-2	Lower Shell	0.110	0.500	5.400	73.0	Tables	0	0.0	17.0	34.0	137.5	129.8
J726/25017	Nozzle to Int Shell Circ Weld	0.330	0.100	0.496	152.0	Tables	0	20.0	28.0	68.8	191.1	171.0
SA-1585/72445	Int. to Low Sh. Circ (ID 40%)	0.220	0.540	4.700	131.4	Surv. Data	-73.2	11.9	28.0	60.8	170.3	155.9
SA-1650/72445	Int. to Low Sh. Circ (OD 60%)	0.220	0.540	4.700	131.4	Surv. Data	-73.2	11.9	28.0	60.8	170.3	155.9
SA-1494/8T1554	Int Shell Long. Welds L3 & L4	0.160	0.570	0.914	143.9	Tables	-67.9	20.1	28.0	68.9	141.3	121.4
SA-1494/8T1554	Lower Shell Long. Weld L1	0.160	0.570	0.790	143.9	Tables	-67.9	20.1	28.0	68.9	135.4	115.7
SA-1526/299L44	Lower Shell Long. Weld L2	0.340	0.680	0.790	220.6	Tables	-78.8	12.0	28.0	60.9	188.1	157.8

* 1/4-T ART value of 238.2 F was used in the determination of P/T limits

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 on 3/27/03).

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)	Sigma(I)	Sigma(delta)	Margin	Inner Surf. ART or RT(PTS)	1/4-T ART*
123V303VA1	Nozzle Shell Forging	0.110	0.720	0.471	75.8	Tables	30	0.0	17.0	34.0	123.9	114.0
C4331-2	Intermediate Shell	0.120	0.600	5.340	83.0	Tables	-10	0.0	17.0	34.0	141.5	132.7
C4339-2	Intermediate Shell	0.110	0.540	5.340	73.4	Tables	-20	0.0	17.0	34.0	117.9	110.2
C4208-2	Lower Shell	0.150	0.550	5.340	107.3	Tables	-30	0.0	17.0	34.0	155.8	144.5
C4339-1	Lower Shell	0.107	0.530	5.340	70.8	Tables	-10	0.0	17.0	34.0	124.2	116.8
L737/4275	Nozzle to Int Shell Circ Weld	0.350	0.100	0.471	160.5	Tables	0	20.0	28.0	68.8	195.7	174.6
R3008/0227	Int. to Lower Shell Circ Weld	0.187	0.545	5.340	132.4	Surv. Data	0	20.0	14.0	48.8	238.2	222.3
WF-4/8T1762	Int. Shell Long. L4 (ID 50%)	0.190	0.570	1.080	152.4	Tables	-67.9	20.1	28.0	68.9	156.7	135.6
SA-1585/72445	Int. Sh. L3 (100%), L4 (OD 50)	0.220	0.540	1.080	131.4	Surv. Data	-73.2	11.9	28.0	60.8	121.9	103.7
WF-4/8T1762	LS L2 (ID 63%), L1 (100)	0.190	0.570	1.080	152.4	Tables	-67.9	20.1	28.0	68.9	156.7	135.6
WF-8/8T1762	LS Long. Weld L2 (OD 37%)	0.190	0.570	1.080	152.4	Tables	-67.9	20.1	28.0	68.9	156.7	135.6

* 1/4-T ART value of 238.2 F was used in the determination of P/T limits

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 on 3/27/03).