

VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

September 19, 2019

10 CFR 50.90

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Serial No.: 19-374
NRA/GDM: R1
Docket Nos.: 50-280/281
License Nos.: DPR-32/37

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2
PROPOSED LICENSE AMENDMENT REQUEST
UPDATE OF REACTOR COOLANT SYSTEM HEATUP AND COOLDOWN
PRESSURE-TEMPERATURE LIMITATIONS FIGURES FOR SUBSEQUENT
LICENSE RENEWAL

Pursuant to 10 CFR 50.90, Virginia Electric and Power Company (Dominion Energy Virginia) requests amendments, in the form of changes to the Technical Specifications (TS) to Facility Operating License Numbers DPR-32 and DPR-37 for Surry Power Station (Surry) Units 1 and 2. The proposed change revises TS Figures 3.1-1 and 3.1-2, *Surry Units 1 and 2 Reactor Coolant System Heatup Limitations* and *Surry Units 1 and 2 Reactor Coolant System Cooldown Limitations*, respectively, to: 1) update the cumulative core burnup applicability limit (Effective Full Power Years; EFPY) from 48 to 68 EFPY, and 2) revise and relocate the limiting material property basis from the TS figures to the TS Basis. The cumulative core burnup applicability limit is also updated for the Low Temperature Overpressure Protection System (LTOPS) Setpoint and the LTOPS Enabling Temperature (T-enable) at Surry Units 1 and 2; however, no additional TS changes are required since these two values remain conservative with respect to the existing TS limits. The proposed change is being requested as a result of evaluations performed for the Surry Subsequent License Renewal (SLR) effort. Associated TS Basis changes are included for information.

Attachment 1 provides a discussion of the proposed change, and Attachment 2 includes an excerpt from the Surry SLR application dated October 15, 2018 (Serial No. 18-359) [ADAMS Accession No. ML18291A842] that provides the supporting technical evaluation for the proposed change. Marked-up and typed TS pages reflecting the proposed change are provided in Attachments 3 and 4, respectively.

We have evaluated the proposed amendment request 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 off-site 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

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or environmental assessment is needed in connection with the approval of the proposed change. The proposed TS change included in this license amendment request has been reviewed and approved by the Facility Safety Review Committee.

Dominion Energy Virginia requests approval of the proposed license amendment request by June 30, 2020, with a 60-day implementation period, to coincide with the issuance of the renewed Surry Units 1 and 2 SLR operating licenses.

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

Respectfully,



Mark D. Sartain
Vice President – Nuclear Engineering and Fleet Support

Commitments contained in this letter: None

Attachments:

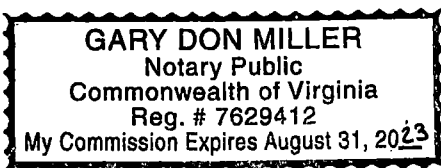
1. Discussion of Change
2. Supporting Technical Information from SLR Application (SLRA), Section 4.2, "Reactor Vessel Neutron Embrittlement Analysis"
3. Marked-up Technical Specifications and Bases Pages
4. Proposed Technical Specifications and Bases Pages

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Mr. Mark D. Sartain, who is Vice President – Nuclear Engineering and Fleet Support, 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 19th day of September, 2019.

My Commission Expires: August 31, 2023




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Surry Power Station

Attachment 1

DISCUSSION OF CHANGE

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

DISCUSSION OF CHANGE TABLE OF CONTENTS

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DISCUSSION OF CHANGE

1.0 SUMMARY DESCRIPTION

In accordance with the provisions of 10 CFR 50.90, Virginia Electric and Power Company (Dominion Energy Virginia) is submitting a license amendment request (LAR) to revise the Surry Power Station (Surry) Units 1 and 2 Technical Specifications (TS). Specifically, TS Figures 3.1-1 and 3.1-2, *Surry Units 1 and 2 Reactor Coolant System Heatup Limitations* and *Surry Units 1 and 2 Reactor Coolant System Cooldown Limitations*, respectively, are being revised to: 1) update the cumulative core burnup applicability limit (Effective Full Power Years; EFPY), and 2) revise and relocate the limiting material property basis from the TS figures to the TS Basis. The cumulative core burnup applicability limit is also updated for the Low Temperature Overpressure Protection System (LTOPS) Setpoint and the LTOPS Enabling Temperature (T-enable) at Surry Units 1 and 2. The proposed changes are being implemented as a result of evaluations performed for the Surry Subsequent License Renewal (SLR) application (SLRA) [Reference 6.1]. Associated TS Basis changes are included for information.

2.0 DETAILED DESCRIPTION

2.1 System Design and Operation

RCS components are designed to withstand the effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. The pressure/temperature (P-T) limitations curves limit the pressure and temperature changes during RCS heatup and cooldown within the design assumptions and the stress limits for cyclic operation.

The P-T limit curves are for heatup, cooldown, inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature. Each P-T limit curve defines an acceptable region for normal operation. The typical use of the curves is operational guidance during heatup or cooldown maneuvering when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The P-T limit curves establish operating limits that provide a margin to brittle failure of the reactor vessel (RV) and piping of the reactor coolant pressure boundary (RCPB). The RV is the component most subject to brittle failure, and the P-T limit curves' limits apply mainly to the RV. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

10 CFR 50, Appendix G, requires the establishment of P-T limits for specific material fracture toughness requirements of the RCPB materials and requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G. The neutron embrittlement effect on the material toughness is reflected by increasing the nil ductility reference temperature (RT_{NDT}) as exposure to neutron fluence increases.

The actual shift in the RT_{NDT} of the RV material is established periodically by removing and evaluating the irradiated RV material specimens, in accordance with ASTM E 185 and Appendix H of 10 CFR 50. The operating P-T limit curves are adjusted, as necessary, based on the evaluation findings and the recommendations of Regulatory Guide 1.99, *Radiation Embrittlement of Reactor Vessel Materials*.

The P-T limit curves are calculated using the most limiting value of RT_{NDT} corresponding to the limiting beltline region material for the RV. The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the RV wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The LTOPS controls RCS pressure at low temperatures so the integrity of the RCPB is not compromised. The LTOPS actuation logic monitors RCS pressure and determines when a condition is not acceptable. Each time the P-T limit curves are revised, the LTOPS setpoint and T-enable value must also be re-evaluated to ensure functional requirements can still be met to ensure the integrity of the RCPB.

2.2 Current Technical Specifications Requirements

The current Surry TS Figures 3.1-1 and 3.1-2 RCS Heatup and Cooldown P-T Limitations curves reflect the cumulative core burnup applicability limit of 48 EFPY and the material property bases associated with the period of extended operation for the renewed Surry Units 1 and 2 operating licenses. Surry TS 3.1.G.1 specifies an LTOPS arming temperature of 350° F and an LTOPS pressurizer PORV setpoint of ≤ 390 psig for operation.

2.3 Reason for the Proposed Change

By letter dated October 15, 2018, [Reference 6.1], Dominion Energy Virginia submitted an application to the US NRC for the subsequent license renewal of Renewed Facility Operating License Nos. DPR-32 and DPR-37 for Surry Units 1 and 2. The operating licenses will be extended from 60 years to 80 years thereby requiring a TS change for the Surry TS RCS heatup and cooldown P-T Limits figures, LTOPS Setpoint, and T-enable value from 48 EFPY to 68 EFPY cumulative core burnup applicability limit.

2.4 Description of Proposed Change

The proposed change updates TS Figures 3.1-1 and 3.1-2, *Surry Units 1 and 2 Reactor Coolant System Heatup Limitations* and *Surry Units 1 and 2 Reactor Coolant System Cooldown Limitations*, respectively, to reflect the increased cumulative core burnup applicability limit for RCS P-T Limits, LTOPS Setpoints, and LTOPS T-Enable values from 48 to 68 EFPY cumulative core burnup applicability limit. The proposed change also revises and relocates the limiting material property bases information from the TS figures to the TS 3.1.B Basis for Surry Units 1 and 2.

The proposed revisions to TS Figures 3.1-1 and 3.1-2 and the TS 3.1.B Basis are summarized as follows:

1. TS Figures 3.1-1 and 3.1-2

- The proposed change revises the existing TS Figure 3.1-1 and 3.1-2 titles, respectively, from:

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

and

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

to

*Surry Units 1 and 2 Reactor Coolant System Heatup Limitations (Heatup Rates up to 60°F/hr) Applicable to **68 EFPY [BOLD emphasis added]***

and

*Surry Units 1 and 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable to **68 EFPY [BOLD emphasis added]***

- The proposed change revises and relocates to the TS 3.1 Basis the existing Material Property Basis information contained in the informational block located above each curve. (See Item 3 below.) The revised information includes the Limiting Material and Limiting Adjusted Reference Temperature (ART) values associated with plant operation up to the SLR cumulative core burnup applicability limit of 68 EFPY, as opposed to the existing limit of 48 EFPY associated with the first license renewal for Surry Units 1 and 2.

2. LTOPS Setpoint and LTOPS T-enable value

The LTOPS Setpoint and LTOPS T-enable for Surry Units 1 and 2 were also evaluated for cumulative core burnups up to 68 EFPY. It was determined that the existing LTOPS setpoint and the T-enable value remain valid. The evaluation determined an LTOP enabling temperature of 283°F and LTOP PORV setpoint of 399.6 psig. TS 3.1.G.1 requires the system to be operable (i.e., arming temperature) when the RCS temperature is 350°F and specifies a PORV lift setting of ≤ 390 psig. Thus, the existing TS values, which are also reflected in UFSAR Section 4.3.4, are bounding and require no revision.

3. TS Basis

- The proposed TS 3.1.B Basis Insert for SLR is as follows:

The technical basis for the data points and the associated Adjusted Reference Temperature (ART) values used to generate the heatup and cooldown curves is provided in WCAP-14177 (Reference 2) and were determined to be applicable to the 48 EFPY period of extended operation under first license renewal. The associated ART values used to calculate the heatup and cooldown curves provided in WCAP-14177 are based on the Surry Unit 1 Intermediate to Lower Shell Circumferential Weld:

*1/4-T, 228.4°F, and
3/4-T, 189.5°F*

The heatup and cooldown curves for operation through 48 EFPY were based upon the K_{IR} methodology. These heatup and cooldown curves were subsequently evaluated using the K_{IC} methodology for Subsequent License Renewal (SLR) at 68 EFPY in WCAP-18243-NP (Reference 3).

The limiting reactor vessel materials at 68 EFPY were determined to be the Surry Unit 1 Lower Shell Longitudinal Weld L2 at 1/4-T and the Surry Unit 2 Intermediate to Lower Shell Circumferential Weld at 3/4-T. The associated ART values calculated at 68 EFPY are:

*1/4-T, 219.4 °F, and
3/4-T, 179.8 °F*

The data points and the associated ART values used to generate the heatup and cooldown curves in TS Figures 3.1-1 and 3.1-2, respectively, are conservative based upon use of the K_{IC} methodology. Therefore, the heatup and cooldown curves did not require revision as a result of SLR. However, the fluence applicability is updated from 48 EFPY to 68 EFPY.

- The following two items are being added to the TS 3.1 Basis References list:
 2. WCAP-14177, "Surry Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," (October 1994)
 3. WCAP-18243-NP, Rev. 3, "Surry Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," (January 2019)

3.0 TECHNICAL EVALUATION

RCS heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} corresponding to the limiting material in the beltline region of the RV. The most limiting RT_{NDT} of the material in the core region (beltline) of the RV is determined by using the unirradiated RV material fracture toughness properties and estimating the irradiation induced shift (ΔRT_{NDT}). RT_{NDT} increases as the material is exposed to fast neutron irradiation; therefore, to find the most limiting core region (beltline) RT_{NDT} at any time, ΔRT_{NDT} due to the neutron radiation exposure associated with that time must be added to the original unirradiated RT_{NDT} . Using the Adjusted Reference Temperature (ART) values, P-T limit curves are determined in accordance with the requirements of 10 CFR Part 50, Appendix G, as augmented by the ASME Code, Section XI, Appendix G.

The current P-T limits included in the Surry Units 1 and 2 TS are based on the K_{Ia} methodology and the latest fluence data through 48 EFPY. K_{Ia} is the crack arrest fracture toughness, which is the critical value of the stress intensity factor (K_I) for crack arrest as a function of temperature.

RV nozzle materials were evaluated in WCAP-18242-NP [Reference 6.3] at 48 EFPY and 68 EFPY. The evaluated nozzle forging materials are documented in SLRA Tables 4.2.4-1, 4.2.4-3, 4.2.4-5, and 4.2.4-7. The nozzle materials were assigned the fluence values at the postulated 1/4T flaw location for each specific nozzle in Table 4.2.1-1 and Table 4.2.1-2. Thus, Unit 1 Inlet Nozzle 1, Unit 2 Inlet Nozzle 1, and Outlet Nozzle 3 have neutron fluence values greater than 1.0×10^{17} n/cm² ($E > 1.0$ MeV) at 68 EFPY. To fully assess the Units 1 and 2 P-T limit curves applicability to 68 EFPY, a nozzle corner fracture mechanics analysis was completed for the nozzle materials. These nozzle P-T limit curves were generated and compared to the beltline P-T limit curves in TS to ensure the beltline curves are bounding. The detailed nozzle forging fracture mechanics evaluation and comparison to the applicable RV beltline P-T limit curves were documented in WCAP-18243-NP [Reference 6.4]. The current TS beltline P-T curves were confirmed to remain more limiting than the nozzle P-T curves through 68 EFPY.

The development of the current P-T limit curves for normal heatup and cooldown of the RCS for Units 1 and 2 was documented in WCAP-14177 [Reference 6.2]. The existing P-T limit curves are based on the K_{Ia} methodology and the limiting beltline material ART values, which are influenced by both the fluence and the initial material properties of that material. The Units 1 and 2 P-T limit curves were developed by calculating ART values utilizing the vessel fluence at the clad/base metal interface corresponding to each RV material. Since the development of the curves, the applicability of the curves has been extended and the fluence values and initial material properties used to calculate ART values have been updated.

The K_{Ic} methodology was used to confirm the applicability of the P-T limit curves developed based on WCAP-14177. K_{Ic} is the plane strain fracture toughness, which is the material toughness property measured in terms of the stress intensity factor, KI, that will lead to nonductile crack propagation.

The limiting RV material ART values with consideration of the updated 68 EFPY fluence values, revised Position 2.1 chemistry factor values, and updated initial RT_{NDT} values, must be shown to be less than or equal to the limiting beltline material ART values used in development of the P-T limit curves contained in WCAP-14177 and the Units 1 and 2 TS. The Regulatory Guide 1.99 methodology [Reference 6.5] was used along with the surface fluence of Section 2 of WCAP-18242-NP to calculate ART values for the Units 1 and 2 RV materials at 48 EFPY and 68 EFPY.

Comparisons of the use of the K_{Ic} reference stress intensity factor instead of the older, more conservative K_{Ia} reference stress intensity factor were conducted to validate that the P-T limits for 48 EFPY are conservative for operation through the subsequent period of extended operation (i.e., 68 EFPY). The comparisons of the limiting ART values calculated as part of the RV integrity Time Limiting Aging Analysis (TLAA) evaluation, using updated fluence and initial material properties, to those used in calculation of the existing P-T limit curves are contained in Table 4.2.4-9 for Units 1 and 2. With the consideration of fluence projections, the applicability of the P-T limit curves in WCAP-14177 may be extended to 68 EFPY for the Units 1 and 2 cylindrical shell materials. Nozzle P-T limit curves were developed per WCAP-18243-NP and compared to the cylindrical shell beltline curves. ART values were generated without consideration of the methodology in TLR-RES/DE/CIB-2013-01, "Evaluation of the Beltline Region for Nuclear Reactor Pressure Vessels, U.S. NRC Technical Letter Report, Office of Nuclear Regulatory Research [RES]" [Reference 6.6].

The evaluations described in WCAP-18242 and WCAP-18243 have been used to prepare Section 4.2 of the Surry SLRA to support extension of the cumulative core burnup applicability limit from 48 EFPY to 68 EFPY for Surry Units 1 and 2. The discussions in WCAP-18242, WCAP-18243, and Section 4.2 of the Surry SLRA affirms the conservatism

of the existing RCS P-T Limits, LTOPS Setpoint, and LTOPS T-enable value for Surry Units 1 and 2 cumulative core burnups up to 68 EFPY.

The following subsections in Section 4.2 of the Surry SLRA (provided in Attachment 2) include detailed supporting information:

- Neutron Fluence Projections (Section 4.2.1)
- Upper-Shelf Energy (Section 4.2.2)
- Pressurized Thermal Shock (Section 4.2.3)
- Adjusted Reference Temperature (Section 4.2.4)
- Pressure-Temperature Limits (Section 4.2.5)
- Low Temperature Overpressure Protection Analyses (Section 4.2.6)

Per WCAP-18243-NP, the applicability of the RCS P-T limit curves may be extended through SLR because the current TS P-T limit curves bound the new P-T limit curves developed in WCAP-18243-NP regardless of the use of the TLR-RES/DE/CIB-2013-01 methodology. Per WCAP-18243-NP, the applicability of the current TS P-T limit curves may be extended through the subsequent period of extended operation.

In addition, the applicable RV flange and closure head initial RT_{NDT} values are bounding and the P-T limit curves flange notch requires no change or further consideration. Finally, the lowest service temperature requirements are not applicable to Surry Units 1 and 2 because the plants are Westinghouse-designed per ASME Code, Section III, and utilize stainless steel reactor coolant system piping.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements

Design Requirements: 10 CFR 50, Appendix A – General Design Criteria

The regulations in Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 establish minimum principal design criteria for water-cooled nuclear power plants, while 10 CFR 50 Appendix B and the licensee quality assurance programs establish quality assurance requirements for the design, manufacture, construction, and operation of structures, systems, and components. The current regulatory requirements of 10 CFR 50 Appendix A that are applicable to the Reactor Coolant Pressure Boundary (RCPB), including the Reactor Vessel, include: General Design Criteria (GDC) 14 (Reactor Coolant Pressure Boundary), GDC 31 (Fracture Prevention of Reactor Coolant Pressure Boundary), and GDC 32 (Inspection of Reactor Coolant Pressure Boundary).

During the initial plant licensing of Surry Units 1 and 2, it was demonstrated that the design of the RCPB, met the regulatory requirements in place at that time. The GDC included in

Appendix A to 10 CFR 50 did not become effective until May 21, 1971. The Construction Permits for Surry Units 1 and 2 were issued prior to May 21, 1971; consequently, Surry Units 1 and 2 were not subject to current GDC requirements (SECY-92-223, dated September 18, 1992). The following information demonstrates Surry Units 1 and 2 meet the intent of the GDC published in 1967 (Draft GDC). Specifically, Section 1.4 of the Surry Updated Final Safety Analysis Report (UFSAR) discusses Surry compliance with these criteria. The draft GDC associated with the RCPB are addressed below.

- Quality Standards (Criterion 1 - draft)

Those systems and components of reactor facilities that are essential to the prevention of accidents which could affect the public health and safety or to the mitigation of their consequences shall be identified and then designed, fabricated, and erected in accordance with quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. A showing of sufficiency and applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance levels used is required.

Design Conformance

Structures, systems, and components important to safety are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

The Quality Assurance Program was established to provide assurance that safety-related structures, systems, and components satisfactorily perform their intended safety functions.

- Reactor Coolant Pressure Boundary (Criterion 9 - draft)

The reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.

Design Conformance

The reactor coolant pressure boundary is designed, fabricated, and constructed to have an exceedingly low probability of gross rupture or significant uncontrolled leakage throughout its design lifetime. The reactor coolant system, in conjunction with its control and protective provisions, is designed to accommodate the system

pressures and temperatures attained under all expected modes of unit operation or anticipated system interactions, and to remain within the applicable code stress limits.

The fabrication of the components that constitute the pressure-retaining boundary of the reactor coolant system is carried out in strict accordance with the applicable codes. In addition, there are areas where equipment specifications for reactor coolant system components are more restrictive than applicable codes.

The materials of construction of the pressure-retaining boundary of the reactor coolant system are protected by the control of coolant chemistry so as to prevent corrosion phenomena that might otherwise reduce the system structural integrity during its service lifetime.

- Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention (Criterion 34 - draft)

The reactor coolant pressure boundary shall be designed to minimize the probability of rapidly propagating type failures. Consideration shall be given (a) to the notch toughness properties of materials extending to the upper shelf of the Charpy transition curve, (b) to the state of stress of materials under static and transient loadings, (c) to the quality control specified for materials and component fabrication to limit flaw sizes, and (d) to the provisions for control over service temperature and irradiation effects which may require operational restrictions.

Design Conformance

The reactor coolant pressure boundary is designed and operated to reduce to an acceptable level the probability of a rapidly propagating failure. Consideration is given to (1) the provisions for control over service temperature and irradiation effects that may require operational restrictions, (2) the design and construction of the reactor pressure vessel in accordance with applicable codes, including those that establish the requirements for the absorption of energy within the elastic strain energy range and for the absorption of energy by plastic deformation, and (3) the design and construction of reactor coolant pressure boundary piping and equipment in accordance with applicable codes.

The reactor coolant pressure boundary is designed to reduce the probability of a rapidly propagating failure to an acceptable level. The fast neutron exposure of the core region of the reactor vessel changes the notch toughness of the vessel material. This change is indicated by the increase in the nil ductility transition temperature and allowance for it is made in the operating procedures by ensuring that the vessel is not subjected to full operating pressure until its temperature exceeds the design transition temperature, defined to be the nil ductility transition temperature plus a 60°F margin. The pressure during unit start-up and shutdown at temperatures below the nil ductility

transition temperature are maintained below the threshold of concern for safe operation.

The design transition temperature dictates the procedures to be followed in hydrostatic testing and in station operations to avoid excessive cold stress. The value of the design transition temperature is increased during the life of the station as required by the expected shift in the nil ductility transition temperature, which is confirmed by the experimental data obtained from irradiated specimens of reactor vessel materials during the unit lifetime.

All pressure-containing components of the reactor coolant system are designed, fabricated, inspected, and tested in conformance with the applicable codes.

- Reactor Coolant Pressure Boundary Brittle Fracture Prevention (Criterion 35 - draft)

Under conditions where reactor coolant pressure boundary system components constructed of ferritic materials may be subjected to potential loadings, such as a reactivity-induced loading, service temperature shall be at least 120° F above the nil ductility transition (NDT) temperature of the component material if the resulting energy release is expected to be absorbed by plastic deformation or 60° F above the NDT temperature of the component material if the resulting energy release is expected to be absorbed within the elastic strain range.

Design Conformance

For conditions under which reactor coolant pressure boundary system components constructed of ferritic materials may be subjected to potential loadings, such as a reactivity-induced loading, service temperatures shall be at least 120°F above the nil ductility transition temperature of the component material if the resulting energy release is expected to be absorbed by plastic deformation, or 60°F above the nil ductility temperature of the component material if the resulting energy release is expected to be absorbed within the elastic strain energy range.

Sufficient testing and analysis of materials used in reactor coolant system components are performed to ensure that the required nil ductility transition temperature limits specified in the criterion are met. Removable test capsules are installed in the reactor vessel and removed and tested at various times in the unit lifetime to determine the effects of the operation on system materials.

- Reactor Coolant Pressure Boundary Surveillance (Criterion 36 - draft)

Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime. For the reactor

vessel, a material surveillance program conforming with ASTM-E-185-66 shall be provided.

Design Conformance

Reactor coolant pressure boundary components have provisions for the inspection, testing, and surveillance of critical areas by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming to current applicable codes is provided.

The design of the reactor vessel and its arrangement in the system permit accessibility during service life to all internal surfaces of the vessel and to certain external zones such as the areas of the nozzle-to-piping welds and the top and bottom heads. The reactor arrangement within the containment provides sufficient space for the inspection of the external surfaces of the reactor coolant piping, except for the area of pipe within the primary shielding concrete.

The monitoring of the nil ductility transition temperature properties of the core region plates, forgings, weldments, and associated heat-treated zones is performed in accordance with ASTM E 185, *Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors*. Samples of reactor vessel plate materials are retained and cataloged in case future engineering development shows the need for further testing.

The material properties surveillance program includes not only the conventional tensile tests, but also tests of fracture mechanics specimens. The fracture mechanics specimens are the wedge-opening-loading-type specimens. The observed irradiation shifts in the nil ductility transition temperature of the core region materials are used to confirm the calculated limits to start-up and shutdown transients.

Quality Assurance

Quality assurance criteria provided in 10 CFR Part 50, Appendix B, that apply to the reactor coolant pressure boundary and reactor vessel include: Criteria III, V, XI, XVI, and XVII. Criteria III and V require measures to ensure that applicable regulatory requirements and the design basis, as defined in 10 CFR 50.2, "Definitions," and as specified in the license application, are correctly translated into controlled specifications, drawings, procedures, and instructions. Criterion XI requires a test program to ensure that the subject systems will perform satisfactorily in service and requires that test results shall be documented and evaluated to ensure that test requirements have been satisfied. Criterion XVI requires measures to ensure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances, are promptly identified and corrected, and that significant conditions

adverse to quality are documented and reported to management. Criterion XVII requires maintenance of records of activities affecting quality.

