VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

May 18, 2020 10 CFR 50 Appendix H

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

Serial No.: 20-182
NRA/GDM: R1
Docket Nos.: 50-338/339
License Nos.: NPF-4/7

VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION ENERGY VIRGINIA)
NORTH ANNA POWER STATION UNITS 1 AND 2
REVISED REACTOR VESSEL MATERIALS SURVEILLANCE CAPSULE WITHDRAWAL SCHEDULES
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

By letter dated November 25, 2019, Dominion Energy Virginia submitted a request to revise the surveillance capsule withdrawal schedules for North Anna Power Station Units 1 and 2. The request would, in part, revise projected fluence values for the standby capsules beyond end of life and update the fluence values for capsules removed to date. The NRC staff has reviewed the request and determined that additional information is required to complete its review regarding the provided fluence values. The Dominion Energy Virginia response to the NRC request is provided in Attachment 1. Attachments 2 through 4 provide supporting documents used in determining the revised and updated fluence values.

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

Sincerely,

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

Commitment made in this letter: None

Attachments:

1. Response to NRC Request for Additional Information
2. WCAP-18015-NP, “Extended Beltline Pressure Vessel Fluence Evaluations Applicable to North Anna 1 & 2”
3. WCAP-14040-NP-A, Revision 4, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves”
4. WCAP-18363-NP, Revision 1, “North Anna Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation”

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Attachment 1

Response to NRC Request for Additional Information

Virginia Electric and Power Company
(Dominion Energy Virginia)
North Anna Power Station Units 1 and 2

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
REACTOR VESSEL MATERIAL SURVEILLANCE CAPSULE WITHDRAWAL SCHEDULE

NORTH ANNA POWER STATION UNITS 1 AND 2

NRC Comment:

By letter dated November 25, 2019, Dominion Energy Virginia, the licensee for the North Anna Power Station, Units 1 and 2 (NAPS 1 and 2), submitted a request to revise the surveillance capsule withdrawal schedules for both units. Among other things, the request would revise projected fluence values for the standby capsules beyond end of life and update the fluence values for capsules removed to date. The NRC staff has reviewed the request and determined that additional information is required to complete its review, regarding the fluence values.

Regulatory Basis

The NRC staff review of the revised fluence projections was performed in consideration of the requirements contained in the General Design Criteria (GDCs) located in Appendix A, "General Design Criteria for Nuclear Power Plants," of Part 50, "Domestic Licensing of Production and Utilization Facilities," to Title 10, "Energy" of the US Code of Federal Regulations (10 CFR 50). Specifically, GDCs 14, "Reactor Coolant Pressure Boundary," 30, "Quality of Reactor Coolant Pressure Boundary," and 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," apply. These GDC require the design, fabrication, and maintenance of the reactor coolant pressure boundary with adequate margin to assure that the probability of rapidly propagating failure of the boundary is minimized. In particular, GDC 31 explicitly requires consideration of the effects of irradiation on material properties.

NRC Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," provides guidance on methods for determining reactor pressure vessel fluence that are acceptable to the NRC staff, based on the requirements identified above (ML010890301).

Request

The request stated that revised fluence values were generated and documented in WCAP-18105-NP, Revision 2, "Extended Beltline Pressure Vessel Fluence Evaluations Applicable to NAPS 1 & 2." Please provide this document, as well as any additional detail required to evaluate the acceptability of the fluence estimates in accordance with RG 1.190. If the fluence evaluations do not adhere to RG 1.190, provide additional justification that the fluence calculations address the requirement to consider the effects of irradiation on vessel material properties, consistent with the GDCs identified above.

Dominion Energy Response

In response to the NRC request for additional information, a copy of WCAP-18015-NP, Revision 2, "Extended Beltline Pressure Vessel Fluence Evaluations Applicable to North Anna 1 & 2," is provided in Attachment 2. The fluence calculations outlined in WCAP-18015-NP meet the requirements outlined in Regulatory Guide (RG) 1.190. The methodology used to perform the fluence calculations in WCAP-18015-NP is outlined in WCAP-14040-NP-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," which was approved by the NRC on February 27, 2004 [ADAMS Accession No. ML050120209]. The methodology outlined in WCAP-14040-NP-A, Revision 4, fully meets RG 1.190. A copy of WCAP-14040-NP-A, Revision 4, is provided in Attachment 3 for your reference. The credibility analysis of the NAPS 1 and 2 surveillance capsule program is documented in Appendix E of WCAP-18363-NP, Revision 1, "North Anna Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation." The credibility analysis uses updated fluence values. A copy of WCAP-18363-NP is provided in Attachment 4.

Attachment 2

<u>WCAP-18015-NP, REVISION 2, "EXTENDED BELTLINE PRESSURE VESSEL FLUENCE EVALUATIONS APPLICABLE TO NORTH ANNA 1 & 2"</u>

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

WCAP-18015-NP
Revision 2

September 2018

Extended Beltline Pressure Vessel Fluence Evaluations Applicable to North Anna 1 & 2

Westinghouse

WCAP-18015-NP
Revision 1

Extended Beltline Pressure Vessel Fluence Evaluations Applicable to North Anna 1 & 2

Jared L. Geer*
Radiation Engineering & Analysis

September 2018

Reviewer: Benjamin W. Amiri*
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*Electronically approved records are authenticated in the electronic document management system.

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Record of Revisions

Rev.	Revision Description	Completed
0	Original Issue	November 2015
1	In support of the North Anna Time Limited Aging Analysis (TLAA) on reactor vessel integrity for subsequent License Renewal project, fluence results at the surveillance capsules have been extracted, as well as additional vessel fluence projections. Additional formatting changes have been made. CAP IR-2018-8334 identified an error in the power level used for Unit 2 in several of the cycles in Revision 0 of this document; including the projection cycle. This error has been corrected in Revision 1. See Section 1.1 for more information.	May 2018
2	CAP IR-2018-15270 identified an inconsistency between Table 2-2 and Table 2-6 for the fluence reported at the pressure vessel cladding / base metal interface at the end-of-cycle 24. Table 2-6 has been updated to the correct value. No other changes have been made. Changes are indicated by change bars in the left margin.	September 2018

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1 INTRODUCTION

In the assessment of the state of embrittlement of light water reactor (LWR) pressure vessels, an accurate evaluation of the neutron exposure of each of the materials comprising the beltline region of the vessel is required. In Section II F of 10 CFR 50 Appendix G [Ref. 2], the beltline region is defined