4.2 No Significant Hazards Consideration Analysis

Virginia Electric and Power Company (Dominion Energy Virginia) proposes a change to the Surry Power Station (Surry) Units 1 and 2 Technical Specifications (TS) to modify the Reactor Coolant System (RCS) Heatup and Cooldown Limitations, Figures 3.1-1 and 3.1-2, respectively. The proposed change revises the two TS figures to reflect a cumulative core burnup applicability limit of 68 EFPY for the RCS Pressure/Temperature (P-T) Limits. The cumulative core applicability limit is also increased to 68 EFPY for the Low Temperature Overpressure Protection System (LTOPS) Setpoint and LTOPS Enabling Temperature (T-enable). In addition, the material properties bases currently included on the TS figures will be relocated to the TS 3.1 Basis. In accordance with the criteria set forth in 10 CFR 50.92, Dominion Energy Virginia 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. Does the change involve a significant increase in the probability of consequence of an accident previously evaluated?

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 68 EFPY. The existing Surry TS RCS P-T Limits, LTOPS Setpoint, and T-enable value remain valid and conservative for cumulative core burnup up to 68 EFPY, thus increasing the cumulative core burnup applicability limit for RCS P-T Limits, LTOPS Setpoints and LTOPS T-enable to 68 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 is accomplished through application of improved analytical margins using the K_{IC} reference stress intensity factor, instead of the older, more conservative K_{Ia} reference stress intensity factor. Dominion Energy Virginia assessed the effect of 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 burnup to 68 EFPY. No changes to plant systems, structures or components are proposed, and no new operating modes are established. 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?

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 Setpoints, and LTOPS T-enable value remain valid and conservative for cumulative core burnups up to 68 EFPY. Analysis supporting the increased cumulative core burnup applicability limit was performed in accordance with applicable regulatory guidance and confirms that design functions (i.e., ensuring that combined pressure and thermal stresses under normal operating 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. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The increased cumulative core burnup applicability limit is accomplished through application of improved analytical margins provided by using the K_{IC} reference stress intensity factor, instead of the older, more conservative K_{Ia} reference stress intensity factor. Dominion Energy Virginia assessed the effect of the use of the analytical margins and determined that the existing TS P-T Limits, LTOPS Setpoint, and LTOPS T-enable value governing reactor vessel integrity remain valid and conservative for cumulative core burnups up to 68 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 TS P-T Limits, 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 68 EFPY. Therefore, the proposed change does not result in a significant reduction in the margin of safety.

Based on the above, Dominion Energy Virginia 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 considerations" is justified.

4.3 Conclusion

Dominion Energy Virginia concludes, based on consideration discussed herein, that (1) there is reasonable assurance that the health and safety of the public will not be

endangered by the proposed change, (2) such activities will be conducted in accordance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL ASSESSMENT

A review has determined that the proposed amendment would not change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would not change an inspection or surveillance requirement. As such, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

- 6.1 Letter from Virginia Electric and Power Company to the US Nuclear Regulatory Commission dated October 15, 2018 (Serial No. 18-340), "Virginia Electric and Power Company, Surry Power Station Units 1 and 2, Application for Subsequent Renewed Operating Licenses." [ADAMS Accession No. ML18291A842]
- 6.2 WCAP-14177, "Surry Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," October 1994.
- 6.3 WCAP-18242-NP, Revision 2, "Surry Units 1 and 2 Time-Limited Aging Analysis on Reactor Vessel Integrity for Subsequent License Renewal," July 2018.
- 6.4 WCAP-18243-NP, Rev. 3, "Surry Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," January 2019.
- 6.5 Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.
- 6.6 TLR-RES/DE/CIB-2013-01, "Evaluation of the Beltline Region for Nuclear Reactor Pressure Vessels, U.S. NRC Technical Letter Report, Office of Nuclear Regulatory Research [RES]," November 14, 2014. [ADAMS Accession No. ML14318A177]

Attachment 2

**SUPPORTING TECHNICAL INFORMATION FROM SLR APPLICATION (SLRA),
SECTION 4.2, “REACTOR VESSEL NEUTRON EMBRITTLEMENT ANALYSIS”**

(*Excerpted from SLRA submitted by Virginia Electric and Power Company to the US NRC by letter dated October 15, 2018 [ADAMS Accession No. ML18291A842])

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

4.2 REACTOR VESSEL NEUTRON EMBRITTLEMENT ANALYSIS

10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Reactors for Normal Operation," requires that all light water reactors meet the fracture toughness, P-T limits, and materials surveillance program requirements for the reactor coolant pressure boundary as set forth in 10 CFR 50, Appendices G and H. The materials included in the surveillance capsule program remain unchanged for the subsequent period of extended operation based upon the provisions outlined in earlier versions of ASTM E185, "Standard Practice for Design of Surveillance Programs for Light-Water Moderated Nuclear Power Reactor Vessels" (Reference 4.8-1) that existed at the time of initial plant construction. 10 CFR 50.61 requires that all light water reactors meet the fracture toughness requirements for protection against pressurized thermal shock events. The *Reactor Vessel Material Surveillance* program is described in Section B2.1.19.

Inputs for reactor vessel (RV) integrity assessments are discussed in this section.

The best estimate copper (Cu) and nickel (Ni) chemical compositions for the Units 1 and 2 RV materials are presented in Table 4.2.2-1 through Table 4.2.2-4. The best estimate weight percent Cu and Ni values for the RV materials were reported in PWROG-16045-NP, "Determination of Unirradiated RT_{NDT} and Upper-Shelf Energy Values of the Units 1 and 2 Reactor Vessel Materials" (Reference 4.8-2) and were included in RV integrity evaluations as part of this TLAA effort.

Prior to updating the RV integrity assessments for the subsequent period of extended operation both the fluence projections and material properties were reviewed and updated by WCAP-18028-NP, "Extended Beltline Pressure Vessel Fluence Evaluations Applicable to Surry Units 1 & 2" (Reference 4.8-3), WCAP-18242-NP, "Surry Units 1 and 2 Time Limited Aging Analysis on Reactor Vessel Integrity for Subsequent License Renewal" (Reference 4.8-4) and PWROG-16045-NP. Revised initial material properties, including chemistry factors and fluence projections, through 68 EFPY are included in and Table 4.2.3-2 for Units 1 and 2 respectively.

The neutron fluence axial boundary of the 1.0×10^{17} n/cm² fluence threshold is depicted in Figures 4.2.2-1 and 4.2.2-2 for Units 1 and 2 respectively. The configuration of the RVs, including the weld identification (ID) numbers, is illustrated in Figures 4.2.2-3 and 4.2.2-4 for Units 1 and 2, respectively.

Reactor vessel integrity assessments are performed for both the beltline region (identified in 10 CFR 50, Appendix G) and extended beltline region (fluence values $>1.0 \times 10^{17}$ n/cm², E >1 MeV).

The beltline region is the region of the RV (shell material, including welds, heat-affected zones, and plate or forgings) that directly surrounds the effective height of the active core and the adjacent regions of the RV that are predicted to experience sufficient neutron irradiation damage to be

considered in the selection of the most limiting material with regard to radiation damage during the licensed period.

The extended beltline means the region of the RV (shell material, including welds, heat-affected zones, and plate or forgings) adjacent to the beltline region that will have associated fluence values projected to exceed 1.0×10^{17} n/cm² during the subsequent period of extended operation.

The ferritic materials of the RV are subject to embrittlement due to high energy ($E > 1.0$ MeV) neutron exposure. Embrittlement means the material has lower toughness (i.e., will absorb less strain energy during crack propagation or rupture), thus allowing a crack to propagate more easily under thermal and pressure loading. Neutron embrittlement analyses account for the reduction in fracture toughness associated with the cumulative neutron fluence. Because these neutron embrittlement analyses use a fluence assumption based on the plant's current operating term, they are identified as time-limited aging analyses.

Fracture toughness (indirectly measured in foot-pounds of absorbed energy in a Charpy impact test) is temperature dependent in ferritic materials. An initial nil-ductility reference temperature (RT_{NDT}) is associated with the transition from ductile to brittle behavior and is determined for vessel materials through a combination of Charpy and drop-weight testing. Toughness increases with temperature up to a maximum value called the "upper-shelf energy," or USE. Neutron embrittlement results in the USE decrease of RV steels. This means that RV materials may no longer behave in a ductile manner at postulated plant operating temperatures. For beltline materials the limit for initial USE is 75 ft-lbs. The limit for reduced USE of beltline materials following irradiation is 50 ft-lbs. The material outside the beltline was originally qualified using the requirements of the codes in effect at the time of the initial design and fabrication of the RVs for Units 1 and 2, which were a minimum Charpy impact energy value of 30 ft-lbs at 10°F as specified by ASTM E208, "Standard Test Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels" (Reference 4.8-5) and required by ASME Code, Section III, "Rules for Construction of Nuclear Facility Components" (Reference 4.8-6).

To reduce the potential for brittle fracture during RV operation, changes in material toughness as a function of neutron radiation exposure (fluence) are accounted for during development of operating pressure temperature (P-T) limits that are included in the Technical Specifications. The P-T limits account for the decrease in material toughness of RV materials during plant operation. Since the cumulative neutron fluence will increase during the subsequent period of extended operation, a review is needed to determine if additional components require evaluation for neutron embrittlement.

10 CFR 50.61 requirements for pressurized thermal shock events specify screening criteria of 270°F for plates, forgings, and axial welds and 300°F for circumferential welds. The RT_{PTS} values have been projected through the subsequent period of extended operation.

USE and RT_{PTS} calculations are performed for each beltline and extended beltline material to determine if the components will continue to have adequate fracture toughness with the reduction in toughness resulting from exposure to the predicted neutron fluence. While the decrease in USE for materials in the extended beltline approaches (but remains greater than) 50 ft-lbs, as a conservative measure, an equivalent margins analysis has been performed for the inlet and outlet nozzle welds.

The NRC has approved use of revised initial (unirradiated) RT_{NDT} values and associated uncertainties for Linde 80 weld material. The NRC approved Topical Report BAW-2308 (Revision 1-A) in the "Final Safety Evaluation for Topical Report BAW-2308, Revision 1, 'Initial RT_{NDT} of Linde 80 Weld Materials'" (Reference 4.8-7). Table 3 of the Topical Report contains the revised initial reference temperature (IRT_{T0}) and initial margin (I) values for Linde 80 weld materials that are approved by the NRC for the purpose of RV material property determination.

P-T limit curves are generated to provide minimum temperature limits that must be satisfied during operations. The P-T limit curves are based upon the RT_{NDT} and ΔRT_{NDT} values computed for the licensed operating period along with appropriate margins.

The enabling temperature and LTOP setpoint are validated as they are impacted by fluence.

The RV material evaluations, calculated on the basis of neutron fluence, are part of the current licensing basis and support safety determinations. Therefore, these calculations have been identified as TLAAs.

The evaluations of TLAAs related to neutron embrittlement are described in the following subsections:

- Neutron Fluence Projections (Section 4.2.1)
- Upper-Shelf Energy (Section 4.2.2)
- Pressurized Thermal Shock (Section 4.2.3)
- Adjusted Reference Temperature (Section 4.2.4)
- Pressure-Temperature Limits (Section 4.2.5)
- Low Temperature Overpressure Protection Analyses (Section 4.2.6)

4.2.1 NEUTRON FLUENCE PROJECTIONS

TLAA Description:

Neutron fluence is the term used to represent the cumulative number of neutrons per square centimeter that contact the RV shell. The fluence projections that quantify the number of neutrons that contact these surfaces have been used as inputs to the neutron embrittlement analyses that evaluate the reduction of fracture toughness aging effect resulting from neutron irradiation and will be treated as a TLAA.

TLAA Evaluation:

Per NUREG-1766, "Safety Evaluation Report Related to the License Renewal of North Anna Power Station, Units 1 and 2, and Surry Power Station, Units 1 and 2" (Reference 4.8-8), RV beltline neutron fluence values applicable to the 60-year period of operation were calculated using the NRC approved VEP-NAF-3-A, "Virginia Power Reactor Vessel Fluence Methodology Topical Report" (Reference 4.8-9). The methodology described in that report was developed in accordance with Draft Regulatory Guide DG-1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" (Reference 4.8-10).

EFPY Projections

EFPY values for Unit 1 and 2 as of January 5, 2017 are as follows:

Unit 1	33.78 EPFY
Unit 2	33.69 EPFY

The first step in updating fluence projections for 80 years is to estimate the power history based upon actual unit operating history and a conservative capacity factor estimate for future cycles. Units 1 and 2 are licensed for 60 years of operation; therefore, with a 20-year license renewal, the subsequent license renewal term is 80 years.

The EPFY projections through the end of the subsequent period of extended operation for a unit is the sum of the accumulated EPFY and the projected future EPFY. EPFY at the end of 60 years of operation was calculated to be 48 EPFY, assuming a 95% capacity factor for cycles beyond Cycle 19 for Unit 1, and for cycles beyond Cycle 18 for Unit 2. An estimate of the EPFY at the end of 80 years of operation can be made conservatively assuming a 100% capacity factor for the 20-year subsequent period of extended operation. Using this conservative approach the projected 80-year EPFY for both Units 1 and 2 is 68 EPFY.

Fluence Projections

Reactor vessel integrity is assured by demonstrating that RV material fracture toughness will remain at levels that resist brittle fracture throughout the subsequent period of extended operation. The first step in the analysis of vessel embrittlement is calculation of the neutron fluence that causes increased embrittlement.

Fluence is projected for both beltline and extended beltline materials. The fluence methodology for beltline materials is approved by the NRC SER included in WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" (Reference 4.8-11). NUREG-2191, X.M2, indicates the use of Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," (Reference 4.8-12) adherent methods to estimate neutron fluence for RV regions significantly above and below the active fuel region of the core and RVI components may require additional justification. Figures 4.2.2-1 and 4.2.2-2 depict the axial boundary of the 1.0×10^{17} n/cm² fluence in the Z direction. The nozzle shell to intermediate shell circumferential weld is located close to the active fuel region and has historically been treated as beltline material. The lower extent of the nozzle shell forging, connected to the nozzle shell to intermediate shell circumferential weld, is also beltline material. Fluence projections for these two materials satisfy Regulatory Guide 1.190. The inlet and outlet nozzles are located above the active fuel region. Some of the inlet and outlet nozzles are projected to experience neutron fluence in excess of 1.0×10^{17} n/cm². These inlet and outlet nozzles are treated as extended beltline material for subsequent license renewal.

While the fluence projections for the inlet and outlet nozzles may have greater uncertainty than other beltline materials, these fluence projections are acceptable for performing RV integrity assessments for the subsequent license renewal period. The basis for this determination is consistent with LTR-SDA-18-049, "Evaluation of Conservatism and Margins Associated with Surry Units 1 and 2 Reactor Vessel Integrity Extended Beltline Evaluations for Subsequent License Renewal" (Reference 4.8-13) and LTR-REA-18-75, "Surry Extended Beltline Region Reactor Pressure Vessel Materials Fast Neutron Fluence Sensitivity Study on Material Mixture Above and Below the Active Core" (Reference 4.8-14), and is summarized as follows:

- The fluence at the inlet and outlet nozzles is significantly less than the fluence for the beltline materials, and the highest PTS and ART values are associated with the beltline materials,
- The projected fluence for the postulated flaw for the inlet and outlet nozzles assessed for the P-T Limit curves is based upon the lowest axial extent of the clad/ base metal interface on the inside radial surface of the RV without attenuation,
- Studies to date have shown that the DORT model calculates fluence in the Z direction above the core more conservatively than three-dimensional models such as RAPTOR-M3G,
- The controlling materials for PTS and P-T Limit curves continue to be the beltline materials,
- The fluence projections used in the SLR application conservatively utilized a constant material mixture of 90% water and 10% steel above and below the core. A sensitivity study was performed to show that this assumption was conservative compared to an analysis based upon more representative plant specific material mixture data above and below the core,

- The minimum fluence margin between the beltline materials and inlet and outlet nozzles is 178% (or 2.78 times) and represents the allowable increase in uncertainty that can be tolerated, and
- The SPS-specific minimum fluence margin of 178% (or 2.78 times) represents the increase in fluence uncertainty that is available relative to the customary use of +/-20% margin used in Regulatory Guide 1.190 for fluence projection.

Updated neutron fluence evaluations were performed and documented in WCAP-18028-NP. The fluence methodology used in WCAP-18028-NP is based on nuclear cross-section data derived from Evaluated Nuclear Data File/B Version VI (ENDF/B-VI). Furthermore, the neutron transport evaluation methodologies follow the guidance of Regulatory Guide 1.190. The methods used to develop the calculated pressure vessel fluence are consistent with the NRC-approved methodology described in WCAP-14040-A and are documented in the UFSAR, Section 4.1.7.3, "Calculation of Integrated Fast Neutron (E Greater than 1.0 MeV) Flux at the Irradiation Samples." The final safety evaluation report for WCAP-14040, Revision 3, dated February 27, 2004, states that the proposed fluence methodology adheres to the guidance of Regulatory Guide 1.190 and is therefore acceptable. Updated neutron fluence evaluations were used as an input to the RV integrity evaluations in support of initial license renewal.

Consistent with Sections 3.1 and 4.2 of NUREG-2192, materials exceeding a fast neutron fluence ($E > 1.0$ MeV) of 1.0×10^{17} n/cm² at the end of the subsequent period of extended operation are evaluated for changes in fracture toughness. Therefore, fast neutron fluence ($E > 1.0$ MeV) calculations were performed for the Units 1 and 2 RV circumferential welds (lower shell to lower vessel head, intermediate shell to lower shell, and nozzle shell to intermediate shell), inlet and outlet nozzle forging to vessel shell welds at the lowest extent, postulated 1/4T flaw location in the inlet and outlet nozzle, longitudinal welds (lower shell and intermediate shell), and plates (lower shell and intermediate shell), to determine if they will exceed a fast neutron fluence ($E > 1.0$ MeV) of 1.0×10^{17} n/cm² at the end of the subsequent period of extended operation. The materials that exceed the 1.0×10^{17} n/cm² fast neutron fluence ($E > 1.0$ MeV) threshold are evaluated to determine the effect of neutron irradiation embrittlement during the subsequent period of extended operation.

Table 4.2.1-1 and Table 4.2.1-2 summarize the results of the fluence projections to 68 EFPY for the Units 1 and 2 materials.

Table 4.2.1-1 indicates that some inlet and outlet nozzles have fast neutron fluence ($E > 1.0$ MeV) greater than 1.0×10^{17} n/cm² at the nozzle forging to vessel shell weld and one inlet nozzle has fast neutron fluence ($E > 1.0$ MeV) greater than 1.0×10^{17} n/cm² at the postulated 1/4T nozzle flaw location at 68 EFPY for Unit 1. Table 4.2.1-2, indicates that some inlet and outlet nozzles have fast neutron fluence ($E > 1.0$ MeV) greater than 1.0×10^{17} n/cm² at the nozzle forging to vessel shell

weld and one outlet and one inlet nozzle have fast neutron fluence ($E > 1.0$ MeV) greater than 1.0×10^{17} n/cm² at the 1/4T nozzle flaw location at 68 EFPY for Unit 2. Table 4.2.1-1 and Table 4.2.1-2 indicate that the lower shell to lower vessel head circumferential weld will remain below 1.0×10^{17} n/cm² through the subsequent period of extended operation for both Units 1 and 2.

TLAA Disposition: 10 CFR 54.21(c)(1)(ii)

The fluence analyses have been projected to the end of the subsequent period of extended operation. The results are to be used as inputs in the RV neutron embrittlement TLAA evaluations in Sections 4.2.2 through 4.2.6.

Table 4.2.1-1 Unit 1 - Maximum Fast Neutron Fluence ($E > 1.0$ MeV) Experienced by the Pressure Vessel Materials in the Traditional Beltline and Extended Beltline Regions at 68 EFPY at the Clad/Base Metal Interface

Material	Fast Neutron Fluence (n/cm^2)	
	68 EFPY	Region Applicability
Postulated 1/4T Flaw in Outlet Nozzle		
Nozzle 1	3.45E+16	N/A
Nozzle 2	2.49E+16	N/A
Nozzle 3	9.62E+16	N/A
Postulated 1/4T Flaw in Inlet Nozzle		
Nozzle 1	1.24E+17	Extended
Nozzle 2	3.22E+16	N/A
Nozzle 3	4.46E+16	N/A
Outlet Nozzle Forging to Vessel Shell Welds – Lowest Extent		
Nozzle 1	8.13E+16	N/A
Nozzle 2	5.86E+16	N/A
Nozzle 3	2.27E+17	Extended
Inlet Nozzle Forging to Vessel Shell Welds – Lowest Extent		
Nozzle 1	3.04E+17	Extended
Nozzle 2	7.84E+16	N/A
Nozzle 3	1.09E+17	Extended
Nozzle Shell	7.54E+18	Traditional
Nozzle Shell to Intermediate Shell Circumferential Weld	7.54E+18	Traditional
Intermediate Shell		
Plate 1	6.29E+19	Traditional
Plate 2	6.29E+19	Traditional
Intermediate Shell Longitudinal Welds		
Weld 1	1.25E+19	Traditional
Weld 2	1.25E+19	Traditional
Intermediate Shell to Lower Shell Circumferential Weld	6.31E+19	Traditional
Plate 2	6.35E+19	Traditional
Lower Shell		
Plate 1	6.35E+19	Traditional
Plate 2	6.35E+19	Traditional

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Material	Fast Neutron Fluence (n/cm ²)	
	68 EFPY	Region Applicability
Lower Shell Longitudinal Welds		
Weld 1	1.26E+19	Traditional
Weld 2	1.26E+19	Traditional
Lower Shell to Lower Vessel Head Circumferential Weld	<1E+17	Other ^(a)

(a) Dominion used N/A for situations where it is possible during the life of the plant to reach 1.0×10^{17} n/cm². EMAs were generated for these situations. The Lower Shell to Lower Vessel Head Circumferential Weld will never be in that category and are identified with a Region Applicability of "Other". EMAs were performed for all N/As.

Table 4.2.1-2 Unit 2 - Maximum Fast Neutron Fluence ($E > 1.0$ MeV) Experienced by the Pressure Vessel Materials in the Traditional Beltline and Extended Beltline Regions at 68 EFY at the Clad/Base Metal Interface

Material	Fast Neutron Fluence (n/cm^2)	
	68 EFY	Region Applicability
Postulated 1/4T Flaw in Outlet Nozzle ^(a)		
Nozzle 1	3.38E+16	N/A
Nozzle 2	2.48E+16	N/A
Nozzle 3	1.07E+17	Extended
Postulated 1/4T Flaw in Inlet Nozzle ^(a)		
Nozzle 1	1.39E+17	Extended
Nozzle 2	3.21E+16	N/A
Nozzle 3	4.37E+16	N/A
Outlet Nozzle Forging to Vessel Shell Welds – Lowest Extent		
Nozzle 1	7.96E+16	N/A
Nozzle 2	5.85E+16	N/A
Nozzle 3	2.53E+17	Extended
Inlet Nozzle Forging to Vessel Shell Welds – Lowest Extent		
Nozzle 1	3.40E+17	Extended
Nozzle 2	7.84E+16	N/A
Nozzle 3	1.07E+17	Extended
Nozzle Shell	8.65E+18	Traditional
Nozzle Shell to Intermediate Shell Circumferential Weld	8.65E+18	Traditional
Intermediate Shell		
Plate 1	7.20E+19	Traditional
Plate 2	7.20E+19	Traditional
Intermediate Shell Longitudinal Welds		
Weld 1	1.29E+19	Traditional
Weld 2	1.29E+19	Traditional
Intermediate Shell to Lower Shell Circumferential Weld	7.22E+19	Traditional
Lower Shell		
Plate 1	7.26E+19	Traditional
Plate 2	7.26E+19	Traditional

Material	Fast Neutron Fluence (n/cm ²)	
	68 EFPY	Region Applicability
Lower Shell Longitudinal Welds		
Weld 1	1.30E+19	Traditional
Weld 2	1.30E+19	Traditional
Lower Shell to Lower Vessel Head Circumferential Weld	<1E+17	Other ^(b)

(a) Nozzle 1/4T flaw maximum fluence values are taken at the surface of the nozzle.

(b) Dominion used N/A for situations where it is possible during the life of the plant to reach 1.0×10^{17} n/cm². EMAs were generated for these situations. The Lower Shell to Lower Vessel Head Circumferential Weld will never be in that category and are identified with a Region Applicability of "Other". EMAs were performed for all N/As.

4.2.2 UPPER-SHELF ENERGY

TLAA Description:

Upper-shelf energy (USE) is the parameter used to indicate the toughness of a material at elevated temperature. There are two sets of rules that govern USE acceptance criteria. 10 CFR 50, Appendix G, Paragraph IV.A.1.a, states that RV beltline materials must have Charpy USE of no less than 75 ft-lb initially, and must maintain Charpy USE throughout the life of the vessel of no less than 50 ft-lb, unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of Charpy USE will provide margins of safety against fracture equivalent to those required by ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," (Reference 4.8-15) Appendix G, "Fracture Toughness Criteria for Protection Against Failure." For materials outside the beltline, a minimum value of 30 ft-lbs at 10°F was specified by ASTM E208, and required by ASME Code, Section III, at the time of the design and fabrication of the RVs for Units 1 and 2.

The current licensing basis Charpy USE calculations were prepared for the Units 1 and 2 RV beltline materials for 48 EFPY. Since the USE value is a function of 48 EFPY fluence, associated with the 60-year licensed operating period, these USE calculations meet the criteria of 10 CFR 54.3(a) and have been identified as TLAA's requiring evaluation for 80 years.

TLAA Evaluation:

Per Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," (Reference 4.8-16) the Charpy USE should be assumed to decrease as a function of fluence according to Figure 2 of the Regulatory Guide, which provides percent decrease in USE as a function of 1/4T fluence and the copper content for plates and welds, when credible surveillance data is not available. If credible surveillance data is available, the decrease in USE may be obtained by plotting the reduced plant surveillance data on Figure 2 of Regulatory Guide 1.99 and fitting the data with a line drawn parallel to the existing lines as the upper bound of all of the data. The 1/4T fluence at 68 EFPY is used to determine the reduction of the initial USE.

As documented in WCAP-18242-NP, the materials projected to exceed 1.0×10^{17} n/cm² (E > 1.0 MeV) at 68 EFY are evaluated to determine their impact on USE during the proposed subsequent period of extended operation. The forgings and welds corresponding to some inlet and outlet nozzles are predicted to experience neutron fluence greater than 1.0×10^{17} n/cm² at the end of the subsequent period of extended operation. However, for conservatism all of the inlet and outlet nozzle materials are considered part of the extended beltline in the USE evaluation. The Units 1 and 2 materials include three (3) inlet nozzles, three (3) outlet nozzles, three (3) inlet nozzle to upper-shell welds, and three (3) outlet nozzle to upper-shell welds per unit. (Note: nozzle-shell and upper-shell refer to the same component and are used interchangeably).

The identification of the RV plate and weld materials is shown in Figure 4.2.2-1 for Unit 1 and Figure 4.2.2-2 for Unit 2. The material property inputs used for the RV integrity evaluations are described in this section. The initial material properties were updated from previous RV integrity evaluations per PWROG-16045-NP and WCAP-18242-NP, Appendix E, and the fluence values were updated per WCAP-18028-NP and WCAP-18242-NP, Section 2. Additionally, initial USE values are supplied in Table 4.2.2-1 and Table 4.2.2-3.

The requirements on USE for beltline materials are included in 10 CFR 50, Appendix G, which requires utilities to submit an analysis at least three years prior to the time that the USE of any RV material is predicted to drop below 50 ft-lb. Dominion has conservatively elected to perform equivalent margins analyses (EMAs) for inlet and outlet nozzle welds with Charpy USE near 50 ft-lb at the end of the subsequent period of extended operation.

Two methods can be used to predict the decrease in USE with irradiation, depending on the availability of credible surveillance capsule data as defined in Regulatory Guide 1.99. For vessel beltline materials that are not in the surveillance program or have non-credible data, the Charpy USE (Position 1.2) is assumed to decrease as a function of fluence and copper content, as indicated in Regulatory Guide 1.99. When two or more credible surveillance sets become available from the reactor, they may be used to determine the Charpy USE of the surveillance material. The surveillance data are then used in conjunction with Regulatory Guide 1.99 to predict the change in USE (Position 2.2) of the RV material due to irradiation. Per Regulatory Guide 1.99 (Revision 2), when credible data exists the Position 2.2 projected USE value should be used in preference to the Position 2.1 projected USE value. Such cases exist in Table 4.2.2-5 wherein SLR USE values in the Position 1.2 section that fall below 50 ft-lbs are not an issue because corresponding values in the Position 2.2 section are above 50 ft-lbs when considering credible surveillance data.

The 68 EFY Position 1.2 USE values of the vessel materials can be predicted using the corresponding fluence projections (1/4T for beltline materials and surface for inlet/outlet nozzles), the copper content of the materials, and Figure 2 in Regulatory Guide 1.99.

The predicted Position 2.2 USE values are determined for the RV materials that are contained in the surveillance program by using the reduced plant surveillance data along with the corresponding fluence projection (1/4T for beltline materials and surface for inlet/outlet nozzles). The reduced plant surveillance data was obtained from Table 7-6 of BAW-2324, "Analysis of Capsule X Virginia Power Surry Unit No. 1, Reactor Vessel Material Surveillance Program" (Reference 4.8-17) for Unit 1. The reduced plant surveillance data was obtained from Table 5-12 of WCAP-16001, "Analysis of Capsule Y from Dominion Surry Unit 2 Reactor Vessel Radiation Surveillance Program" (Reference 4.8-18) for Unit 2. The surveillance data was plotted in Regulatory Guide 1.99, Figure 2 using the surveillance capsule fluence values documented in Table 2-1 of WCAP-18242-NP, for Unit 1 and Table 2-2 of WCAP-18242-NP, for Unit 2.

The projected USE values were calculated to determine if the values for Units 1 and 2 materials remain above the 50 ft-lb criterion at 68 EFPY. The projected USE values for the inlet and outlet nozzle forgings were conservatively calculated using the maximum fluence values corresponding to the lowest extent of the nozzle to shell welds. These calculations are summarized in Table 4.2.2-5 and Table 4.2.2-6.

Conclusion

For Unit 1, the limiting USE value at 68 EFPY is 32 ft-lb (see Table 4.2.2-5); this value applies to the Intermediate to Lower Shell Circumferential Weld using Position 1.2. For Unit 2, the limiting USE value at 68 EFPY is 41 ft-lb (see Table 4.2.2-6); this value applies to the Upper to Intermediate Shell Circumferential Weld using Position 1.2.

The NRC has previously approved the use of the equivalent margins analysis (EMA) BAW-2494, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessel of Surry Power Station Units 1 and 2 for Extended Life through 48 Effective Full Power Years" (Reference 4.8-19) to qualify all of the materials currently projected to drop below 50 ft-lb USE at 68 EFPY. These materials are identified by the notes in Table 4.2.2-1, Table 4.2.2-3, Table 4.2.2-5, Table 4.2.2-6 herein and are summarized below. The EMAs for these materials are updated for the subsequent period of extended operation under ANP-3679NP, "Low Upper-Shelf Toughness Fracture Mechanics Analysis for Surry Units 1 and 2 Reactor Vessels for Levels A & B Service Loads at 80 Years," (Reference 4.8-20) and ANP-3680NP, "Low Upper-Shelf Toughness Fracture Mechanics Analysis for Surry Units 1 and 2 Reactor Vessels for Levels C & D Service Loads at 80 Years" (Reference 4.8-21). The updated EMA is based upon the provisions outlined in ASME Code, Section XI, Appendix K. The selection of design transients for Levels C & D service loads are based on the guidance in Regulatory Guide 1.161, "Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 Ft-Lb." (Reference 4.8-22) and ASME Code, Section XI, Appendix K.

An EMA should be submitted three years before a material is projected to drop below 50 ft-lbs; however, no additional materials are projected to drop below 50 ft-lb USE during the subsequent period of extended operation.

The following Unit 1 and Unit 2 materials are addressed by EMAs for the subsequent period of extended operation:

Unit 1:

- Upper to Intermediate Shell Circumferential Weld, Heat # 25017 (J726)
- Intermediate Shell Longitudinal Welds L3 and L4, Heat # 8T1554
- Intermediate to Lower Shell Circumferential Weld, Heat # 72445
- Lower Shell Longitudinal Weld L1, Heat # 8T1554
- Lower Shell Longitudinal Weld L2, Heat # 299L44
- Inlet Nozzle to Shell Welds, Heat # 299L44 and # 8T1762; (Projected USE > 50 ft-lbs at 68 EFPY)
- Outlet Nozzle to Shell Welds, Heat # 8T1762 and # 8T1554B; (Projected USE > 50 ft-lbs at 68 EFPY)

Unit 2:

- Upper to Intermediate Shell Circumferential Weld, Heat # 4275 (J737)
- Intermediate Shell Longitudinal Welds L3 and L4, Heat # 72445
- Intermediate Shell Longitudinal Weld L4, Heat # 8T1762
- Intermediate to Lower Shell Circumferential Weld, Heat # 0227
- Lower Shell Longitudinal Weld L1 and L2, Heat # 8T1762
- Inlet Nozzle to Shell Welds, Heat # 8T1762; (Projected USE not projected > 50 ft-lbs at 68 EFPY)
- Outlet Nozzle to Shell Welds, Rotterdam Weld; (Projected USE > 50 ft-lbs at 68 EFPY)

An EMA has been completed for the Unit 1 and Unit 2 Inlet and Outlet Nozzle to Shell Welds even though these materials are not projected to drop below 50 ft-lbs through 68 EFPY using the methods herein. The inlet and outlet nozzle welds are the only materials included in ANP-3679NP and ANP-3680NP that were not previously addressed by EMA. The EMA is applicable to the

Units 1 and 2 nozzle to shell welds which exceed the fluence criterion of 1.0×10^{17} n/cm² before 68 EFPY. These materials include those listed below.

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

For Unit 1, the limiting USE value for materials not requiring an EMA at 68 EFPY is 54 ft-lb (see Table 4.2.2-5); this value corresponds to the Inlet Nozzle to Upper Shell Welds (Heat # 299L44) using Position 2.2. For Unit 2, the limiting USE value for materials not requiring an EMA at 68 EFPY is also 54 ft-lb (see Table 4.2.2-6); this value corresponds to the Outlet Nozzle to Upper Shell Welds (Rotterdam) using Position 1.2. Except for the materials listed above, all of the beltline and extended beltline materials in the Units 1 and 2 RVs are projected to remain above the USE screening criterion value of 50 ft-lb (per 10 CFR 50, Appendix G) through the subsequent period of extended operation (68 EFPY).

Equivalent Margins Analysis

The ASME Code, Section XI, acceptance criteria for Levels A through D Service Loadings for all Units 1 and 2 RV beltline and extended beltline Linde 80 welds are satisfied and are reported in Framatome Reports BAW-2192, Supplement 1 (Revision 0), "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads Topical Report," (Reference 4.8-23) and BAW-2178, Supplement 1 (Revision 0), "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels C & D Service Loads Topical Report," (Reference 4.8-24) submitted to the NRC in December 2017. The Surry Power Plant specific versions of the EMA are documented in following reports:

- ANP-3679P, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis for Surry Units 1 and 2 Reactor Vessels for Levels A & B Service Loads at 80-Years" (Reference 4.8-25)
- ANP-3679NP, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis for Surry Units 1 and 2 Reactor Vessels for Levels A & B Service Loads at 80-Years"
- ANP-3680P, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis for Surry Units 1 and 2 Reactor Vessels for Levels C & D Service Loads at 80-Years" (Reference 4.8-26)
- ANP-3680NP, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis for Surry Units 1 and 2 Reactor Vessels for Levels C & D Service Loads at 80-Years"

The plant-specific EMA reports contain the same information as in BAW-2192, Supplement 1 and BAW-2178, Supplement 1 except that the information for Oconee 1, 2, and 3 and Turkey Point 3 and 4 has been removed.

The 80-year clad/base metal fluence values reported in Table 3 -1 of BAW-2178, Supplement 1, and Table 3-1 of BAW-2192, Supplement 1 have been confirmed to bound the 68 EFPY fluence values reported in Table 4.2.1-1 and Table 4.2.1-2. The EMAs conservatively utilized 80-year fluence values shown in (e) of at least an order of magnitude higher than the 68 EFPY nozzle fluence reported in Table 4.2.1-1 and Table 4.2.1-2. In addition, the weld chemistry data reported in Table 3-1 of BAW-2178, Supplement 1, and Table 3-1 of BAW-2192, Supplement 1 is consistent with weld chemistry reported in Tables 4.2.2-1 through Table 4.2.2-4. The level C and D limiting design transients reported in Section 4.3.2 of BAW-2178P, Supplement 1, are applicable to Units 1 and 2 and are based on a review of the ASME Code, Section III, Reactor Vessel Design Specification transients and the UFSAR Chapter 14 events relative to transients that would result in the highest thermal stresses coupled with pressure stresses relative to the EMA analysis; this satisfies Regulatory Guide 1.161 with respect to Level C and D transient selection. The materials of construction, RV geometry, and range of explanatory variables for the J-R model (Section A.5 of BAW-2192, Supplement 1) reported in the topical reports are confirmed to be applicable to Linde 80 and Rotterdam beltline and extended beltline welds at Units 1 and 2.

As such, Units 1 and 2 are bounded by topical report submittals BAW-2178, Supplement 1, and BAW-2192, Supplement 1 relative to fluence, weld chemistry, geometry, materials of construction, design transients and the J-R model applicability. The results of the EMA for Units 1 and 2, as reported in BAW-2178 P/NP and BAW-2192 P/NP, are summarized below.

Levels A & B Service Loadings

Reactor Vessel Shell Welds (Beltline)

The limiting RV shell weld is Unit 1 axial weld SA-1526.

- With factors of safety of 1.15 on pressure and 1.0 on thermal loading, the applied J-integral (J_1) is less than the J-integral of the material at a ductile flaw extension of 0.10 in. ($J_{0.1}$). The ratio $J_{0.1}/J_1$ is greater than the required value of 1.0.
- With a factor of safety of 1.25 on pressure and 1.0 on thermal loading, flaw extensions are ductile and stable since the slope of the applied J-integral curve is less than the slope of the lower bound J-R curve at the point where the two curves intersect.

Reactor Vessel Transition Welds and RV Nozzle Welds (Extended Beltline)

- The limiting weld for Units 1 and 2 considering RV transition welds (upper and lower) and the RV inlet and outlet nozzle-to-shell welds is the longitudinal weld SA-1585 near the base of the transition section.
- With factors of safety of 1.15 on pressure and 1.0 on thermal loading, the applied J-integral (J1) is less than the J-integral of the material at a ductile flaw extension of 0.10 in. (J0.1). The ratio $J0.1/J1$ is greater than the required value of 1.0.
- With a factor of safety of 1.25 on pressure and 1.0 on thermal loading, flaw extensions are ductile and stable since the slope of the applied J-integral curve is less than the slope of the lower bound J-R curve at the point where the two curves intersect.

Levels C & D Service Loadings

Reactor Vessel Shell Welds (Beltline)

The limiting weld among the RV shell welds is Unit 1 longitudinal weld SA-1526. The limiting transient for Level C & D service Loads is the SSDC 1.3 steam line break.

- With a factor of safety of 1.0 on loading, the applied J-integral (J1) for the limiting RV shell weld (Unit 1, SA-1526) is less than the lower bound J-integral of the material at a ductile flaw extension of 0.10 inch (J0.1) with a ratio $J0.1/J1$ that is greater than the required value of 1.0.
- With a factor of safety of 1.0 on loading, flaw extensions are ductile and stable for the limiting RV shell weld (SA-1526) since the slope of the applied J-integral curve is less than the slopes of both the lower bound and mean J-R curves at the points of intersection.
- For weld SA-1526 it was demonstrated that flaw growth is stable at much less than 75% of the vessel wall thickness. It has also been shown that the remaining ligament is sufficient to preclude tensile instability.

Reactor Vessel Transition Welds and RV Nozzle Welds (Extended Beltline)

The upper transition weld and RV inlet and outlet nozzle-to-shell welds were evaluated for Levels C and D Service Loadings. The limiting transient for Level C & D service loads is the SSDC 1.3 steam line break.

- With a factor of safety of 1.0 on loading, the applied J-integral (J1) for the RV nozzle-to-shell welds and upper transition weld are less than the lower bound J-integral of the material at a ductile flaw extension of 0.10 inch (J0.1). All ratios are greater than 1.0.
- With a factor of safety of 1.0 on loading, flaw extensions are ductile and stable for the limiting RV outlet nozzle-to-shell weld (i.e., limiting location considering RV nozzle-to-shell welds and upper transition weld).
- For the RV outlet nozzle-to-shell weld it was demonstrated that flaw growth is stable at much less than 75% of the vessel wall thickness. Tensile instability was not explicitly calculated but because this section of the RV is thicker compared to the RV shell welds, it is considered to be bounded by the RV shell location.

B&WOG J-R Model

The original B&WOG J-R Model 4B reported in BAW-2192PA, Supplement 1, Appendix A, was used to obtain J material (i.e., J(0.1)) for the 80-year equivalent margins analyses reported in BAW-2192, Supplement 1, and BAW-2178, Supplement 1. Model 4B is based on fracture toughness data (1352 J delta-a data points) irradiated to a fluence ranging from 0.0 to 8.45×10^{18} n/cm², which is less than the peak 1/10T 80-year fluence projected for Units 1 and 2. To further substantiate the use of the B&WOG J-R model, the original J delta-a data used to generate the B&WOG J-R model 4B was used to independently benchmark the original B&WOG model using the R-project statistical tool. The benchmark is designated B&WOG J-R Model 5B. New J-R data (419 new J delta-a data points with fluence to 5.8×10^{19} n/cm²) were then added to the original population of welds (total population of 1774 data points) and the fitting coefficients (assuming the same model form) were generated. The B&WOG model that includes the total population of J delta-a data (1774) is designated Model 6B. Model 6B is based on test data out to a fluence of 5.8×10^{19} n/cm², which is greater than the peak 1/4T fluence of 8.16×10^{18} n/cm² and 1/10T fluence of 1.083×10^{19} n/cm² for Units 1 and 2 limiting weld SA-1526.

Use of Model 6B for fluence values in excess of 5.8×10^{19} n/cm² is considered to be a model extrapolation and the uncertainty may increase (i.e., -2SE). Fluence estimates at T/4 and T/10 are well below 5.8×10^{19} n/cm² and the J-R Model is used well within the interpolation range (i.e., for weld SA-1526 fluence equals 8.16×10^{18} n/cm² at 1/4T and 1.083×10^{19} n/cm² at T/10). For Units 1 and 2, use of Model 6B (model extrapolation) increased the J(0.1)/J1 by approximately 6% for Level A and B, and 5% for Level C and D when compared to Model 4B, and, all margins remain above the acceptance criterion of 1.0. In addition, the combination of Level C and D acceptance criteria applied to Level D transients provides additional conservatism in the equivalent margins analyses. The B&WOG J-R models (including Models 4B and 6B) are discussed in BAW-2192, Supplement 1, Appendix A.

TLAA Disposition: 10 CFR 54.21(c)(1)(ii)

The USE analyses have been projected to the end of the subsequent period of extended operation.

Figure 4.2.2-1 Unit 1 - Axial Boundary of the $1.0E+17$ n/cm² Fluence Threshold in the +Z Direction (at 54 and 72 EFPY)

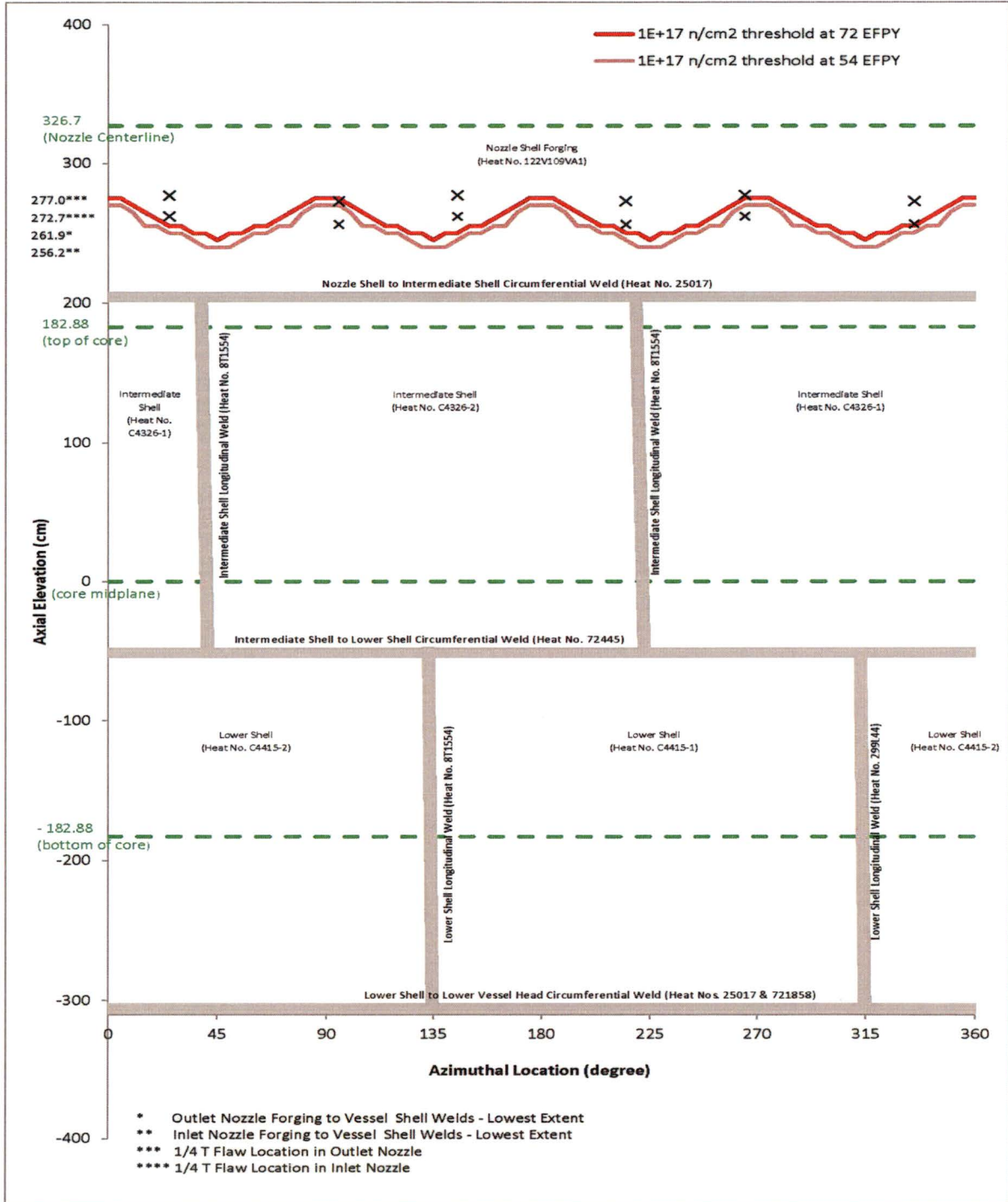


Figure 4.2.2-2 Unit 2 - Axial Boundary of the $1.0E+17$ n/cm² Fluence Threshold in the +Z Direction (at 54 and 72 EFPY)

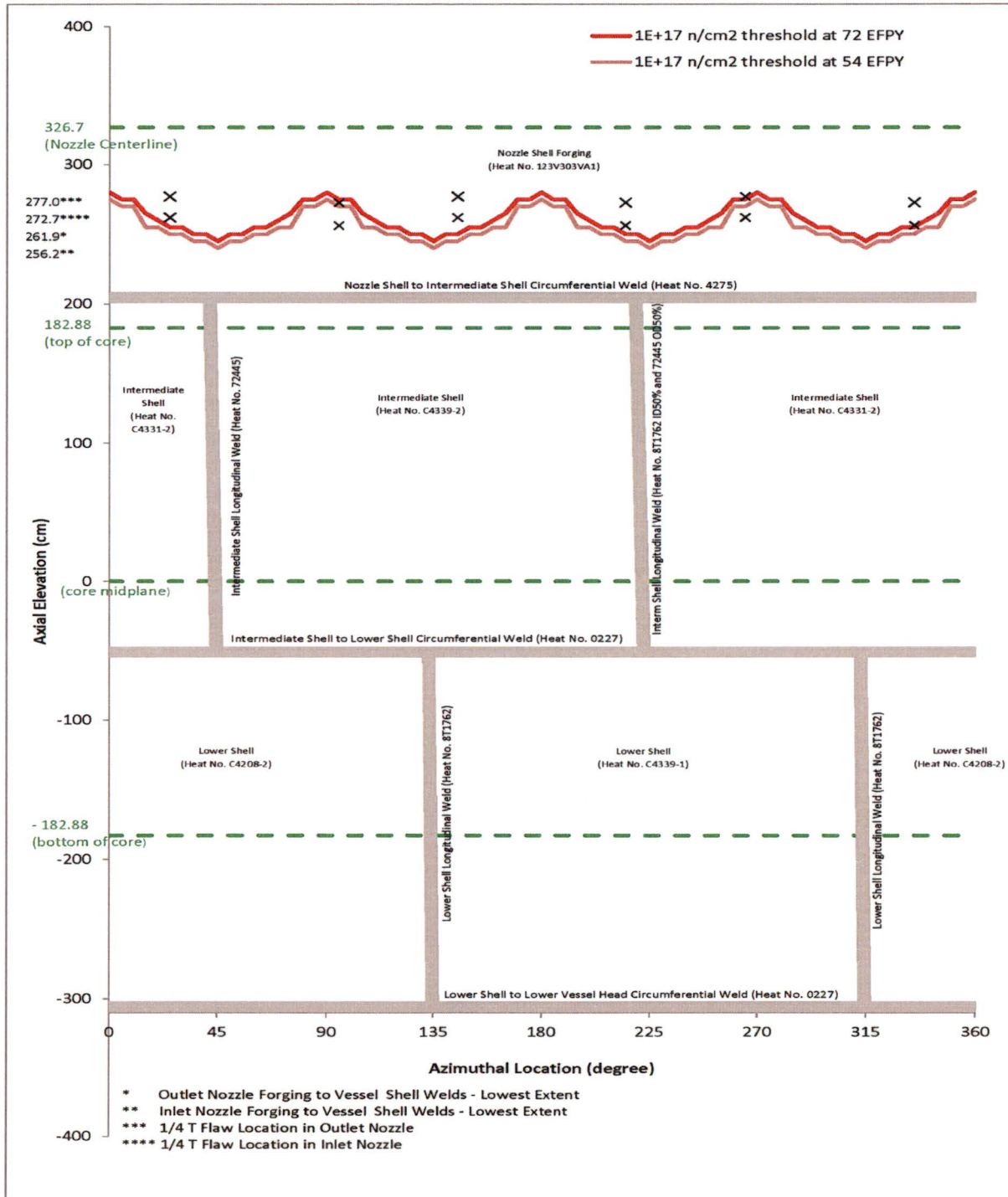
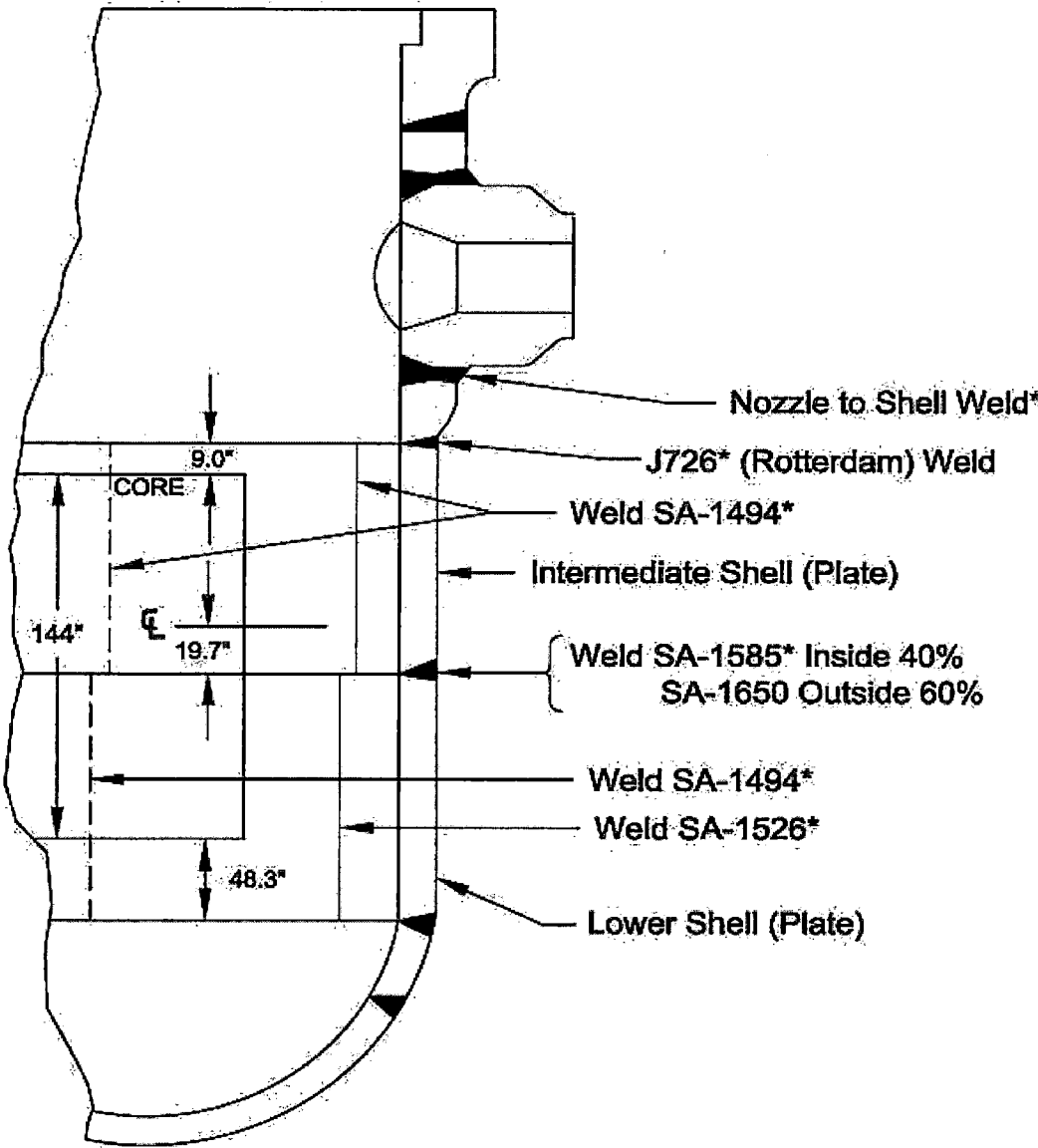
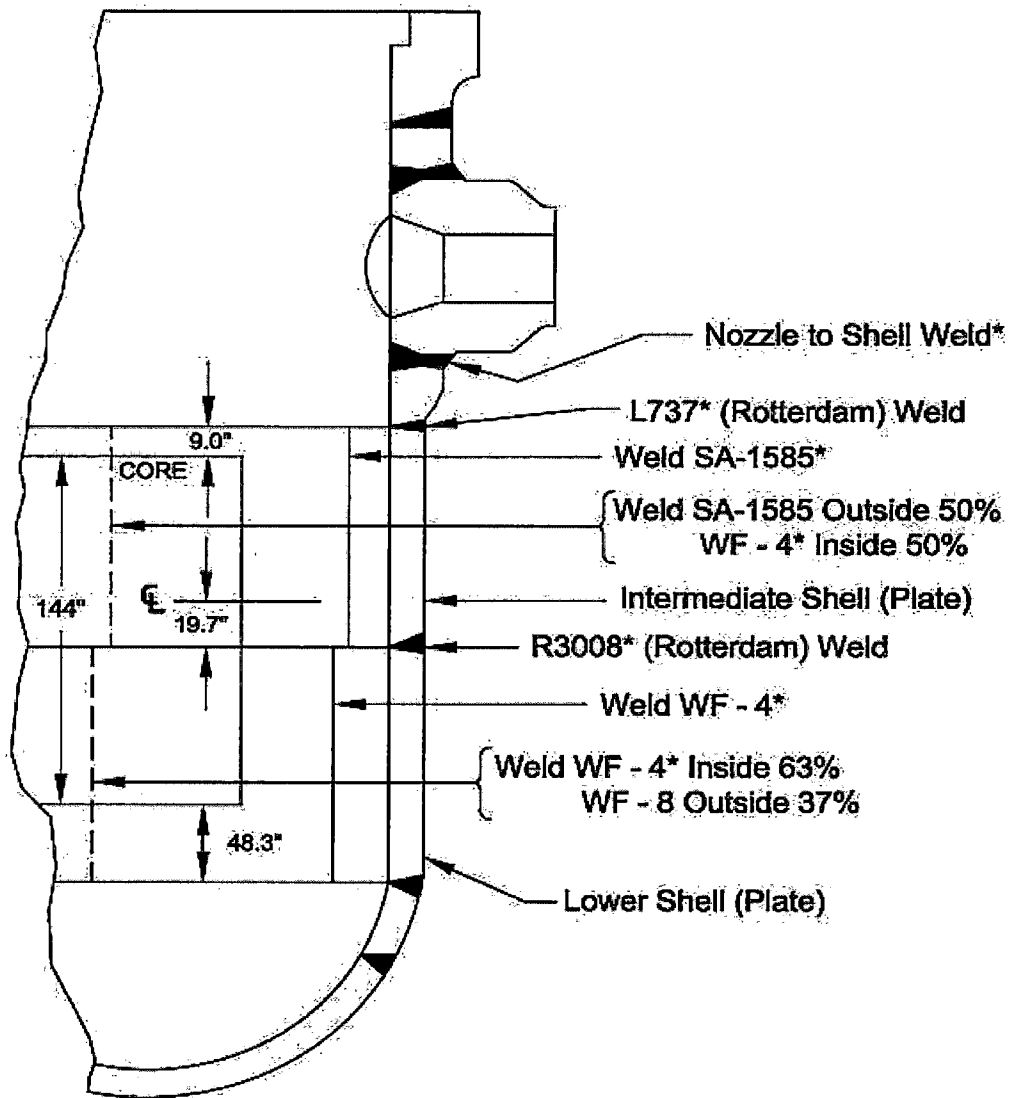


Figure 4.2.2-3 Reactor Vessel - Unit 1



* Equivalent Margins Analysis performed for these Linde 80 and Rotterdam Welds.

Figure 4.2.2-4 Reactor Vessel - Unit 2



* Equivalent Margins Analysis performed for these Linde 80 and Rotterdam Welds.

Table 4.2.2-1 Best Estimate Cu and Ni Weight Percent Values, Initial RT_{NDT} Values, and Initial USE Values for the Unit 1 RV Beltline and Surveillance Materials

RV Material	Wt.% Cu	Wt.% Ni	RT _{NDT(U)} (°F)	σ ₁ (°F)	Initial USE (ft-lb)
RV Beltline Materials ^(a)					
Upper Shell Forging 122V109VA1	0.11	0.74	40	0	114
Intermediate Shell Plate C4326-1	0.11	0.55	10	0	115
Intermediate Shell Plate C4326-2	0.11	0.55	11.4	0	94
Lower Shell Plate C4415-1	0.102	0.493	20	0	103
Lower Shell Plate C4415-2	0.11	0.5	4.6	0	82
Upper to Intermediate Shell Circumferential Weld (Heat # 25017)	0.33	0.1	0	20	≥ 64 ^(b)
Intermediate Shell Longitudinal Welds L3 and L4 (Heat # 8T1554)	0.16	0.57	-48.6	18	64 ^(b)
Intermediate to Lower Shell Circumferential Weld (Heat # 72445)	0.22	0.54	-72.5	12	64 ^(b)
Lower Shell Longitudinal Weld L1 (Heat # 8T1554)	0.16	0.57	-48.6	18	64 ^(b)
Lower Shell Longitudinal Weld L2 (Heat # 299L44)	0.34	0.68	-74.3	12.8	64 ^(b)
RV Surveillance Materials ^(c)					
Lower Shell Plate C4415-1	0.102	0.493	20	0	103
Surveillance Weld (Heat # 299L44)	0.23	0.64	---	---	70

Notes:

- (a) All values were taken from Table 8 of PWROG-16045 NP, unless otherwise noted.
- (b) Per UFSAR (Tables 4.1-14 and 4.1-15), RV Equivalent Margins Analysis (EMA) report BAW-2494, was approved for these welds for 48 EFPY. The EMAs are updated for the subsequent period of extended operation. Linde 80 initial USE values are set to the generic value of 64 ft-lbs per BAW-2313, Supplement 1. Only limited Charpy test information is available for Heat # 25017. Based on the average Charpy energy value of the weld qualification tests completed at 10°F, the USE for Heat # 25017 is at least 64 ft-lbs. This value of 64 ft-lbs is conservative compared to the generic Rotterdam-weld results documented in PWROG-17090-NP, "Generic Rotterdam Forging and Weld Initial Upper-Shelf Energy Determination." (Reference 4.8-27)
- (c) The surveillance plate data was taken to be the same as the vessel plate data. The surveillance weld data was obtained from BAW-2324.

Table 4.2.2-2 Best Estimate Cu and Ni Weight Percent Values, Initial RT_{NDT} Values, and Initial USE Values for the Unit 1 RV Materials

RV Material	Wt.% Cu	Wt.% Ni	RT _{NDT(U)} (°F)	σ _I (°F)	Initial USE (ft-lb)	
RV Extended Beltline Materials ^(a)						
Inlet Nozzle 1 (Heat # 9-4787)	0.159	0.85	10.3	0	63	
Inlet Nozzle 2 (Heat # 9-5078)	0.159	0.87	11.6	0	64	
Inlet Nozzle 3 (Heat # 9-4819)	0.159	0.84	-47.2	0	68	
Outlet Nozzle 1 (Heat # 9-4825-1)	0.159	0.85	-44.9	0	68	
Outlet Nozzle 2 (Heat # 9-4762)	0.159	0.83	-87.5	0	82	
Outlet Nozzle 3 (Heat # 9-4788)	0.159	0.84	-50.2	0	71	
Inlet Nozzle to Upper Shell Welds	Heat # 299L44	0.34	0.68	-7	20.6	64
	Heat # 8T1762	0.19	0.57	-4.9	19.7	64
Outlet Nozzle to Upper Shell Welds	Heat # 8T1762	0.19	0.57	-4.9	19.7	64
	Heat # 8T1554 B	0.16	0.57	-4.9	19.7	64

Note:

(a) All values were taken from Table 8 of PWROG-16045-NP.

Table 4.2.2-3 Best Estimate Cu and Ni Weight Percent Values, Initial RT_{NDT} Values, and Initial USE Values for the Unit 2 RV Beltline and Surveillance Materials

RV Material	Wt.% Cu	Wt.% Ni	RT _{NDT(U)} (°F)	σ_1 (°F)	Initial USE (ft-lb)
RV Beltline Materials ^(a)					
Upper Shell Forging 123V303VA1	0.11	0.72	30	0	104
Intermediate Shell Plate C4331-2	0.12	0.6	15	0	84
Intermediate Shell Plate C4339-2	0.11	0.54	7.8	0	83
Lower Shell Plate C4208-2	0.15	0.55	-30	0	94
Lower Shell Plate C4339-1	0.107	0.53	-4.4	0	101
Upper to Intermediate Shell Circumferential Weld (Heat # 4275)	0.35	0.1	0	20	≥ 68 ^(b)
Intermediate Shell Longitudinal Welds L3 and L4 (OD 50%) (Heat # 72445)	0.22	0.54	-72.5	12	64 ^(b)
Intermediate Shell Longitudinal Weld L4 (ID 50%) (Heat # 8T1762)	0.19	0.57	-48.6	18	64 ^(b)
Intermediate to Lower Shell Circumferential Weld (Heat # 0227)	0.187	0.545	0 ^(c)	0 ^(c)	82 ^(c)
Lower Shell Longitudinal Welds L1 and L2 (Heat # 8T1762)	0.19	0.57	-48.6	18	64 ^(b)
RV Surveillance Materials ^(d)					
Lower Shell Plate C4339-1	0.107	0.53	-4.4	0	101
Surveillance Weld (Heat # 0227)	0.19	0.56	---	---	91

Notes:

- (a) All values were taken from Table 9 of PWROG-16045-NP, unless otherwise noted.
- (b) Per UFSAR (Tables 4.1-14 and 4.1-15), RV EMA report BAW-2494 was approved for these welds for 48 EFPY. The EMAs are updated for the subsequent period of extended operation. Linde 80 initial USE values are set to the generic value of 64 ft-lbs per BAW-2313, Supplement 1, "Supplement to B&W Fabricated Reactor Vessel Materials and Surveillance Data Information for Surry Unit 1 and Unit 2" (Reference 4.8-28). Only limited Charpy test information is available for Heat # 4275. Based on the average Charpy energy value of the weld qualification tests completed at 10°F, the USE for Heat # 4275 is at least 68 ft-lbs. This value of 64 ft-lbs is conservative compared to the generic Rotterdam weld results documented in PWROG-17090-NP.
- (c) Initial properties are established in Appendix B of WCAP-18242-NP. Since the initial RT_{NDT} is based on measured data, σ_u is equal to 0°F. Per UFSAR (Tables 4.1-14 and 4.1-15), RV EMA report BAW-2494 was approved for this weld for 48 EFPY. The EMA is updated for the subsequent period of extended operation.
- (d) The surveillance plate data was taken to be the same as the vessel plate data. The surveillance weld data was obtained from WCAP-16001.

Table 4.2.2-4 Best Estimate Cu and Ni Weight Percent Values, Initial RT_{NDT} Values, and Initial USE Values for the Unit 2 RV Extended Beltline Materials

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

Notes:

- (a) All values were taken from Table 9 of PWROG-16045-NP. Associated σ₁ values are also available from PWROG-16045-NP.
- (b) Per PWROG-16045-NP, this initial USE value is set equal to the USE value of the first tested capsule from WCAP-16001 (Reference 4.8-18). This methodology utilizes BTP 5-3 (Reference 4.8-29), Position 1.2 guidance, as no USE data is available from the supplier. The value used herein is conservative in comparison. In addition, Dominion conservatively elected to complete an EMA on this material.

Table 4.2.2-5 Predicted USE Values at 68 EPFY for Unit 1

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

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RV Material	Wt.% Cu ^(a)	SLR 1/4T Fluence ^(b) (x 10 ¹⁹ n/cm ²)	Initial USE ^(a) (ft-lb)	Projected USE Decrease ^(c) (%)	SLR USE (ft-lb)
Outlet Nozzle 3 to Upper Shell Weld (Heat # 8T1554B)	0.16	0.0140	64	12	56
Inlet Nozzle 1 (Heat # 9-4787)	0.159	0.0304	63	11	56
Inlet Nozzle 2 (Heat # 9-5078)	0.159	0.0078	64	10	58
Inlet Nozzle 3 (Heat # 9-4819)	0.159	0.0109	68	10	61
Outlet Nozzle 1 (Heat # 9-4825-1)	0.159	0.0081	68	10	61
Outlet Nozzle 2 (Heat # 9-4762)	0.159	0.0059	82	10	74
Outlet Nozzle 3 (Heat # 9-4788)	0.159	0.0227	71	10.5	64
Position 2.2 ^(d)					
Lower Shell Plate C4415-1	0.102	3.9200	103	28	74
Lower Shell Plate C4415-2	0.11	3.9200	82	28	59
Lower Shell Longitudinal Weld L2 ^(e) (Heat # 299L44)	0.34	0.7770	64	35	42 ^(e)
Inlet Nozzle 1 to Upper Shell Weld (Heat # 299L44)	0.34	0.0188	64	15	54
Inlet Nozzle 2 to Upper Shell Weld (Heat # 299L44)	0.34	0.0048	64	15	54
Inlet Nozzle 3 to Upper Shell Weld (Heat # 299L44)	0.34	0.0067	64	15	54

Notes:

- (a) Material data is from Table 4.2.2-1 and Table 4.2.2-2.
- (b) The 1/4T fluence was calculated using the fluence data in Table 4.2.1-1, the Regulatory Guide 1.99 correlation, and the Units 1 and 2 RV wall thickness of 8.05 inches. The surface fluence at the lowest extent of the nozzle weld was used to represent the inlet and outlet nozzle forgings; this approach is conservative. Bounding material fluence values, only, are shown in Figure 5-1 of WCAP-18242-NP for the nozzle materials.
- (c) The Position 1.2 USE decrease values were calculated by plotting the 1/4T fluence values on Figure 2 of Regulatory Guide 1.99 and using the material specific Cu wt. percent values.
- (d) Surveillance data (deemed credible per Appendix A of WCAP-18242-NP) from Table 7-6 of BAW-2324 were used in the calculation of Unit 1 Position 2.2 USE projections. Regulatory Guide 1.99, Position 2.2 indicates that an upper bound line drawn parallel to the existing lines (in Figure 2 of the Regulatory Guide) through the surveillance data points should be used in preference to the existing graph lines for determining the decrease in USE.
- (e) These weld materials were previously addressed by EMA Report BAW-2494 for 48 EFPY. EMAs for these materials have been generated.
- (f) Per Regulatory Guide 1.99 (Revision 2), when credible data exists the Position 2.2 projected USE value should be used in preference to the Position 2.1 projected USE value.

Table 4.2.2-6 Predicted USE Values at 68 EPFY for Unit 2

RV Material	Wt.% Cu ^(a)	SLR 1/4T Fluence ^(b) (x10 ¹⁹ n/cm ²)	Initial USE ^(a) (ft-lb)	Projected USE Decrease ^(c) (%)	SLR USE (ft-lb)
Position 1.2					
Upper Shell Forging 123V303VA1	0.11	0.5340	104	18	85
Upper to Intermediate Shell Circumferential Weld ^(e) Heat # 4275	0.35	0.5340	68	39	41 ^(e)
Intermediate Shell Plate C4331-2	0.12	4.4400	84	30	59
Intermediate Shell Plate C4339-2	0.11	4.4400	83	29	59
Intermediate Shell Longitudinal Welds L3 and L4 (OD 50%) ^(e) (Heat # 72445)	0.22	0.7960	64	34	42 ^(e)
Intermediate Shell Longitudinal Weld L4 (ID 50%) ^(e) (Heat # 8T1762)	0.19	0.7960	64	32	44 ^(e)
Intermediate to Lower Shell Circ. Weld ^(e) (Heat # 0227)	0.187	4.4500	82	47	43 ^(e)
Lower Shell Plate C4208-2	0.15	4.4800	94	35	61
Lower Shell Plate C4339-1	0.107	4.4800	101	29	72
Lower Shell Longitudinal Weld L1 and L2 ^(e) (Heat # 8T1762)	0.19	0.8020	64	33	43 ^(e)
Inlet Nozzle 1 to Upper Shell Weld (Heat # 8T1762)	0.19	0.0210	64	14	55
Inlet Nozzle 2 to Upper Shell Weld (Heat # 8T1762)	0.19	0.0048	64	13.5	55
Inlet Nozzle 3 to Upper Shell Weld (Heat # 8T1762)	0.19	0.0066	64	13.5	55
Outlet Nozzle 1 to Upper Shell Weld (Rotterdam)	0.35	0.0049	71	24	54
Outlet Nozzle 2 to Upper Shell Weld (Rotterdam)	0.35	0.0036	71	24	54
Outlet Nozzle 3 to Upper Shell Weld (Rotterdam)	0.35	0.0156	71	24	54
Inlet Nozzle 1 (Heat # 9-5104)	0.159	0.0340	73	12.5	64
Inlet Nozzle 2 (Heat # 9-4815)	0.159	0.0078	66	10	59
Inlet Nozzle 3 (Heat # 9-5205)	0.159	0.0107	67	10	60
Outlet Nozzle 1 (Heat # 9-4825-2)	0.159	0.0080	73	10	66
Outlet Nozzle 2 (Heat # 9-5086-1)	0.159	0.0059	77	10	69
Outlet Nozzle 3 (Heat # 9-5086-2)	0.159	0.0253	71	10.5	64
Position 2.2 ^(d)					
Lower Shell Plate C4339-1	0.107	4.4800	101	19	82
Intermediate Shell Plate C4339-2	0.11	4.4400	83	19	67
Intermediate to Lower Shell Circ. Weld ^(e) (Heat # 0227)	0.187	4.4500	82	42	48 ^(e)

Notes:

- (a) Material data is from Table 4.2.2-3 and Table 4.2.2-4.
- (b) The 1/4T fluence was calculated using the fluence data in Table 4.2.1-2, the Regulatory Guide 1.99 correlation, and the Units 1 and 2 RV wall thickness of 8.05 inches. The surface fluence at the lowest extent of the nozzle weld was used to represent the inlet and outlet nozzle forgings; this approach is conservative. Bounding material fluence values, only, are shown in Figure 5-2 of WCAP-18242-NP for the nozzle materials.
- (c) The Position 1.2 USE decrease values were calculated by plotting the 1/4T fluence values on Figure 2 of Regulatory Guide 1.99 and using the material specific Cu wt. percent values.
- (d) Surveillance data (deemed credible and non-credible per Appendix A of WCAP-18242-NP) from Table 5-12 of WCAP-16001 were used for Unit 2 Position 2.2 USE projections. Regulatory Guide 1.99, Position 2.2 indicates that an upper bound line drawn parallel to the existing lines (in Figure 2 of the Regulatory Guide) through the surveillance data points should be used in preference to the existing graph lines for determining the decrease in USE. Credibility Criterion 3 in the Discussion Section of Regulatory Guide 1.99 indicates that even if the surveillance data are not considered credible for determination of ΔRT_{NDT} , "they may be credible for determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E 185-82." Thus, the surveillance data may be used for Unit 2 USE projections.
- (e) These weld materials were previously addressed by EMA Report BAW-2494 for 48 EFPY. EMAs for these materials have been generated.

Table 4.2.2-7 Reactor Vessel Weld Locations and 80-Year Fluence Projections

RV Material	Material ID and /or Heat Number	(IS) Inside Wetted Surface Fluence or (*) clad/base metal n/cm ² E> 1.0 MeV
Unit 1, 80 Year Fluence (E > 1.0 MeV)		
Nozzle Shell to Outlet Nozzle Forging Welds	SA-1493 (Wire Ht. 8T1762)	(IS) 1.50E+18
	SA-1494 (Wire Ht. 8T1554B)	(IS) 1.50E+18
Nozzle Shell to Inlet Nozzle Forging Welds	SA-1526 (Wire Ht. 299L44)	(IS) 1.50E+18
	SA-1580 (Wire Ht. 8T1762)	(IS) 1.50E+18
Nozzle Shell to Intermediate Shell Circ. Weld	J726 (Wire Ht. 25017)	(*) 7.98E+18
Intermediate Shell Long. Welds (Both)	SA-1494 (Wire Ht. 8T1554)	(*)1.33E+19
Intermediate Shell to Lower Shell Circ. Weld (ID 40%)	SA-1585 (Wire Ht. 72445)	(*)6.67E+19
Intermediate Shell to Lower Shell Circ. Weld (OD 60%)	SA-1650 (Wire Ht. 72445)	NA
Lower Shell Long. Weld (1)	SA-1494 (Wire Ht. 8T1554)	(*)1.34E+19
Lower Shell Long. Weld (2)	SA-1526 (Wire Ht. 299L44)	(*)1.34E+19
Unit 2, 80 Year Fluence (E > 1.0 MeV)		
Nozzle Shell to Outlet Nozzle Forging Welds	Rotterdam	(IS) 1.50E+18
Nozzle Shell to Inlet Nozzle Forging Welds	WF-4 (Wire Ht. 8T1762)	(IS) 1.50E+18
	WF-8 (Wire Ht. 8T1762)	(IS) 1.50E+18
Nozzle Shell to Intermediate Shell Circ. Weld	L737 (Wire Ht. 4275)	(*) 9.21E+18
Intermediate Shell Long. Weld (1), and (2) (100% and OD 50%)	SA-1585 (Wire Ht. 72445)	(*) 1.36E+19
Intermediate Shell Long. Weld (2) (ID 50%)	WF-4 (Wire Ht. 8T1762)	(*) 1.36E+19

Table 4.2.2-7 Reactor Vessel Weld Locations and 80-Year Fluence Projections

RV Material	Material ID and /or Heat Number	(IS) Inside Wetted Surface Fluence or (*) clad/base metal n/cm ² E> 1.0 MeV
Intermediate Shell to Lower Shell Circ. Weld	R3008 (Wire Ht. 0227)	(*) 7.67E+19
Lower Shell Long. Weld (Both)	WF-4 (Wire Ht. 8T1762)	(*) 1.37E+19

Note: No Surry Unit 2 Outlet Nozzle to Upper Shell weld data is available. Generic chemistry values were taken from Regulatory Guide 1.99, Revision 2. The initial RT_{NDT} value was determined using ASME Code, Section III, minimum criteria and BTP 5-3, Position 1.1 guidance. ASME Code, Section III, minimum criteria require measured data; thus, $\sigma_u = 0^\circ\text{F}$. The initial USE value was determined using results from the first surveillance capsule removed and tested from the Surry Unit 2 RV and BTP 5-3, Position 2.1 guidance.

4.2.3 PRESSURIZED THERMAL SHOCK

TLAA Description:

A limiting condition on RV integrity known as Pressurized Thermal Shock (PTS) may occur during a severe system transient such as a small-break loss-of-coolant accident (LOCA) or steam line break. Such transients may challenge the integrity of the RV under the following conditions: severe overcooling of the inside surface of the vessel wall followed by repressurization, significant degradation of vessel material toughness caused by radiation embrittlement, and the presence of a critical-size defect anywhere within the vessel wall.

10 CFR 50.61(b)(1) (Reference 1.7-15) provides rules for protection against PTS events for pressurized water reactors. Licensees are required to perform an updated assessment of the projected values of the PTS reference temperature (RT_{PTS}) whenever there is a significant change in projected values of RT_{PTS} or upon a request for a change in the expiration date for operation of the facility. The current analyses, evaluated for 48 EFPY fluence values predicted for 60 years of operation, are TLAAs requiring evaluation for 80 years since a change in the operating license term of the facility is being requested.

TLAA Evaluation:

10 CFR 50.61(c) provides two methods for determining RT_{PTS}. These methods are also described as Positions 1 and 2 in Regulatory Guide 1.99. Position 1 applies for material without credible surveillance data available and Position 2 is used for material with two or more credible surveillance data sets available. The RT_{PTS} values are calculated for both Positions 1 and 2 by following the guidance in Regulatory Guide 1.99 (Sections 1.1 and 2.1, respectively), using the copper and

nickel content of the Units 1 and 2 beltline materials, and subsequent period of extended operation fluence projections.

These accepted methods were used with the surface fluence values above to calculate the following RT_{PTS} values for the Units 1 and 2 RV materials at 68 EFPY. The subsequent period of extended operation RT_{PTS} calculations are summarized in Table 4.2.3-1 and Table 4.2.3-2 for Units 1 and 2, respectively.

PWROG-16045-NP, summarizes the results and methodologies used in the determination of the unirradiated nil ductility transition temperature (RT_{NDT}) for the Unit 1 and Unit 2 RV materials.

Appendix E of WCAP-18242-NP provides the RT_{PTS} calculations for the beltline and extended beltline materials.

10 CFR 50.61(b)(2) establishes screening criteria for RT_{PTS} as 270°F for plates, forgings, and longitudinal welds and 300°F for circumferential welds.

All of the beltline materials in the Unit 1 and Unit 2 RV are below the RT_{PTS} screening criteria values of 270°F for base metal and longitudinal welds, and 300°F for circumferentially oriented welds through the subsequent period of extended operation (68 EFPY). It is recognized in SECY-82-465, "Pressurized Thermal Shock (PTS)" (Reference 4.8-30), Enclosure A, that the RT_{PTS} screening criteria values of 270°F for base metal and longitudinal welds, and 300°F for circumferentially oriented welds are applicable to cylindrical beltline materials. The adjusted reference temperatures for all extended beltline materials are well below 270°F.

The Units 1 and 2 limiting RT_{PTS} value for base metal or longitudinal weld materials at 68 EFPY is 253.2°F (see and Table 4.2.3-2), which applies to Unit 1 Lower Shell Longitudinal Weld L2 Heat # 299L44 (using credible surveillance data). The Units 1 and 2 limiting RT_{PTS} value for circumferentially oriented welds at 68 EFPY is 229.8°F (see and Table 4.2.3-2), which applies to the Unit 1 Intermediate to Lower Shell Circumferential Weld Heat # 72445.

Appendix E of WCAP-18242-NP provides RT_{PTS} calculations for the nozzle materials. The Units 1 and 2 materials remain below the 10 CFR 50.61 screening criteria.

TAA Disposition: 10 CFR 54.21(c)(1)(ii)

The PTS analyses have been projected to the end of the subsequent period of extended operation.

Table 4.2.3-1 Calculation of Unit 1 RT_{PTS} Values for 68 EFPY at the Clad/Base Metal Interface

RV Material	R.G. 1.99, Rev. 2 Position	Wt.% Cu ^(a)	Wt.% Ni ^(a)	CF ^(a) (°F)	Surface Fluence ^(b) (x 10 ¹⁹ n/cm ² , E > 1.0 MeV)	Surface FF ^(b)	RT _{NDT(U)} ^(c) (°F)	ΔRT _{NDT} ^(d) (°F)	σ _U ^(c) (°F)	σ _Δ ^(e) (°F)	Margin (°F)	RT _{PTS} ^(f) (°F)
RV Beltline Materials												
Upper Shell Forging 122V109VA1	1.1	0.11	0.74	76.1	0.754	0.921	40	70.1	0	17	34	144.1
Upper to Intermediate Shell Circumferential Weld (Heat # 25017)	1.1	0.33	0.1	152	0.754	0.921	0	140	20	28	68.8	208.8
Intermediate Shell Plate C4326-1	1.1	0.11	0.55	73.5	6.29	1.445	10	106.2	0	17	34	150.2
Intermediate Shell Plate C4326-2	1.1	0.11	0.55	73.5	6.29	1.445	11.4	106.2	0	17	34	151.6
Intermediate Shell Longitudinal Welds L3 and L4 (Heat # 8T1554)	1.1	0.16	0.57	167	1.25	1.062	-48.6	177.4	18	28	66.6	195.4
Intermediate to Lower Shell Circumferential Weld (Heat # 72445)	1.1	0.22	0.54	167	6.31	1.445	-72.5	241.4	12	28	60.9	229.8
<i>Using credible surveillance data</i>	2.1	---	---	167	6.31	1.445	-72.5	241.4	12	28	60.9	229.8
Lower Shell Plate C4415-1	1.1	0.102	0.493	66.6	6.35	1.447	20	96.3	0	17	34	150.3
<i>Using credible surveillance data</i>	2.1	---	---	83.1	6.35	1.447	20	120.2	0	8.5	17	157.2
Lower Shell Plate C4415-2	1.1	0.11	0.5	73	6.35	1.447	4.6	105.6	0	17	34	144.2
<i>Using credible surveillance data</i>	2.1	---	---	83.1	6.35	1.447	4.6	120.2	0	8.5	17	141.8
Lower Shell Longitudinal Weld L1 (Heat # 8T1554)	1.1	0.16	0.57	167	1.26	1.064	-48.6	177.8	18	28	66.6	195.7
Lower Shell Longitudinal Weld L2 (Heat # 299L44)	1.1	0.34	0.68	220.6	1.26	1.064	-74.3	234.8	12.8	28	61.6	222.1
<i>Using credible surveillance data</i>	2.1	---	---	249.8	1.26	1.064	-74.3	265.9	12.8	28	61.6	253.2

RV Material	R.G. 1.99, Rev. 2 Position	Wt.% Cu ^(a)	Wt.% Ni ^(a)	CF ^(a) (°F)	Surface Fluence ^(b) (x 10 ¹⁹ n/cm ² , E > 1.0 MeV)	Surface FF ^(b)	RT _{NDT(U)} ^(c) (°F)	ΔRT _{NDT} ^(d) (°F)	σ _U ^(c) (°F)	σ _Δ ^(e) (°F)	Margin (°F)	RT _{PT} ^(f) _S (°F)
RV Extended Beltline Materials												
Inlet Nozzle 1 to Upper Shell Weld (Heat # 299L44)	1.1	0.34	0.68	220.6	0.0304	0.221	-7	48.8	20.6	24.4	63.9	105.7
<i>Using credible surveillance data</i>	2.1	---	---	249.8	0.0304	0.221	-7	55.3	20.6	14	49.8	98.1
Inlet Nozzle 2 to Upper Shell Weld (Heat # 299L44)	1.1	0.34	0.68	220.6	0.00784	0.093	-7	0.0 (20.4)	20.6	0	41.2	34.2
<i>Using credible surveillance data</i>	2.1	---	---	249.8	0.00784	0.093	-7	0.0 (23.2)	20.6	0	41.2	34.2
Inlet Nozzle 3 to Upper Shell Weld (Heat # 299L44)	1.1	0.34	0.68	220.6	0.0109	0.116	-7	25.6	20.6	12.8	48.5	67.2
<i>Using credible surveillance data</i>	2.1	---	---	249.8	0.0109	0.116	-7	29	20.6	14	49.8	71.8
Inlet Nozzle 1 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.0304	0.221	-4.9	33.7	19.7	16.9	51.9	80.7
Inlet Nozzle 2 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.00784	0.093	-4.9	0.0 (14.1)	19.7	0	39.4	34.5
Inlet Nozzle 3 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.0109	0.116	-4.9	17.7	19.7	0	43.2	56.0
Outlet Nozzle 1 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.00813	0.095	-4.9	0.0 (14.5)	19.7	0	39.4	34.5
Outlet Nozzle 2 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.00586	0.075	-4.9	0.0 (11.5)	19.7	0	39.4	34.5
Outlet Nozzle 3 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.0227	0.186	-4.9	28.3	19.7	14.2	48.5	72

RV Material	R.G. 1.99, Rev. 2 Position	Wt.% Cu ^(a)	Wt.% Ni ^(a)	CF ^(a) (°F)	Surface Fluence ^(b) (x 10 ¹⁹ n/cm ² , E > 1.0 MeV)	Surface FF ^(b)	RT _{NDT(U)} ^(c) (°F)	ΔRT _{NDT} ^(d) (°F)	σ _U ^(c) (°F)	σ _Δ ^(e) (°F)	Margin (°F)	RT _{PTS} ^(f) (°F)
Outlet Nozzle 1 to Upper Shell Weld (Heat # 8T1554B)	1.1	0.16	0.57	143.9	0.00813	0.095	-4.9	0.0 (13.7)	19.7	0	39.4	34.5
Outlet Nozzle 2 to Upper Shell Weld (Heat # 8T1554B)	1.1	0.16	0.57	143.9	0.00586	0.075	-4.9	0.0 (10.8)	19.7	0	39.4	34.5
Outlet Nozzle 3 to Upper Shell Weld (Heat # 8T1554B)	1.1	0.16	0.57	143.9	0.0227	0.186	-4.9	26.8	19.7	13.4	47.6	69.5
Inlet Nozzle 1 (Heat # 9-4787)	1.1	0.159	0.85	123.5	0.0304	0.221	10.3	27.3	0	13.7	27.3	65
Inlet Nozzle 2 (Heat # 9-5078)	1.1	0.159	0.87	123.7	0.00784	0.093	11.6	0.0 (11.5)	0	0	0	11.6
Inlet Nozzle 3 (Heat # 9-4819)	1.1	0.159	0.84	123.4	0.0109	0.116	-47.2	14.3	0	7.2	14.3	-18.5
Outlet Nozzle 1 (Heat # 9-4825-1)	1.1	0.159	0.85	123.5	0.00813	0.095	-44.9	0.0 (11.7)	0	0	0	-44.9
Outlet Nozzle 2 (Heat # 9-4762)	1.1	0.159	0.83	123.3	0.00586	0.075	-87.5	0.0 (9.3)	0	0	0	-87.5
Outlet Nozzle 3 (Heat # 9-4788)	1.1	0.159	0.84	123.4	0.0227	0.186	-50.2	0.0 (22.9)	0	11.5	22.9	-4.3

Notes:

- (a) Chemical composition values taken from Table 4.2.2-1 and Table 4.2.2-2. Chemistry factor values taken from Table 3-10 of WCAP-18242-NP.
- (b) Surface fluence values taken from Section 2 of WCAP-18242-NP. FF = fluence factor = $f^{(0.28-0.10 \cdot \log(f))}$.
- (c) Initial RT_{NDT} and σ_u values are taken from Table 4.2.2-1 and Table 4.2.2-2.
- (d) Per NRC RIS 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components," (Reference 4.8-31), embrittlement effects may be neglected for materials with fluence values less than 1.0×10^{17} n/cm² (E > 1.0 MeV). These materials have fluence values at the clad/base metal interface surface less than 1.0×10^{17} n/cm²; therefore, ΔRT_{NDT} values for these materials are set equal to zero. Calculated ΔRT_{NDT} values are listed in parentheses for information purposes only.
- (e) Per Appendix A of WCAP-18242-NP, all Unit 1 surveillance data was deemed credible. Per the guidance of 10 CFR 50.61, the base metal $\sigma_\Delta = 17^\circ\text{F}$ for Position 1.1, and $\sigma_\Delta = 8.5^\circ\text{F}$ for Position 2.1 with credible surveillance data. Also per 10 CFR 50.61, the weld metal $\sigma_\Delta = 28^\circ\text{F}$ for Position 1.1, and with credible surveillance data $\sigma_\Delta = 14^\circ\text{F}$ for Position 2.1. However, σ_Δ need not exceed $0.5 \cdot \Delta RT_{NDT}$. For welds utilizing initial RT_{NDT} values based on BAW-2308, $\sigma_\Delta = 28^\circ\text{F}$ per BAW-2308, (Revision 1-A), SE and BAW-2308, (Revision 2-A) SE, "Final Safety Evaluation for Pressurized Water Reactors Owners Group (PWROG) Topical Report (TR) BAW-2308, (Revision 2), 'Initial RT_{NDT} of Linde 80 Weld Materials'" (Reference 4.8-32).
- (f) RT_{PTS} values calculated in accordance with 10 CFR 50.61 methodology.

Table 4.2.3-2 Calculation of Unit 2 RT_{PTS} Values for 68 EFPY at the Clad/Base Metal Interface

RV Material	R.G. 1.99, Rev. 2 Position	Wt.% Cu ^(a)	Wt.% Ni ^(a)	CF ^(a) (°F)	Surface Fluence ^(b) (x 10 ¹⁹ n/cm ² , E > 1.0 MeV)	Surface FF ^(b)	RT _{NDT(U)} ^(c) (°F)	ΔRT _{NDT} ^(d) (°F)	σ _U ^(c) (°F)	σ _Δ ^(e) (°F)	Margin (°F)	RT _{PTS} ^(f) (°F)
RV Beltline Materials												
Upper Shell Forging 123V303VA1	1.1	0.11	0.72	75.8	0.865	0.959	30	72.7	0	17	34	136.7
Upper to Intermediate Shell Circumferential Weld (Heat # 4275)	1.1	0.35	0.1	160.5	0.865	0.959	0	154	20	28	68.8	222.8
Intermediate Shell Plate C4331-2	1.1	0.12	0.6	83	7.2	1.467	15	121.8	0	17	34	170.8
Intermediate Shell Plate C4339-2	1.1	0.11	0.54	73.4	7.2	1.467	7.8	107.7	0	17	34	149.5
<i>Using non-credible surveillance data</i>	2.1	---	---	75.7	7.2	1.467	7.8	111.1	0	17	34	152.9
Intermediate Shell Longitudinal Welds L3 and L4 (OD 50%) (Heat # 72445)	1.1	0.22	0.54	167	1.29	1.071	-72.5	178.8	12	28	60.9	167.3
<i>Using credible surveillance data</i>	2.1	---	---	167	1.29	1.071	-72.5	178.8	12	28	60.9	167.3
Intermediate Shell Longitudinal Weld L4 (ID 50%) (Heat # 8T1762)	1.1	0.19	0.57	167	1.29	1.071	-48.6	178.8	18	28	66.6	196.8
Intermediate to Lower Shell Circumferential Weld (Heat # 0227)	1.1	0.187	0.545	147.5	7.22	1.468	0	216.5	0	28	56	272.5

Table 4.2.3-2 Calculation of Unit 2 RT_{PTS} Values for 68 EFPY at the Clad/Base Metal Interface

RV Material	R.G. 1.99, Rev. 2 Position	Wt.% Cu ^(a)	Wt.% Ni ^(a)	CF ^(a) (°F)	Surface Fluence ^(b) (x 10 ¹⁹ n/cm ² , E > 1.0 MeV)	Surface FF ^(b)	RT _{NDT(U)} ^(c) (°F)	ΔRT _{NDT} ^(d) (°F)	σ _U ^(c) (°F)	σ _Δ ^(e) (°F)	Margin (°F)	RT _{PTS} ^(f) (°F)
RV Beltline Materials												
<i>Using credible surveillance data</i>	2.1	---	---	132.5	7.22	1.468	0	194.5	0	14	28	222.5
Lower Shell Plate C4208-2	1.1	0.15	0.55	107.3	7.26	1.469	-30	157.6	0	17	34	161.6
Lower Shell Plate C4339-1	1.1	0.107	0.53	70.8	7.26	1.469	-4.4	104	0	17	34	133.6
<i>Using non-credible surveillance data</i>	2.1	---	---	75.7	7.26	1.469	-4.4	111.2	0	17	34	140.8
Lower Shell Longitudinal Welds L1 and L2 (Heat # 8T1762)	1.1	0.19	0.57	167	1.3	1.073	-48.6	179.2	18	28	66.6	197.2

RV Material	R.G. 1.99, Rev. 2 Position	Wt.% Cu ^(a)	Wt.% Ni ^(a)	CF ^(a) (°F)	Surface Fluence ^(b) (x 10 ¹⁹ n/cm ² , E > 1.0 MeV)	Surface FF ^(b)	RT _{NDT(U)} ^(c) (°F)	ΔRT _{NDT} ^(d) (°F)	σ _U ^(c) (°F)	σ _Δ ^(e) (°F)	Margin (°F)	RT _P TS ^(f) (°F)
RV Extended Beltline Materials												
Inlet Nozzle 1 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.034	0.236	-4.9	36	19.7	18	53.4	84.4
Inlet Nozzle 2 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.00784	0.093	-4.9	0.0 (14.1)	19.7	0	39.4	34.5
Inlet Nozzle 3 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.0107	0.115	-4.9	0.0 (17.5)	19.7	8.7	43.1	55.7
Outlet Nozzle 1 to Upper Shell Weld (Rotterdam)	1.1	0.35	1	272	0.00796	0.094	30	0.0 (25.5)	0	0.0	0.0	30.0
Outlet Nozzle 2 to Upper Shell Weld (Rotterdam)	1.1	0.35	1	272	0.00585	0.075	30	0.0 (20.5)	0	0	0	30
Outlet Nozzle 3 to Upper Shell Weld (Rotterdam)	1.1	0.35	1	272	0.0253	0.199	30	54	0	27	54	138
Inlet Nozzle 1 (Heat # 9-5104)	1.1	0.159	0.84	123.4	0.034	0.236	-29.7	29.1	0	14.6	29.1	28.6
Inlet Nozzle 2 (Heat # 9-4815)	1.1	0.159	0.87	123.7	0.00784	0.093	4.5	0.0 (11.5)	0	0	0	4.5
Inlet Nozzle 3 (Heat # 9-5205)	1.1	0.159	0.86	123.6	0.0107	0.115	6.5	14.2	0	7.1	14.2	34.9
Outlet Nozzle 1 (Heat # 9-4825-2)	1.1	0.159	0.85	123.5	0.00796	0.094	-58.1	0.0 (11.6)	0	0	0	-58.1
Outlet Nozzle 2 (Heat # 9-5086-1)	1.1	0.159	0.86	123.6	0.00585	0.075	-26.6	0.0 (9.3)	0	0	0	-26.6
Outlet Nozzle 3 (Heat # 9-5086-2)	1.1	0.159	0.87	123.7	0.0253	0.199	-33.8	24.6	0	12.3	24.6	15.3

Notes:

- (a) Chemical composition values taken from Table 4.2.2-3 and Table 4.2.2-4. Chemistry factor values taken from Table 3-12 of WCAP-18242-NP.
- (b) Surface fluence values taken from Section 2 of WCAP-18242-NP. FF = fluence factor = $f^{(0.28-0.10 \cdot \log(f))}$.
- (c) Initial RT_{NDT} and σ_u values taken from Table 4.2.2-3 and Table 4.2.2-4.
- (d) Per NRC RIS 2014-11, embrittlement effects may be neglected for materials with fluence values less than 1.0×10^{17} n/cm² (E > 1.0 MeV). These materials have fluence values at the clad/base metal interface surface less than 1.0×10^{17} n/cm²; therefore, ΔRT_{NDT} values for these materials are set equal to zero. Calculated ΔRT_{NDT} values are listed in parentheses for information purposes only.
- (e) Per Appendix A of WCAP-18242-NP, the surveillance plate data were deemed non-credible, whereas the surveillance data for the weld materials were deemed credible. Per the guidance of 10 CFR 50.61, the base metal $\sigma_u = 17^\circ\text{F}$ for Position 1.1 and for Position 2.1 with non-credible surveillance data. Per 10 CFR 50.61, the weld metal $\sigma_u = 28^\circ\text{F}$ for Position 1.1, and with credible surveillance data $\sigma_u = 14^\circ\text{F}$ for Position 2.1. However, σ_u need not exceed $0.5 \cdot \Delta RT_{NDT}$. For welds utilizing initial RT_{NDT} values based on BAW-2308, (Revisions 1-A SE and 2-A SE), $\sigma_u = 28^\circ\text{F}$.
- (f) RT_{PTS} values calculated in accordance with 10 CFR 50.61 methodology.

4.2.4 ADJUSTED REFERENCE TEMPERATURE

TLAA Description:

The adjusted reference temperature (ART) of the limiting beltline material is used to adjust the beltline P-T limit curves to account for irradiation effects. Regulatory Guide 1.99 provides the methodology for determining the ART of the limiting material. The initial nil-ductility reference temperature, RT_{NDT} , is the temperature at which a non-irradiated metal (ferritic steel) changes in fracture characteristics from ductile to brittle behavior. Neutron fluence increases the RT_{NDT} beyond its initial value.

RT_{NDT} was evaluated in accordance with PWROG-16045-NP, which includes the generally accepted techniques outlined in:

- ASME Code, Section III, Paragraph NB 2331,
- Branch Technical Position 5-3,
- BWRVIP-173-A, "BWR Vessel and Internals Project: Evaluation of Chemistry Data for BWR Vessel Nozzle Forging Materials" (Reference 4.8-33), and
- BAW-2313.

10 CFR 50, Appendix G, defines the fracture toughness requirements for the vessel. The shift in the initial RT_{NDT} (ΔRT_{NDT}) is evaluated as the difference in the 30 ft-lb index temperatures from the average Charpy curves measured before and after irradiation. This increase (ΔRT_{NDT}) means that higher temperatures are required for the material to continue to act in a ductile manner. The ART is defined as the sum of the initial (unirradiated) reference temperature (Initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin (M) term.

Since the ΔRT_{NDT} value is a function of 48 EFPY fluence, associated with the 60 year licensed operating period, these ART calculations meet the criteria of 10 CFR 54.3(a) and have been identified as TLAA's requiring evaluation for 80 years.

TLAA Evaluation:

As described in Section 4.2.1, 68 EFPY fluence values were determined for the Units 1 and 2 RV beltline and extended beltline components. These 68 EFPY 1/4T fluence values were used to compute the ART values of Units 1 and 2, in accordance with Regulatory Guide 1.99.

Table 4.2.4-1 through 4.2.4-9 summarize the nozzle, and 1/4T ART calculations for Units 1 and 2 at 48 and 68 EFPY. The 3/4T ART values are included in WCAP-18242-NP. The limiting 48 EFPY and 68 EFPY ART values for Units 1 and 2 apply to the Unit 2 Intermediate to Lower Shell Circumferential Weld (using surveillance data).

The inlet and outlet nozzle forging ARTs are necessary to perform a nozzle corner fracture mechanics analysis. The nozzle forging ART calculations utilize the postulated nozzle forging surface 1/4T flaw fluence values in order to provide a conservative estimate of the fluence at the limiting nozzle corner location. The nozzle ART values are also considered herein because the nozzle fluence values for some nozzle materials exceed 1.0×10^{17} n/cm² (E > 1.0 MeV), and thus all of the nozzle forgings are considered part of the extended beltline for conservatism. Since the surface fluence values are utilized for the ART calculations for the nozzle forging materials, the nozzle forgings are omitted from 1/4T ART calculations.

Table 4.2.4-9 compares the TLAA limiting ART values at 48 EFPY and 68 EFPY to the limiting ART values used in development of the existing 48 EFPY P-T limit curves documented in WCAP-14177, "Surry Power Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation" (Reference 4.8-34). The limiting ART values used to develop the existing P-T limit curves are summarized in Table 4.2.4-9. As shown in Table 4.2.4-9, the TLAA limiting ART values at 48 EFPY and 68 EFPY are less than the limiting ART values used to develop the existing P-T limit curves. Appendix B of WCAP-18243-NP shows that the PT curves for the nozzles lie above and to the left of the PT curves for the beltline materials. Thus, the PT curves for the beltline materials are bounding through the subsequent period of extended operation.

TLAA Disposition: 10 CFR 54.21(c)(1)(ii).

The ART analyses have been projected to the end of the subsequent period of extended operation. They may be used as inputs to 68 EFPY P-T limits for the subsequent period of extended operation.

Table 4.2.4-1 Calculation of the Unit 1 Nozzle ART Values at the Surface Location for 48 EFPY

RV Material	R.G. 1.99, Rev. 2 Position	Wt.% Cu ^(a)	Wt.% Ni ^(a)	CF ^(a) (°F)	Surface Fluence ^(b) (x 10 ¹⁹ n/cm ² , E > 1.0 MeV)	Surface FF ^(b)	RT _{NDT} (U) ^(c) (°F)	ΔRT _{NDT} ^(d) (°F)	σ _I ^(c) (°F)	σ _Δ ^(e) (°F)	Margin (°F)	ART ^(f) (°F)
Inlet Nozzle 1 (Heat # 9-4787)	1.1	0.159	0.85	123.5	0.00870	0.100	10.3	0.0 (12.3)	0.0	0.0	0.0	10.3
Inlet Nozzle 2 (Heat # 9-5078)	1.1	0.159	0.87	123.7	0.00219	0.035	11.6	0.0 (4.4)	0.0	0.0	0.0	11.6
Inlet Nozzle 3 (Heat # 9-4819)	1.1	0.159	0.84	123.4	0.00306	0.046	-47.2	0.0 (5.7)	0.0	0.0	0.0	-47.2
Outlet Nozzle 1 (Heat # 9-4825-1)	1.1	0.159	0.85	123.5	0.00237	0.038	-44.9	0.0 (4.6)	0.0	0.0	0.0	-44.9
Outlet Nozzle 2 (Heat # 9-4762)	1.1	0.159	0.83	123.3	0.0017	0.029	-87.5	0.0 (3.5)	0.0	0.0	0.0	-87.5
Outlet Nozzle 3 (Heat # 9-4788)	1.1	0.159	0.84	123.4	0.00672	0.083	-50.2	0.0 (10.3)	0.0	0.0	0.0	-50.2

Notes:

- (a) Chemical composition data taken from Table 4.2.2-1 and Table 4.2.2-2. Chemistry factor values taken from Table 3-10 of WCAP-18242-NP.
- (b) Surface fluence values were from WCAP-18-242-NP. FF = fluence factor = $f^{(0.28-0.10 \cdot \log(f))}$.
- (c) Initial RT_{NDT} values and σ_U values are from Table 4.2.2-2
- (d) Per NRC RIS 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components," embrittlement effects may be neglected for materials with fluence values less than 1.0 x 10¹⁷ n/cm²(E > 1.0 MeV). These materials have fluence values at the clad/base metal interface surface less than 1.0 x 10¹⁷ n/cm²; therefore, ΔRT_{NDT} values for these materials are set equal to zero. Calculated ΔRT_{NDT} values are listed in parentheses for information purposes only.
- (e) Per the guidance of Regulatory Guide 1.99, the base metal σ_Δ = 17°F for Position 1.1. However, σ_Δ need not exceed 0.5*ΔRT_{NDT}.
- (f) ART values calculated in accordance with Regulatory Guide 1.99, Revision 2 methodology.

Table 4.2.4-2 Calculation of the Unit 1 ART Values at the 1/4T Location for 48 EFPY

RV Material	R.G. 1.99, Rev. 2 Position	Wt.% Cu ^(a)	Wt.% Ni ^(a)	CF ^(a) (°F)	1/4T Fluence ^(b) (x 10 ¹⁹ n/cm ² , E > 1.0 MeV)	1/4T FF ^(b)	RT _{NDT(U)} ^(c) (°F)	ΔRT _{NDT} ^(d) (°F)	σ _I ^(c) (°F)	σ _Δ ^(e) (°F)	Margin (°F)	1/4T ART ^(f) (°F)
RV Beltline Materials												
Upper Shell Forging 122V109VA1	1.1	0.11	0.74	76.1	0.329	0.695	40	52.9	0.0	17.0	34.0	126.9
Upper to Intermediate Shell Circumferential Weld (Heat # 25017)	1.1	0.33	0.10	152	0.329	0.695	0	105.6	20.0	28.0	68.8	174.4
Intermediate Shell Plate C4326-1	1.1	0.11	0.55	73.5	2.79	1.274	10	93.6	0.0	17.0	34.0	137.6
Intermediate Shell Plate C4326-2	1.1	0.11	0.55	73.5	2.79	1.274	11.4	93.6	0.0	17.0	34.0	139
Intermediate Shell Longitudinal Welds L3 and L4 (Heat # 8T1554)	1.1	0.16	0.57	167	0.537	0.826	-48.6	138	18.0	28.0	66.6	156.0
Intermediate to Lower Shell Circumferential Weld (Heat # 72445)	1.1	0.22	0.54	167	2.81	1.275	-72.5	212.9	12.0	28.0	60.9	201.3
<i>Using credible surveillance data</i>	2.1	---	---	167	2.81	1.275	-72.5	212.9	12.0	28.0	60.9	201.3
Lower Shell Plate C4415-1	1.1	0.102	0.493	66.6	2.82	1.276	20	85	0.0	17.0	34.0	139
<i>Using credible surveillance data</i>	2.1	---	---	83.1	2.82	1.276	20	106.0	0.0	8.5	17.0	143.0
Lower Shell Plate C4415-2	1.1	0.11	0.5	73	2.82	1.276	4.6	93.1	0.0	17.0	34.0	131.7
<i>Using credible surveillance data</i>	2.1	---	---	83.1	2.82	1.276	4.6	106.0	0.0	8.5	17.0	127.6
Lower Shell Longitudinal Weld L1 (Heat # 8T1554)	1.1	0.16	0.57	167	0.542	0.829	-48.6	138.4	18.0	28.0	66.6	156.4
Lower Shell Longitudinal Weld L2 (Heat # 299L44)	1.1	0.34	0.68	220.6	0.542	0.829	-74.3	182.8	12.8	28.0	61.6	170.1
<i>Using credible surveillance data</i>	2.1	---	---	249.8	0.542	0.829	-74.3	207.0	12.8	28.0	61.6	194.3

RV Material	R.G. 1.99, Rev. 2 Position	Wt.% Cu ^(a)	Wt.% Ni ^(a)	CF ^(a) (°F)	1/4T Fluence ^(b) (x 10 ¹⁹ n/cm ² , E > 1.0 MeV)	1/4T FF ^(b)	RT _{NDT(U)} ^(c) (°F)	ΔRT _{NDT} ^(d) (°F)	σ _I ^(c) (°F)	σ _Δ ^(e) (°F)	Margin (°F)	1/4T ART ^(f) (°F)
RV Extended Beltline Materials												
Inlet Nozzle 1 to Upper Shell Weld (Heat # 299L44)	1.1	0.34	0.68	220.6	0.0131	0.132	-7	29	20.6	14.5	50.4	72.4
<i>Using credible surveillance data</i>	2.1	---	---	249.8	0.0131	0.132	-7	32.9	20.6	14.0	49.8	75.7
Inlet Nozzle 2 to Upper Shell Weld (Heat # 299L44)	1.1	0.34	0.68	220.6	0.00330	0.049	-7	0.0 (10.8)	20.6	0.0	41.2	34.2
<i>Using credible surveillance data</i>	2.1	---	---	249.8	0.00330	0.049	-7	0.0 (12.2)	20.6	0.0	41.2	34.2
Inlet Nozzle 3 to Upper Shell Weld (Heat # 299L44)	1.1	0.34	0.68	220.6	0.00462	0.063	-7	0.0 (13.9)	20.6	0.0	41.2	34.2
<i>Using credible surveillance data</i>	2.1	---	---	249.8	0.00462	0.063	-7	0.0 (15.8)	20.6	0.0	41.2	34.2
Inlet Nozzle 1 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.0131	0.132	-4.9	20.1	19.7	0.0	44.2	59.4
Inlet Nozzle 2 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.00330	0.049	-4.9	0.0 (7.5)	19.7	0.0	39.4	34.5
Inlet Nozzle 3 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.00462	0.063	-4.9	0.0 (9.6)	19.7	0.0	39.4	34.5
Outlet Nozzle 1 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.00345	0.051	-4.9	0.0 (7.7)	19.7	0.0	39.4	34.5
Outlet Nozzle 2 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.00247	0.039	-4.9	0.0 (5.9)	19.7	0.0	39.4	34.5
Outlet Nozzle 3 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.00981	0.108	-4.9	16.5	19.7	0.0	42.7	34.5
Outlet Nozzle 1 to Upper Shell Weld (Heat # 8T1554B)	1.1	0.16	0.57	143.9	0.00345	0.051	-4.9	0.0 (7.3)	19.7	0.0	39.4	34.5
Outlet Nozzle 2 to Upper Shell Weld (Heat # 8T1554B)	1.1	0.16	0.57	143.9	0.00247	0.039	-4.9	0.0 (5.6)	19.7	0.0	39.4	34.5
Outlet Nozzle 3 to Upper Shell Weld (Heat # 8T1554B)	1.1	0.16	0.57	143.9	0.00981	0.108	-4.9	15.6	19.7	7.8	42.4	53.0

Notes:

- (a) Chemical composition data taken from Table 4.2.2-1 and Table 4.2.2-2. Chemistry factor values taken from Table 3-10 of WCAP-18242-NP.
- (b) 48 EPFY surface fluence values were from WCAP-18242-NP. The 1/4T fluence and 1/4T FF were calculated using the Regulatory Guide 1.99 correlations and the Unit 1 RV wall thickness of 8.05 inches.
- (c) Initial RT_{NDT} values and σ_u values are from Table 4.2.2-1 and Table 4.2.2-2.
- (d) Per NRC RIS 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components," embrittlement effects may be neglected for materials with fluence values less than 1.0×10^{17} n/cm² ($E > 1.0$ MeV). These materials have fluence values at the clad/base metal interface surface less than 1.0×10^{17} n/cm²; therefore, ΔRT_{NDT} values for these materials are set equal to zero. Calculated ΔRT_{NDT} values are listed in parentheses for information purposes only.
- (e) As summarized in Appendix A of WCAP-18242-NP, all surveillance data for Unit 1 were deemed credible. Per the guidance of Regulatory Guide 1.99, the base metal $\sigma_\Delta = 17^\circ\text{F}$ for Position 1.1, and $\sigma_\Delta = 8.5^\circ\text{F}$ for Position 2.1 with credible surveillance data. Also per Regulatory Guide 1.99, the weld metal $\sigma_\Delta = 28^\circ\text{F}$ for Position 1.1, and with credible surveillance data $\sigma_\Delta = 14^\circ\text{F}$ for Position 2.1. However, σ_Δ need not exceed $0.5 \cdot \Delta RT_{NDT}$. For welds utilizing initial RT_{NDT} values based on BAW-2308, (Revisions 1 A SE and 2 A SE), $\sigma_\Delta = 28^\circ\text{F}$.
- (f) ART values calculated in accordance with Regulatory Guide 1.99, Revision 2 methodology.

Table 4.2.4-3 Calculation of the Unit 2 Nozzle ART Values at the Surface Location for 48 EFPY

RV Material	R.G. 1.99, Rev. 2 Position	Wt.% Cu ^(a)	Wt.% Ni ^(a)	CF ^(a) (°F)	Surface Fluence ^(b) ($\times 10^{19}$ n/cm ² , E > 1.0 MeV)	Surface FF ^(b)	RT _{NDT(U)} ^(c) (°F)	Δ RT _{NDT} ^(d) (°F)	σ_U ^(c) (°F)	σ_Δ ^(e) (°F)	Margin (°F)	ART ^(f) (°F)
Inlet Nozzle 1 (Heat # 9-5104)	1.1	0.159	0.84	123.4	0.00935	0.105	-29.7	0.0 (12.9)	0.0	0.0	0.0	-29.7
Inlet Nozzle 2 (Heat # 9-4815)	1.1	0.159	0.87	123.7	0.00223	0.036	4.5	0.0 (4.4)	0.0	0.0	0.0	4.5
Inlet Nozzle 3 (Heat # 9-5205)	1.1	0.159	0.86	123.6	0.00304	0.046	6.5	0.0 (5.7)	0.0	0.0	0.0	6.5
Outlet Nozzle 1 (Heat # 9-4825-2)	1.1	0.159	0.85	123.5	0.00235	0.037	-58.1	0.0 (4.6)	0.0	0.0	0.0	-58.1
Outlet Nozzle 2 (Heat # 9-5086-1)	1.1	0.159	0.86	123.6	0.00172	0.029	-26.6	0.0 (3.6)	0.0	0.0	0.0	-26.6
Outlet Nozzle 3 (Heat # 9-5086-2)	1.1	0.159	0.87	123.7	0.00723	0.088	-33.8	0.0 (10.8)	0.0	0.0	0.0	-33.8

Notes:

- (a) Chemical composition values taken from Table 4.2.2-3 and Table 4.2.2-4. Chemistry factor values taken from Table 3-12 of WCAP-18242-NP.
- (b) Surface fluence values were from WCAP-18242-NP. FF = fluence factor = $f^{(0.28-0.10 \cdot \log(f))}$.
- (c) Initial RT_{NDT} values and σ_U values are from Table 4.2.2-4.
- (d) Per NRC RIS 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components," embrittlement effects may be neglected for materials with fluence values less than 1×10^{17} n/cm² (E > 1.0 MeV). These materials have fluence values at the clad/base metal interface surface less than 1×10^{17} n/cm²; therefore, Δ RT_{NDT} values for these materials are set equal to zero. Calculated Δ RT_{NDT} values are listed in parentheses for information purposes only.
- (e) Per the guidance of Regulatory Guide 1.99, the base metal $\sigma_\Delta = 17^\circ\text{F}$ for Position 1. However, σ_Δ need not exceed $0.5 \cdot \Delta$ RT_{NDT}.
- (f) ART values calculated in accordance with Regulatory Guide 1.99, Revision 2 methodology.

Table 4.2.4-4 Calculation of the Unit 2 ART Values at the 1/4T Location for 48 EFPY

RV Material	R.G. 1.99, Rev. 2 Position	Wt.% Cu ^(a)	Wt.% Ni ^(a)	CF ^(a) (°F)	1/4T Fluence ^(b) (x 10 ¹⁹ n/cm ² , E > 1.0 MeV)	1/4T FF ^(b)	RT _{NDT(U)} ^(c) (°F)	ΔRT _{NDT} ^(d) (°F)	σ ₁ ^(c) (°F)	σ _Δ ^(e) (°F)	Margin (°F)	1/4T ART ^(f) (°F)
RV Beltline Materials												
Upper Shell Forging 123V303VA1	1.1	0.11	0.72	75.8	0.362	0.719	30	54.5	0.0	17.0	34.0	118.5
Upper to Intermediate Shell Circumferential Weld (Heat # 4275)	1.1	0.35	0.10	160.5	0.362	0.719	0	115.4	20.0	28.0	68.8	184.2
Intermediate Shell Plate C4331-2	1.1	0.12	0.6	83	3.07	1.296	15	107.6	0.0	17.0	34.0	156.6
Intermediate Shell Plate C4339-2	1.1	0.11	0.54	73.4	3.07	1.296	7.8	95.1	0.0	17.0	34.0	136.9
<i>Using non-credible surveillance data</i>	2.1	---	---	75.7	3.07	1.296	7.8	98.1	0.0	17.0	34.0	139.9
Intermediate Shell Longitudinal Welds L3 and L4 (OD 50%) (Heat # 72445)	1.1	0.22	0.54	167	0.563	0.839	-72.5	140.2	12.0	28.0	60.9	128.6
<i>Using credible surveillance data</i>	2.1	---	---	167	0.563	0.839	-72.5	140.2	12.0	28.0	60.9	128.6
Intermediate Shell Longitudinal Weld L4 (ID 50%) (Heat # 8T1762)	1.1	0.19	0.57	167	0.563	0.839	-48.6	140.2	18.0	28.0	66.6	158.2
Intermediate to Lower Shell Circumferential Weld (Heat # 0227)	1.1	0.187	0.545	147.5	3.07	1.296	0	191.2	0.0	28.0	56.0	247.2
<i>Using credible surveillance data</i>	2.1	---	---	132.5	3.07	1.296	0	171.8	0.0	14.0	28.0	199.8
Lower Shell Plate C4208-2	1.1	0.15	0.55	107.3	3.09	1.298	-30	139.3	0.0	17.0	34.0	143.3
Lower Shell Plate C4339-1	1.1	0.107	0.53	70.8	3.09	1.298	-4.4	91.9	0.0	17.0	34.0	121.5
<i>Using non-credible surveillance data</i>	2.1	---	---	75.7	3.09	1.298	-4.4	98.2	0.0	17.0	34.0	127.8
Lower Shell Longitudinal Welds L1 and L2 (Heat # 8T1762)	1.1	0.19	0.57	167	0.567	0.841	-48.6	140.5	18.0	28.0	66.6	158.5

RV Material	R.G. 1.99, Rev. 2 Position	Wt.% Cu ^(a)	Wt.% Ni ^(a)	CF ^(a) (°F)	1/4T Fluence ^(b) ($\times 10^{19}$ n/cm ² , E > 1.0 MeV)	1/4T FF ^(b)	RT _{NDT(U)} ^(c) (°F)	Δ RT _{NDT} ^(d) (°F)	σ_1 ^(c) (°F)	σ_Δ ^(e) (°F)	Margin (°F)	1/4T ART ^(f) (°F)
RV Extended Beltline Materials												
Inlet Nozzle 1 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.0141	0.138	-4.9	0.0 (21.0)	19.7	0.0	39.4	34.5
Inlet Nozzle 2 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.00336	0.050	-4.9	0.0 (7.6)	19.7	0.0	39.4	34.5
Inlet Nozzle 3 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.0059	0.063	-4.9	0.0 (9.6)	19.7	0.0	39.4	34.5
Outlet Nozzle 1 to Upper Shell Weld (Rotterdam)	1.1	0.35	1	272	0.00342	0.050	30	0.0 (13.7)	0.0	0.0	0.0	30
Outlet Nozzle 2 to Upper Shell Weld (Rotterdam)	1.1	0.35	1	272	0.0025	0.039	30	0.0 (10.7)	0.0	0.0	0.0	30
Outlet Nozzle 3 to Upper Shell Weld (Rotterdam)	1.1	0.35	1	272	0.0105	0.114	30	30.9	0.0	15.5	30.9	91.9

Notes:

- (a) Chemical composition values taken from Table 4.2.2-3 and Table 4.2.2-4. Chemistry factor values taken from Table 3-12 of WCAP-18242-NP.
- (b) 48 EFPY surface fluence values were from WCAP-18242-NP. The 1/4T fluence and 1/4T FF were calculated using the Regulatory Guide 1.99, correlations and the Unit 2 RV wall thickness of 8.05 inches.
- (c) Initial RT_{NDT} values and σ_u values are from Table 4.2.2-3 and Table 4.2.2-4.
- (d) Per NRC RIS 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components," embrittlement effects may be neglected for materials with fluence values less than 1×10^{17} n/cm² (E > 1.0 MeV). These materials have fluence values at the clad/base metal interface surface less than 1×10^{17} n/cm²; therefore, Δ RT_{NDT} values for these materials are set equal to zero. Calculated Δ RT_{NDT} values are listed in parentheses for information purposes only.
- (e) Per Appendix A of WCAP-18242-NP, the surveillance plate data were deemed non credible, whereas the surveillance data for the weld materials were deemed credible. Per the guidance of Regulatory Guide 1.99, the base metal $\sigma_\Delta = 17^\circ\text{F}$ for Position 1.1 and for Position 2.1 with non-credible surveillance data. Also per Regulatory Guide 1.99, the weld metal $\sigma_\Delta = 28^\circ\text{F}$ for Position 1.1, and with credible surveillance data $\sigma_\Delta = 14^\circ\text{F}$ for Position 2.1. However, σ_Δ need not exceed $0.5 \times \Delta$ RT_{NDT}. For welds utilizing initial RT_{NDT} values based on BAW-2308, (Revisions 1-A SE and 2-A SE), $\sigma_\Delta = 28^\circ\text{F}$.
- (f) ART values calculated in accordance with Regulatory Guide 1.99, Revision 2 methodology.

Table 4.2.4-5 Calculation of the Unit 1 Nozzle ART Values at the Surface Location for 68 EFPY

RV Material	R.G. 1.99, Rev. 2 Position	Wt.% Cu ^(a)	Wt.% Ni ^(a)	CF ^(a) (°F)	Surface Fluence ^(b) (x 10 ¹⁹ n/cm ² , E > 1.0 MeV)	Surface FF ^(b)	RT _{NDT(U)} ^(c) (°F)	ΔRT _{NDT} ^(d) (°F)	σ _l ^(c) (°F)	σ _Δ ^(e) (°F)	Margin (°F)	ART ^{(f)(g)} (°F)
Inlet Nozzle 1 (Heat # 9-4787)	1.1	0.159	0.85	123.5	0.0124	0.127	10.3	15.6	0.0	7.8	15.6	41.6
Inlet Nozzle 2 (Heat # 9-5078)	1.1	0.159	0.87	123.7	0.00322	0.048	11.6	0.0 (5.9)	0.0	0.0	0.0	11.6
Inlet Nozzle 3 (Heat # 9-4819)	1.1	0.159	0.84	123.4	0.00446	0.062	-47.2	0.0 (7.6)	0.0	0.0	0.0	-47.2
Outlet Nozzle 1 (Heat # 9-4825-1)	1.1	0.159	0.85	123.5	0.00345	0.051	-44.9	0.0 (6.3)	0.0	0.0	0.0	-44.9
Outlet Nozzle 2 (Heat # 9-4762)	1.1	0.159	0.83	123.3	0.00249	0.039	-87.5	0.0 (4.8)	0.0	0.0	0.0	-87.5
Outlet Nozzle 3 (Heat # 9-4788)	1.1	0.159	0.84	123.4	0.00962	0.107	-50.2	0.0 (13.2)	0.0	0.0	0.0	-50.2

Notes:

- (a) Chemical composition data taken from Table 4.2.2-1 and Table 4.2.2-2. Chemistry factor values taken from Table 3-10 of WCAP-18242-NP.
- (b) Surface fluence values taken from Section 4.2.1. FF = fluence factor = $f^{(0.28-0.10 \cdot \log(f))}$.
- (c) Initial RT_{NDT} values and σ_l values are from Table 4.2.2-2.
- (d) Per NRC RIS 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components," embrittlement effects may be neglected for materials with fluence values less than 1 x 10¹⁷ n/cm² (E > 1.0 MeV). These materials have fluence values at the clad/base metal interface surface less than 1 x 10¹⁷ n/cm²; therefore, ΔRT_{NDT} values for these materials are set equal to zero. Calculated ΔRT_{NDT} values are listed in parentheses for information purposes only.
- (e) Per the guidance of Regulatory Guide 1.99, the base metal σ_Δ = 17°F for Position 1.1. However, σ_Δ need not exceed 0.5*ΔRT_{NDT}.
- (f) Nozzle materials are not limiting for P-T limit curves per WCAP-18243-NP, "Surry Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation" (Reference 4.8-35).
- (g) ART values calculated in accordance with Regulatory Guide 1.99, Revision 2 methodology.

Table 4.2.4-6 Calculation of the Unit 1 ART Values at the 1/4T Location for 68 EFPY

RV Material	R.G. 1.99, Rev. 2 Position	Wt.% Cu ^(a)	Wt.% Ni ^(a)	CF ^(a) (°F)	1/4T Fluence ^(b) (x 10 ¹⁹ n/cm ² , E > 1.0 MeV)	1/4T FF ^(b)	RT _{NDT(U)} ^(c) (°F)	ΔRT _{NDT} ^(d) (°F)	σ ₁ ^(c) (°F)	σ _Δ ^(e) (°F)	Margin (°F)	1/4T ART ^(f) (°F)
RV Beltline Materials												
Upper Shell Forging 122V109VA1	1.1	0.11	0.74	76.1	0.465	0.787	40	59.9	0.0	17.0	34.0	133.9
Upper to Intermediate Shell Circumferential Weld (Heat # 25017)	1.1	0.33	0.10	152	0.465	0.787	0	119.6	20.0	28.0	68.8	188.4
Intermediate Shell Plate C4326-1	1.1	0.11	0.55	73.5	3.88	1.350	10	99.2	0.0	17.0	34.0	143.2
Intermediate Shell Plate C4326-2	1.1	0.11	0.55	73.5	3.88	1.350	11.4	99.2	0.0	17.0	34.0	144.6
Intermediate Shell Longitudinal Welds L3 and L4 (Heat # 8T1554)	1.1	0.16	0.57	167	0.771	0.927	-48.6	154.8	18.0	28.0	66.6	172.8
Intermediate to Lower Shell Circumferential Weld (Heat # 72445)	1.1	0.22	0.54	167	3.89	1.350	-72.5	225.5	12.0	28.0	60.9	213.9
<i>Using credible surveillance data</i>	2.1	---	---	167	3.89	1.350	-72.5	225.5	12.0	28.0	60.9	213.9
Lower Shell Plate C4415-1	1.1	0.102	0.493	66.6	3.92	1.352	20	90	0.0	17.0	34.0	144.0
<i>Using credible surveillance data</i>	2.1	---	---	83.1	3.92	1.352	20	112.3	0.0	8.5	17.0	149.3
Lower Shell Plate C4415-2	1.1	0.11	0.50	73	3.92	1.352	4.6	98.7	0.0	17.0	34.0	137.3
<i>Using credible surveillance data</i>	2.1	---	---	83.1	3.92	1.352	4.6	112.3	0.0	8.5	17.0	133.9
Lower Shell Longitudinal Weld L1 (Heat # 8T1554)	1.1	0.16	0.57	167	0.777	0.929	-48.6	155.2	18.0	28.0	66.6	173.2
Lower Shell Longitudinal Weld L2 (Heat # 299L44)	1.1	0.34	0.68	220.6	0.777	0.929	-74.3	205	12.8	28.0	61.6	192.3

RV Material	R.G. 1.99, Rev. 2 Position	Wt.% Cu ^(a)	Wt.% Ni ^(a)	CF ^(a) (°F)	1/4T Fluence ^(b) (x 10 ¹⁹ n/cm ² , E > 1.0 MeV)	1/4T FF ^(b)	RT _{NDT(U)} ^(c) (°F)	ΔRT _{NDT} ^(d) (°F)	σ _I ^(c) (°F)	σ _Δ ^(e) (°F)	Margin (°F)	1/4T ART ^(f) (°F)
RV Beltline Materials												
<i>Using credible surveillance data</i>	2.1	---	---	249.8	0.777	0.929	-74.3	232.1	12.8	28.0	61.6	219.4
RV Extended Beltline Materials												
Inlet Nozzle 1 to Upper Shell Weld (Heat # 299L44)	1.1	0.34	0.68	220.6	0.0188	0.165	-7.0	36.5	20.6	18.2	55.0	84.5
<i>Using credible surveillance data</i>	2.1	---	---	249.8	0.0188	0.165	-7.0	41.3	20.6	14.0	49.8	84.1
Inlet Nozzle 2 to Upper Shell Weld (Heat # 299L44)	1.1	0.34	0.68	220.6	0.00484	0.065	-7.0	0.0 (14.4)	20.6	0.0	41.2	34.2
<i>Using credible surveillance data</i>	2.1	---	---	249.8	0.00484	0.065	-7.0	0.0 (16.3)	20.6	0.0	41.2	34.2
Inlet Nozzle 3 to Upper Shell Weld (Heat # 299L44)	1.1	0.34	0.68	220.6	0.00672	0.083	-7.0	18.3	20.6	9.2	45.1	56.4
<i>Using credible surveillance data</i>	2.1	---	---	249.8	0.00672	0.083	-7.0	20.8	20.6	10.4	46.1	59.9
Inlet Nozzle 1 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.0188	0.165	-4.9	25.2	19.7	12.6	46.8	67.1
Inlet Nozzle 2 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.00484	0.065	-4.9	0.0 (10.0)	19.7	0.0	39.4	34.5
Inlet Nozzle 3 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.00672	0.083	-4.9	12.7	19.7	6.3	41.4	49.2
Outlet Nozzle 1 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.00502	0.067	-4.9	0.0 (10.2)	19.7	0.0	39.4	34.5
Outlet Nozzle 2 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.00362	0.052	-4.9	0.0 (8.0)	19.7	0.0	39.4	34.5
Outlet Nozzle 3 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.0140	0.137	-4.9	20.9	19.7	10.5	44.6	60.6
Outlet Nozzle 1 to Upper Shell Weld (Heat # 8T1554B)	1.1	0.16	0.57	143.9	0.00502	0.067	-4.9	0.0 (9.7)	19.7	0	39.4	34.5

RV Material	R.G. 1.99, Rev. 2 Position	Wt.% Cu ^(a)	Wt.% Ni ^(a)	CF ^(a) (°F)	1/4T Fluence ^(b) ($\times 10^{19}$ n/cm ² , E > 1.0 MeV)	1/4T FF ^(b)	RT _{NDT(U)} ^(c) (°F)	ΔRT_{NDT} ^(d) (°F)	σ_I ^(c) (°F)	σ_{Δ} ^(e) (°F)	Margin (°F)	1/4T ART ^(f) (°F)
RV Extended Beltline Materials												
Outlet Nozzle 2 to Upper Shell Weld (Heat # 8T1554B)	1.1	0.16	0.57	143.9	0.00362	0.052	-4.9	0.0 (7.6)	19.7	0	39.4	34.5
Outlet Nozzle 3 to Upper Shell Weld (Heat # 8T1554B)	1.1	0.16	0.57	143.9	0.0140	0.137	-4.9	19.7	19.7	9.9	44.1	58.9

Notes:

- (a) Chemical composition data taken from Table 4.2.2-1 and Table 4.2.2-2. Chemistry factor values taken from Table 3-10 of WCAP-18242-NP.
- (b) The 1/4T fluence and 1/4T FF were taken from Table 5-1 of WCAP-18243-NP.
- (c) Initial RT_{NDT} values and σ_u values are from Table 4.2.2-1 and Table 4.2.2-2.
- (d) Per NRC RIS 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components," embrittlement effects may be neglected for materials with fluence values less than 1×10^{17} n/cm² (E > 1.0 MeV). These materials have fluence values at the clad/base metal interface surface less than 1×10^{17} n/cm²; therefore, ΔRT_{NDT} values for these materials are set equal to zero. Calculated ΔRT_{NDT} values are listed in parentheses for information purposes only.
- (e) As summarized in Appendix G of WCAP-18343-NP, all surveillance data for Unit 1 were deemed credible. Per the guidance of Regulatory Guide 1.99 (Revision 2), the base metal $\sigma_{\Delta} = 17^{\circ}\text{F}$ for Position 1.1, and $\sigma_{\Delta} = 8.5^{\circ}\text{F}$ for Position 2.1 with credible surveillance data. Also per Regulatory Guide 1.99 (Revision 2), the weld metal $\sigma_{\Delta} = 28^{\circ}\text{F}$ for Position 1.1, and with credible surveillance data $\sigma_{\Delta} = 14^{\circ}\text{F}$ for Position 2.1. However, σ_{Δ} need not exceed $0.5 \times \Delta RT_{NDT}$. For welds utilizing initial RT_{NDT} values based on BAW-2308, $\sigma_{\Delta} = 28^{\circ}\text{F}$.
- (f) ART values calculated in accordance with Regulatory Guide 1.99 (Revision 2) methodology.

Table 4.2.4-7 Calculation of the Unit 2 ART Nozzle Values at the Surface Location for 68 EFPY

RV Material	R.G. 1.99, Rev. 2 Position	Wt.% Cu ^(a)	Wt.% Ni ^(a)	CF ^(a) (°F)	Surface Fluence ^(b) (x 10 ¹⁹ n/cm ² , E > 1.0 MeV)	Surface FF ^(b)	RT _{NDT(U)} ^(c) (°F)	ΔRT _{NDT} ^(d) (°F)	σ ₁ ^(c) (°F)	σ _Δ ^(e) (°F)	Margin (°F)	ART ^(f) _g (°F)
Inlet Nozzle 1 (Heat # 9-5104)	1.1	0.159	0.84	123.4	0.0139	0.137	-29.7	16.8	0.0	8.4	16.8	4.0
Inlet Nozzle 2 (Heat # 9-4815)	1.1	0.159	0.87	123.7	0.00321	0.048	4.5	0.0 (5.9)	0.0	0.0	0.0	4.5
Inlet Nozzle 3 (Heat # 9-5205)	1.1	0.159	0.86	123.6	0.00437	0.061	6.5	0.0 (7.5)	0.0	0.0	0.0	6.5
Outlet Nozzle 1 (Heat # 9-4825-2)	1.1	0.159	0.85	123.5	0.00338	0.05	-58.1	0.0 (6.2)	0.0	0.0	0.0	-58.1
Outlet Nozzle 2 (Heat # 9-5086-1)	1.1	0.159	0.86	123.6	0.00248	0.039	-26.6	0.0 (4.8)	0.0	0.0	0.0	-26.6
Outlet Nozzle 3 (Heat # 9-5086-2)	1.1	0.159	0.87	123.7	0.0107	0.115	-33.8	14.2	0.0	7.1	14.2	-5.4

Notes:

- (a) Chemical composition values taken from Table 4.2.2-3 and Table 4.2.2-4. Chemistry factor values taken from Table 3-12 of WCAP-18242-NP.
- (b) Surface fluence values taken from Section 4.2.1. FF = fluence factor = $f^{(0.28-0.10 \log(f))}$.
- (c) Initial RT_{NDT} values and σ_u values are from Table 4.2.2-4.
- (d) Per NRC RIS 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components," embrittlement effects may be neglected for materials with fluence values less than 1 x 10¹⁷ n/cm²(E > 1.0 MeV). These materials have fluence values at the clad/base metal interface surface less than 1 x 10¹⁷ n/cm²; therefore, ΔRT_{NDT} values for these materials are set equal to zero. Calculated ΔRT_{NDT} values are listed in parentheses for information purposes only.
- (e) Per the guidance of Regulatory Guide 1.99, the base metal σ_Δ = 17°F for Position 1.1. However, σ_Δ need not exceed 0.5*ΔRT_{NDT}.
- (f) Nozzle materials are not limiting for P-T limit curves per WCAP-18243-NP.
- (g) ART values calculated in accordance with Regulatory Guide 1.99 (Revision 2) methodology.

Table 4.2.4-8 Calculation of the Unit 2 ART Values at the 1/4T Location for 68 EFPY

RV Material	R.G. 1.99, Rev. 2 Position	Wt.% Cu ^(a)	Wt.% Ni ^(a)	CF ^(a) (°F)	1/4T Fluence ^(b) (x 10 ¹⁹ n/cm ² , E > 1.0 MeV)	1/4T FF ^(b)	RT _{NDT(U)} ^(c) (°F)	ΔRT _{NDT} ^(d) (°F)	σ ₁ ^(c) (°F)	σ _Δ ^(e) (°F)	Margin (°F)	1/4T ART ^(f) (°F)
RV Beltline Materials												
Upper Shell Forging 123V303VA1	1.1	0.11	0.72	75.8	0.534	0.825	30	62.5	0.0	17.0	34.0	126.5
Upper to Intermediate Shell Circumferential Weld (Heat # 4275)	1.1	0.35	0.10	160.5	0.534	0.825	0	132.3	20.0	28.0	68.8	201.2
Intermediate Shell Plate C4331-	1.1	0.12	0.60	83.0	4.44	1.378	15	114.4	0.0	17.0	34.0	163.4
Intermediate Shell Plate C4339-2	1.1	0.11	0.54	73.4	4.44	1.378	7.8	101.2	0.0	17.0	34.0	143.0
<i>Using non-credible surveillance data</i>	2.1	---	---	75.7	4.44	1.378	7.8	104.3	0.0	17.0	34.0	146.1
Intermediate Shell Longitudinal Welds L3 and L4 (OD 50%) (Heat # 72445)	1.1	0.22	0.54	167.0	0.796	0.936	-72.5	156.3	12.0	28.0	60.9	144.7
<i>Using credible surveillance data</i>	2.1	---	---	167.0	0.796	0.936	-72.5	156.3	12.0	28.0	60.9	144.7
Intermediate Shell Longitudinal Weld L4 (ID 50%) (Heat # 8T1762)	1.1	0.19	0.57	167.0	0.796	0.936	-48.6	156.3	18.0	28.0	66.6	174.3
Intermediate to Lower Shell Circumferential Weld (Heat # 0227)	1.1	0.187	0.545	147.5	4.45	1.379	0	203.4	0.0	28.0	56.0	259.4
<i>Using credible surveillance data</i>	2.1	---	---	132.5	4.45	1.379	0	182.7	0.0	14.0	28.0	210.7
Lower Shell Plate C4208-2	1.1	0.15	0.55	107.3	4.48	1.380	-30	148.1	0.0	17.0	34.0	152.1
Lower Shell Plate C4339-1	1.1	0.107	0.53	70.8	4.48	1.380	-4.4	97.7	0.0	17.0	34.0	127.3

RV Material	R.G. 1.99, Rev. 2 Position	Wt.% Cu ^(a)	Wt.% Ni ^(a)	CF ^(a) (°F)	1/4T Fluence ^(b) (x 10 ¹⁹ n/cm ² , E > 1.0 MeV)	1/4T FF ^(b)	RT _{NDT(U)} ^(c) (°F)	ΔRT _{NDT} ^(d) (°F)	σ ₁ ^(c) (°F)	σ _Δ ^(e) (°F)	Margin (°F)	1/4T ART ^(f) (°F)
<i>Using non-credible surveillance data</i>	2.1	---	---	75.7	4.48	1.380	-4.4	104.5	0.0	17.0	34.0	134.1
Lower Shell Longitudinal Welds L1 and L2 (Heat # 8T1762)	1.1	0.19	0.57	167.0	0.802	0.938	-48.6	156.7	18.0	28.0	66.6	174.6
RV Extended Beltline Materials												
Inlet Nozzle 1 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.0210	0.177	-4.9	27	19.7	13.5	47.8	69.9
Inlet Nozzle 2 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.00484	0.065	-4.9	0.0 (10.0)	19.7	0.0	39.4	34.5
Inlet Nozzle 3 to Upper Shell Weld (Heat # 8T1762)	1.1	0.19	0.57	152.4	0.00660	0.082	-4.9	12.5	19.7	6.3	41.3	48.9
Outlet Nozzle 1 to Upper Shell Weld (Rotterdam)	1.1	0.35	1.0	272.0	0.00491	0.066	30	0.0 (18.0)	0.0	0.0	0.0	30.0
Outlet Nozzle 2 to Upper Shell Weld (Rotterdam)	1.1	0.35	1.0	272.0	0.00361	0.052	30	0.0 (14.3)	0.0	0.0	0.0	30.0
Outlet Nozzle 3 to Upper Shell Weld (Rotterdam)	1.1	0.35	1.0	272.0	0.0156	0.147	30	40.0	0.0	20.0	40.0	110.0

Notes:

- (a) Chemical composition values taken from Table 4.2.2-3 and Table 4.2.2-4. Chemistry factor values taken from Table 3-12 of WCAP-18243-NP.
- (b) The 1/4T fluence and 1/4T FF were taken from Table 5-2 of WCAP-18343-NP.
- (c) Initial RT_{NDT} values and σ_U values are from Table 4.2.2-3 and Table 4.2.2-4.
- (d) Per NRC RIS 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components," embrittlement effects may be neglected for materials with fluence values less than 1×10^{17} n/cm² (E > 1.0 MeV). These materials have fluence values at the clad/base metal interface surface less than 1×10^{17} n/cm²; therefore, ΔRT_{NDT} values for these materials are set equal to zero. Calculated ΔRT_{NDT} values are listed in parentheses for information purposes only.
- (e) As summarized in Appendix G of WCAP-18243-NP, the surveillance plate data were deemed non-credible, whereas the surveillance data for the weld materials were deemed credible. Per the guidance of Regulatory Guide 1.99 (Revision 2), the base metal $\sigma_\Delta = 17^\circ\text{F}$ for Position 1.1 and Position 2.1 with non-credible surveillance data. Also per Regulatory Guide 1.99 (Revision 2), the weld metal $\sigma_\Delta = 28^\circ\text{F}$ for Position 1.1, and with credible surveillance data $\sigma_\Delta = 14^\circ\text{F}$ for Position 2.1. However, σ_Δ need not exceed $0.5 \cdot \Delta RT_{NDT}$. For welds utilizing initial RT_{NDT} values based on BAW-2308, $\sigma_\Delta = 28^\circ\text{F}$.
- (f) ART values calculated in accordance with Regulatory Guide 1.99 (Revision 2) methodology.

Table 4.2.4-9 Summary of the Units 1 and 2 Limiting ART Values Used in the Applicability Evaluation of the Reactor Vessel Heatup and Cooldown Curves

Plant	Limiting Material	1/4T Limiting ART (°F)			3/4T Limiting ART (°F)		
		Existing 48 EFPY Curves Documented in WCAP-14177, Rev. 0 ^(a)	TAA Evaluation at 48 EFPY	TAA Evaluation at 68 EFPY	Existing 48 EFPY Curves Documented in WCAP-14177, Rev. 0 ^(a)	TAA Evaluation at 48 EFPY	TAA Evaluation at 68 EFPY
SPS Unit 1	(Circ Flow) Circ. Weld: Intermediate to Lower Shell Circ. Weld, Heat # 72445	228.4	<u>201.3</u>	213.9	189.5	158.5	173.6
	(Axial Flow) Long. Weld: Lower Shell Long. Weld L2 Heat # 299L44 (Position 2.1)		194.3	<u>219.4</u>		131.3	153.8
SPS Unit 2	(Circ Flow) Circ. Weld: Intermediate to Lower Shell Circ. Weld, Heat # 0227 (Position 2.1)		199.8	210.7		<u>166.3</u>	<u>179.8</u>
	(Axial Flow) Plate: Intermediate Shell Plate C4331-2		156.6 ^(b)	163.4 ^(c)		135.6	144.0
	(Axial Flow) Weld: Lower Shell Longitudinal Weld L1 and L2 Heat # 8T1762		158.5	174.6		116.2 ^(d)	130.7 ^(e)

Notes: Limiting values depicted as bold and underlined.

- (a) The limiting 48 EFPY 1/4T and 3/4T ART values in the Technical Specifications correspond to the Unit 1 Intermediate to Lower Shell Circumferential Weld (Heat # 72445). The basis for the P-T limit curves is contained in WCAP-14177; however, the applicability was extended to 48 EFPY in a later analysis. See Appendix C of WCAP-18242-NP for details.
- (b) Value from Table 6.1-5 of WCAP-18242-NP.
- (c) Value from Table 6.1-11 of WCAP-18242-NP.
- (d) Value from Table 6.1-6 of WCAP-18242-NP.
- (e) Value from Table 6.1-12 of WCAP-18242-NP.

4.2.5 PRESSURE-TEMPERATURE LIMITS

TLAA Description:

10 CFR 50 Appendix G requires that the RV be maintained within established pressure-temperature (P-T) limits, including heatup and cooldown operations. These limits specify the maximum allowable pressure as a function of reactor coolant temperature. As the RV is exposed to increased neutron irradiation, its fracture toughness is reduced. The P-T limits must account for the anticipated RV fluence.

The current P-T limits are based upon fluence projections for 60 years of plant operation. Because they were based upon a fluence assumption of 60 years of operation, the P-T limits analyses meet the definition of 10 CFR 54.3(a) (Reference 1.7-2) and have been identified as TLAA's.

TLAA Evaluation:

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} corresponding to the limiting material in the beltline region of the RV. The most limiting RT_{NDT} of the material in the core region (beltline) of the RV is determined by using the unirradiated RV material fracture toughness properties and estimating the irradiation induced shift (ΔRT_{NDT}).

RT_{NDT} increases as the material is exposed to fast neutron irradiation; therefore, to find the most limiting core region (beltline) RT_{NDT} at any time, ΔRT_{NDT} due to the neutron radiation exposure associated with that time must be added to the original unirradiated RT_{NDT} . Using the ART values, P-T limit curves are determined in accordance with the requirements of 10 CFR Part 50, Appendix G, as augmented by ASME Code, Section XI, Appendix G.

The current P-T limits for Units 1 and 2 are based on the K_{Ia} methodology and the latest fluence data through 48 EFPY and are maintained in the Technical Specifications.

According to NUREG-2192, Section 4.2.2.1.4, the P-T limits for the subsequent period of extended operation need not be submitted as part of the SLRA since the P-T limits are required to be updated through the 10 CFR 50.90 licensing process when necessary for P-T limits that are located in the Technical Specifications. The current licensing basis will ensure that the P-T limits for the subsequent period of extended operation will be updated prior to exceeding the EFPY for which they remain valid.

Nozzle materials were evaluated in WCAP-18242-NP at 48 EFPY and 68 EFPY; the nozzle forging materials evaluated are documented in Tables 4.2.4-1, 4.2.4-3, 4.2.2-5, and e. All nozzle materials were assigned the fluence values at the postulated 1/4T flaw location for each specific nozzle in Table 4.2.1-1 and Table 4.2.1-2. Thus, Unit 1 Inlet Nozzle 1 and Unit 2 Inlet Nozzle 1 and Outlet Nozzle 3 have neutron fluence values greater than 1.0×10^{17} n/cm² ($E > 1.0$ MeV) at 68 EFPY. In order to fully assess the Units 1 and 2 P-T limit curves applicability to 68 EFPY, a nozzle corner fracture mechanics analysis was completed for all nozzle materials. These nozzle P-T limit curves

were generated and compared to the beltline P-T limit curves to ensure that the beltline curves are bounding. The detailed nozzle forging fracture mechanics evaluation and comparison to the applicable RV beltline P-T limit curves were documented in WCAP-18243-NP. The current beltline curves were confirmed to remain more limiting than the nozzle curves through 68 EFPY.

The development of the current P-T limit curves for normal heatup and cooldown of the primary reactor coolant system for Units 1 and 2 was documented in WCAP-14177. The existing P-T limit curves are based on the K_{Ic} methodology and the limiting beltline material ART values, which are influenced by both the fluence and the initial material properties of that material. The Units 1 and 2 P-T limit curves were developed by calculating ART values utilizing the vessel fluence at the clad/base metal interface corresponding to each RV material. Since the development of the curves, the applicability of the curves has been extended and the fluence values and initial material properties used to calculate ART values have been updated.

The K_{Ic} methodology was used to confirm the applicability of the P-T limit curves developed based on WCAP-14177. The limiting RV material ART values with consideration of the updated 68 EFPY fluence values, revised Position 2.1 chemistry factor values, and updated initial RT_{NDT} values must be shown to be less than or equal to the limiting beltline material ART values used in development of the P-T limit curves contained in WCAP-14177 and the Units 1 and 2 Technical Specifications. The Regulatory Guide 1.99 methodology was used along with the surface fluence of Section 2 of WCAP-18242-NP to calculate ART values for the Units 1 and 2 RV materials at 48 EFPY and 68 EFPY.

Comparisons of the use of the K_{Ic} reference stress intensity factor, instead of the older, more conservative K_{Ia} reference stress intensity factor were conducted to validate that the PT limits for 48 EFPY are conservative for operation through the subsequent period of extended operation. The comparisons of the limiting ART values calculated as part of this RV integrity TLAA evaluation, using updated fluence and initial material properties, to those used in calculation of the existing P-T limit curves are contained in Table 4.2.4-9 for Units 1 and 2. With the consideration of TLAA fluence projections, the applicability of the P-T limit curves in WCAP-14177 may be extended to 68 EFPY for the Units 1 and 2 cylindrical shell materials. Nozzle P-T limit curves were developed per WCAP-18243-NP and compared to the cylindrical shell beltline curves. ART values were generated without the consideration of the methodology in TLR-RES/DE/CIB-2013-01, "Evaluation of the Beltline Region for Nuclear Reactor Pressure Vessels, U.S. NRC Technical Letter Report, Office of Nuclear Regulatory Research [RES]" (Reference 4.8-36). Per WCAP-18243-NP, the applicability of the P-T limit curves may be extended through SLR, because the current Technical Specifications P-T limit curves bound the new P-T limit curves developed in WCAP-18243-NP regardless of the use of the TLR-RES/DE/CIB-2013-01 methodology. Per WCAP-18243-NP, the applicability of the P-T limit curves may be extended through the subsequent period of extended operation.

In addition, the applicable RV flange and closure head initial RT_{NDT} values are bounding and the P-T limit curves flange notch requires no change or further consideration. Finally, the lowest service temperature requirements are not applicable to Units 1 and 2, because the plants are Westinghouse-designed per ASME Code, Section III, and utilize stainless steel reactor coolant system piping.

TLAA Disposition: 10 CFR 54.21(c)(1)(iii)

Since the P-T limits will be updated through the 10 CFR 50.90 process at a later, appropriate date, the effects of aging on the intended function(s) of the RVs will be adequately managed for the subsequent period of extended operation. The *Reactor Vessel Material Surveillance* program (B2.1.19) and plant Technical Specifications will ensure that updated P-T limits based upon updated ART values will be submitted to the NRC for approval prior to exceeding the period of applicability for Units 1 and 2.

4.2.6 LOW TEMPERATURE OVERPRESSURE PROTECTION

TLAA Description:

Low temperature overpressure protection (LTOP) system (sometimes referred to as the Reactor Coolant System Overpressure Mitigating System, or the RV Overpressure Mitigating System) at Unit 1 and Unit 2 is required by Technical Specification Limited Condition for Operation 3.1.G. Two pressurizer power operated relief valves (PORV) provide the automatic relief capability during the design basis mass input and the design basis heat input transients to automatically prevent the reactor coolant system pressure from exceeding the P-T limit curves based on 10 CFR 50, Appendix G.

LTOP system setpoints are based on the P-T limits calculation which is a TLAA.

TLAA Evaluation:

The LTOP enabling temperature has been determined for 68 EFPY as discussed in Appendix D of WCAP-18243-NP. Using Code Case N-514, the LTOP enabling temperature is 283°F. The Surry Technical Specification 3.1.G.1.c.(4) specifies an arming temperature of 350°F which is conservative and remains valid for the subsequent period of extended operation.

In WCAP-18242-NP the maximum allowable LTOP system PORV setpoint was calculated to be 399.6 psig for the Units 1 and 2 subsequent period of extended operation. The calculation was performed in accordance with the WCAP-14040-A methodology using LTOP input parameters and the limiting axial flaw steady state ASME Code, Section XI, Appendix G limits calculated for the subsequent period of extended operation at 68 EFPY.

The evaluation showed that the current Technical Specification value of ≤ 390.0 psig is bounding and will remain valid for the subsequent period of extended operation. Since the maximum allowable PORV setpoint was determined using the methodology in WCAP-14040-A, this demonstrates that the current licensing basis PORV setpoint that was developed using K_{Ia} ASME Code, Section XI, Appendix G limits without applying uncertainties was sufficiently conservative.

TLAA Disposition: 10 CFR 54.21(c)(1)(ii)

The LTOP system setpoint and enabling temperature have been projected to the end of the subsequent period of extended operation.

Attachment 3

MARKED-UP TECHNICAL SPECIFICATIONS AND BASES PAGES

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

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Insert 1

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 heatup and cooldown limit curves were previously calculated using the most limiting value of RT_{NDT} (228.4°F) which occurred at the 1/4-T, 0° azimuthal location in the Unit 1 intermediate-to-lower shell circumferential weld. Subsequently, the reactor vessel material property basis was amended based upon new data which showed that the most limiting value of RT_{NDT} (222.5°F) at 48 EFPY occurs at the 1/4 T, 0° azimuthal location in the Unit 2 intermediate to lower shell circumferential weld. The revised limiting material property (i.e., Unit 2 RT_{NDT} of 222.5°F) justified continued use of the existing heatup and cooldown limit curves (based on the Unit 1 RT_{NDT} of 228.4°F) to 48 EFPY for Units 1 and 2. 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.

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Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of one and one half T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against non-ductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil ductility reference temperature, RT_{NDT} , is used and this includes the radiation-induced shift, ΔRT_{NDT} , corresponding to the end of the period for which heatup and cooldown curves are generated.

in the 1986 Edition of the ASME Code

The ~~ASME~~ approach for calculating the allowable limit curves for various heatup and cooldown rates ~~specifies~~ that the total stress intensity factor, K_I , for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR} , for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T - RT_{NDT} + 160)] \quad (1)$$

where K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil ductility reference temperature RT_{NDT} . Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C K_{IM} + K_{It} \leq K_{IR} \quad (1)$$

where, K_{IM} is the stress intensity factor caused by membrane (pressure) stress.

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.

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Insert 2

The reactor boltup temperature is defined in 10 CFR 50, Appendix G as “The highest reference temperature of the material in the closure flange region that is highly stressed by the bolt preload.” The reactor vessel may be bolted up at a temperature greater than the initial RT_{NDT} of the material stressed by the boltup (e.g., the vessel flange). As noted on Figures 3.1-1 and 3.1-2, the limiting boltup temperature is 10°F. An administrative minimum boltup temperature limit greater than 10°F is imposed in station procedures to ensure the Reactor Coolant System temperatures are sufficiently high to prevent damage to the reactor vessel closure head/vessel flange during the removal or installation of reactor vessel head bolts. The limiting boltup temperature and the administrative minimum boltup temperature limit are in effect when the reactor vessel head bolts are under tension.

References

- (1) UFSAR, Section 4.1, Design Bases
- (2) WCAP-14177, "Surry Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," (October 1994)
- (3) WCAP-18243, Rev. 2, "Surry Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," (July 2018)

Surry Units 1 and 2 Reactor Coolant System Heatup Limitations

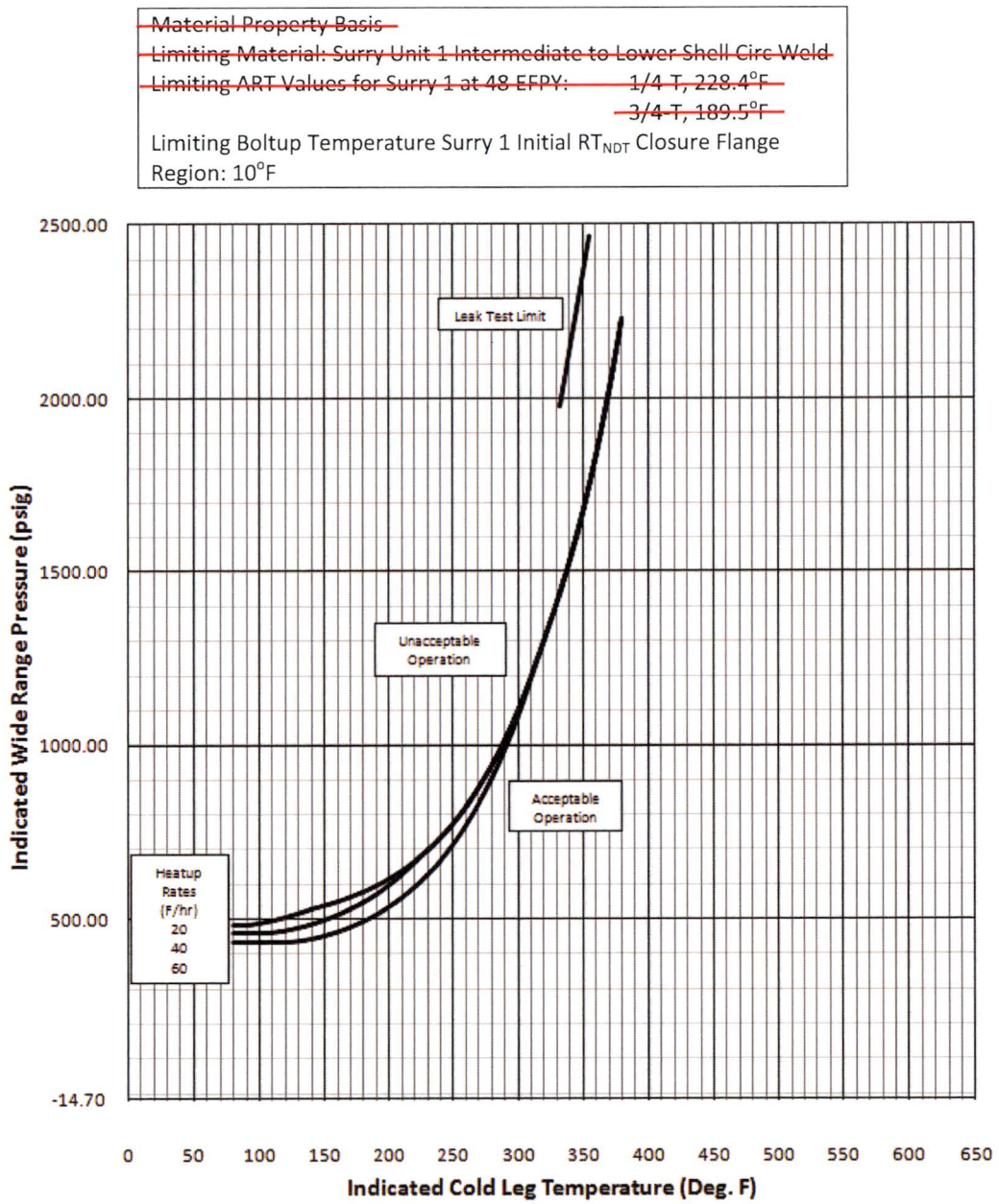


Figure 3.1-1 : Surry Units 1 and 2 Reactor Coolant System Heatup Limitations (Heatup Rates up to 60°F/hr) Applicable for ~~48~~ EFPY

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Surry Units 1 and 2 Reactor Coolant System Cooldown Limitations

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

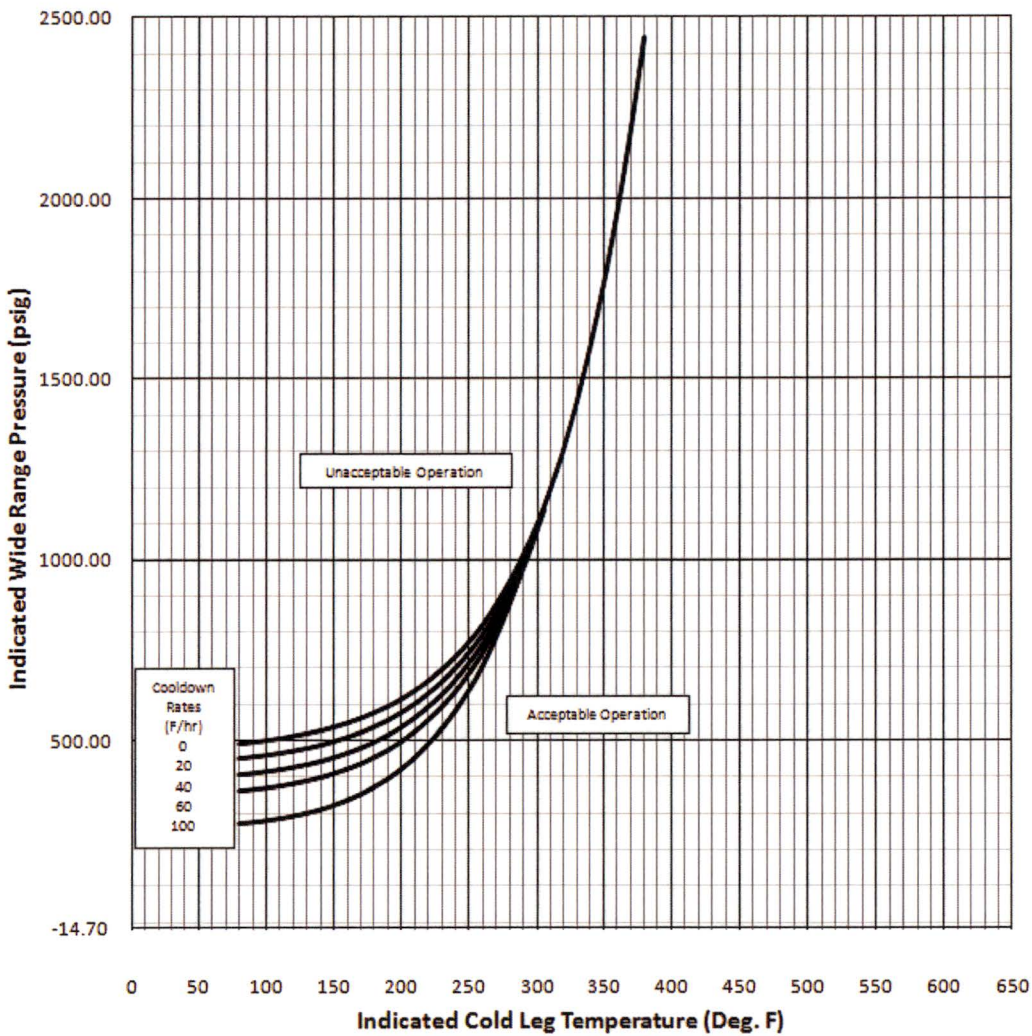


Figure 3.1-2 : Surry Units 1 and 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for ~~48~~ EFPY

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TS 3.1 BASIS INSERTS

INSERT 1

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.

INSERT 2

The technical basis for the data points and the associated RT_{NDT} values used to generate the heatup and cooldown curves is provided in WCAP-14177 (Reference 2) and were determined to be applicable to the 48 EFPY period of extended operation under first license renewal. The associated RT_{NDT} values used to calculate the heatup and cooldown curves provided in WCAP-14177 (Revision 2) are based upon the Surry Unit 1 Intermediate to Lower Shell Circ Weld:

1/4-T, 228.4°F and
3/4-T, 189.5°F

The heatup and cooldown curves for operation through 48 EFPY were based upon the K_{Ir} methodology. These heatup and cooldown curves were subsequently evaluated using the K_{Ic} methodology for Subsequent License Renewal (SLR) at 68 EFPY in WCAP-18243-NP (Reference 3).

The limiting reactor vessel materials at 68 EFPY were determined to be the Surry Unit 1 Lower Shell Longitudinal Weld L2 at 1/4-T and the Surry Unit 2 Intermediate to Lower Shell Circumferential Weld at 3/4-T. The associated RT_{NDT} values calculated at 68 EFPY are:

1/4-T, 219.4 °F and
3/4-T, 179.8 °F

The data points and the associated RT_{NDT} values used to generate the heatup and cooldown curves in TS Figures 3.1-1 and 3.1-2, respectively, are conservative based upon use of the K_{Ic} methodology. Therefore, the heatup and cooldown curves did not require revision as a result of SLR. However, the fluence applicability is updated from 48 EFPY to 68 EFPY.

Attachment 4

PROPOSED TECHNICAL SPECIFICATIONS AND BASES PAGES

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

Heatup and cooldown limit curves are calculated using a bounding value of the nil-ductility reference temperature, RT_{NDT} , at the end of 68 Effective Full Power Years (EFPY) for Units 1 and 2. The heatup and cooldown limit curves were calculated using the most limiting value of RT_{NDT} (228.4°F) which occurred at the 1/4-T, 0° azimuthal location in the Unit 1 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 68 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 68 EFPY for Units 1 and 2 prior to a scheduled refueling outage.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T , and a length of one and one half T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against non-ductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil ductility reference temperature, RT_{NDT} , is used and this includes the radiation-induced shift, ΔRT_{NDT} , corresponding to the end of the period for which heatup and cooldown curves are generated.

The approach for calculating the allowable limit curves for various heatup and cooldown rates in the 1986 Edition of the ASME Code specifies that the total stress intensity factor, K_I , for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR} , for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T - RT_{NDT} + 160)] \quad (1)$$

where K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil ductility reference temperature RT_{NDT} . Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C K_{IM} + K_{It} \leq K_{IR} \quad (1)$$

where, K_{IM} is the stress intensity factor caused by membrane (pressure) stress.

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 68 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.

The technical basis for the data points and the associated RT_{NDT} values used to generate the heatup and cooldown curves is provided in WCAP-14177 (Reference 2) and were determined to be applicable to the 48 EFPY period of extended operation under first license renewal. The associated RT_{NDT} values used to calculate the heatup and cooldown curves provided in WCAP-14177 (Revision 2) are based upon the Surry Unit 1 Intermediate to Lower Shell Circ Weld:

1/4-T, 228.4°F and

3/4-T, 189.5°F

The heatup and cooldown curves for operation through 48 EFPY were based upon the K_{Ir} methodology. These heatup and cooldown curves were subsequently evaluated using the K_{Ic} methodology for Subsequent License Renewal (SLR) at 68 EFPY in WCAP-18243-NP (Reference 3).

The limiting reactor vessel materials at 68 EFPY were determined to be the Surry Unit 1 Lower Shell Longitudinal Weld L2 at 1/4-T and the Surry Unit 2 Intermediate to Lower Shell Circumferential Weld at 3/4-T. The associated RT_{NDT} values calculated at 68 EFPY are:

1/4-T, 219.4°F and

3/4-T, 179.8°F

The data points and the associated RT_{NDT} values used to generate the heatup and cooldown curves in TS Figures 3.1-1 and 3.1-2, respectively, are conservative based upon use of the K_{Ic} methodology. Therefore, the heatup and cooldown curves did not require revision as a result of SLR. However, the fluence applicability is updated from 48 EFPY to 68 EFPY.

The reactor boltup temperature is defined in 10 CFR 50, Appendix G as “The highest reference temperature of the material in the closure flange region that is highly stressed by the bolt preload.” The reactor vessel may be bolted up at a temperature greater than the initial RT_{NDT} of the material stressed by the boltup (e.g., the vessel flange). As noted on Figures 3.1-1 and 3.1-2, the limiting boltup temperature is 10°F. An administrative minimum boltup temperature limit greater than 10°F is imposed in station procedures to ensure the Reactor Coolant System temperatures are sufficiently high to prevent damage to the reactor vessel closure head/vessel flange during the removal or installation of reactor vessel head bolts. The limiting boltup temperature and the administrative minimum boltup temperature limit are in effect when the reactor vessel head bolts are under tension.

References

- (1) UFSAR, Section 4.1, Design Bases
- (2) WCAP-14177, “Surry Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation,” (October 1994)
- (3) WCAP-18243, Rev. 2, “Surry Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation,” (July 2018)

Surry Units 1 and 2 Reactor Coolant System Heatup Limitations

Limiting Boltup Temperature Surry 1 Initial RT_{NDT} Closure Flange Region: 10°F

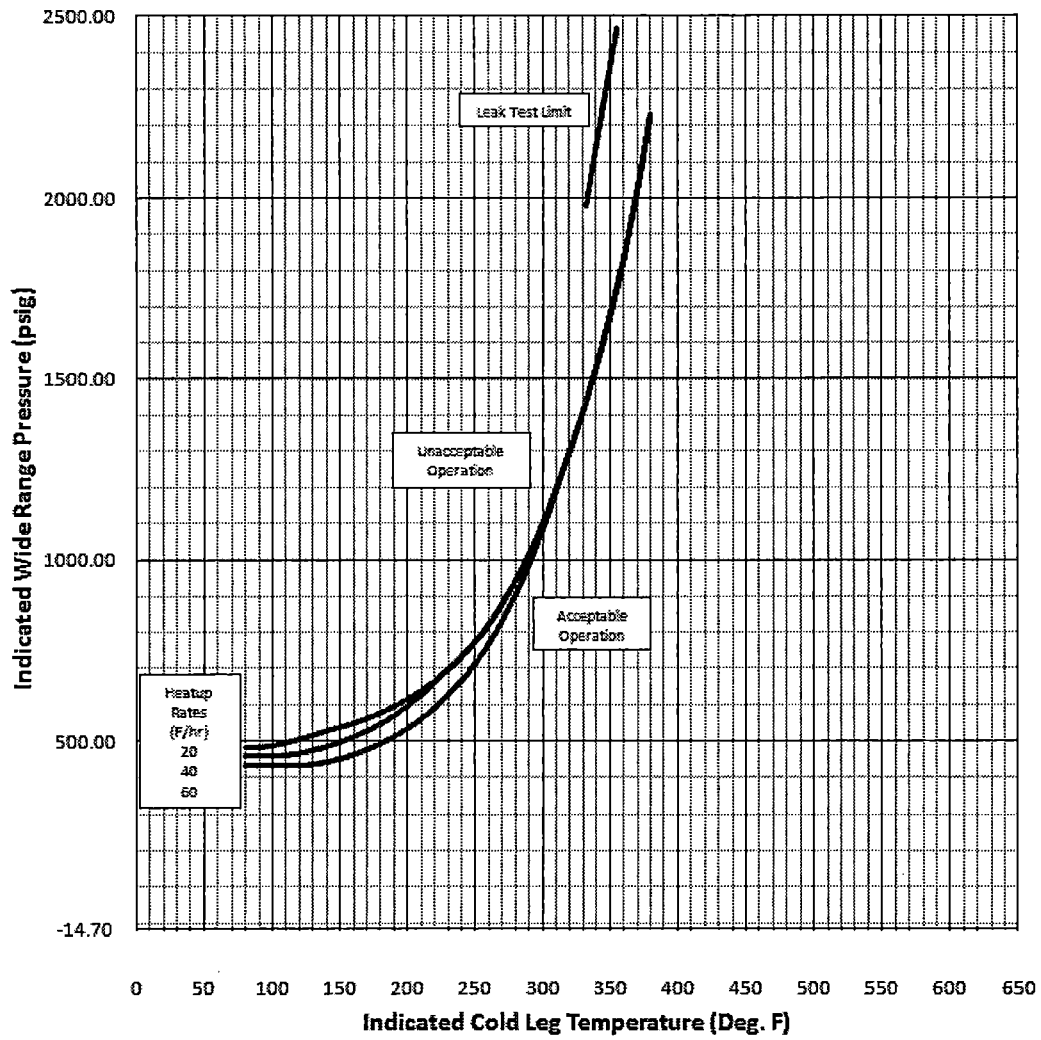


Figure 3.1-1 : Surry Units 1 and 2 Reactor Coolant System Heatup Limitations (Heatup Rates up to 60°F/hr) Applicable for 68 EFPY

Surry Units 1 and 2 Reactor Coolant System Cooldown Limitations

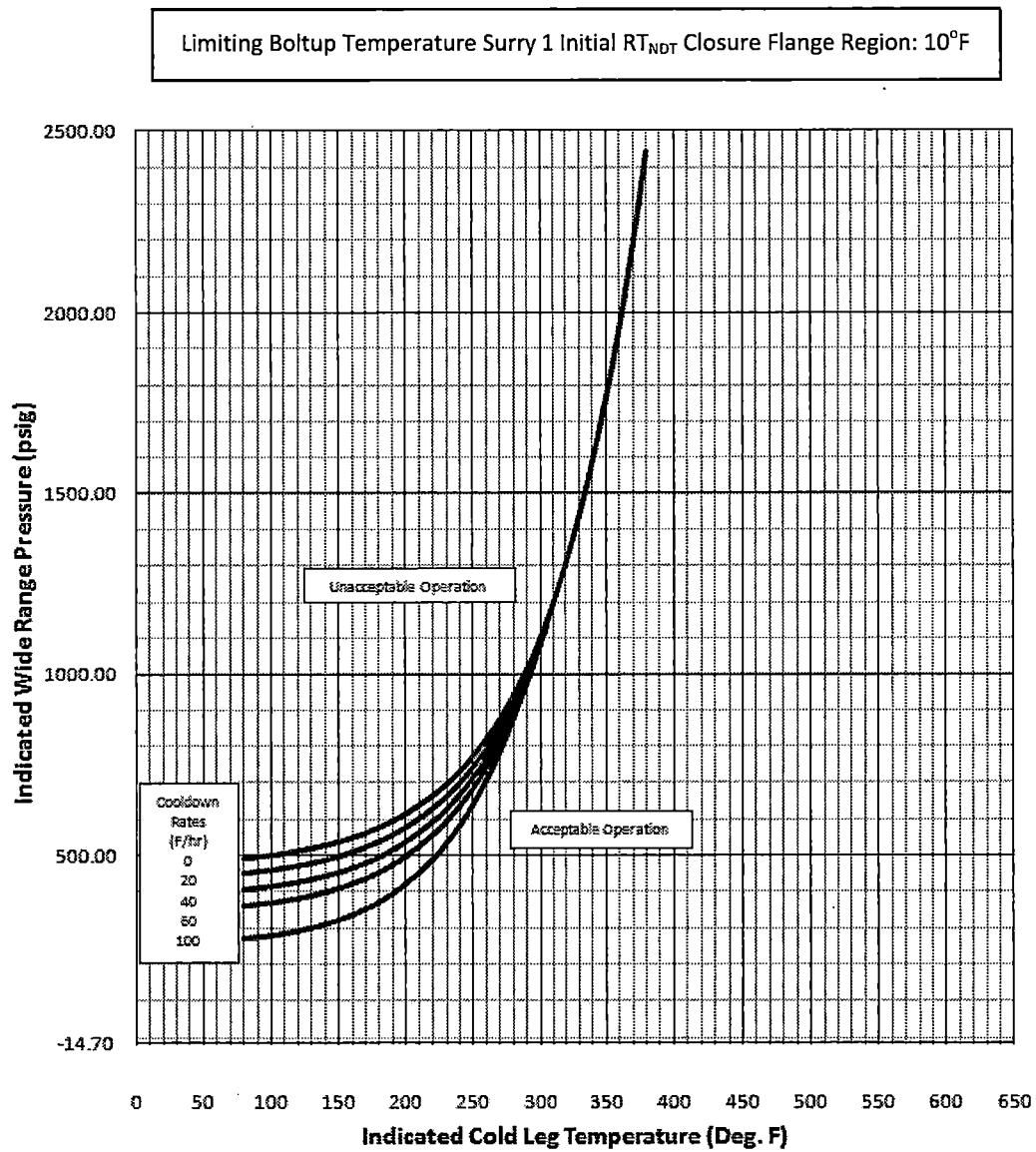


Figure 3.1-2 : Surry Units 1 and 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for 68 EFPY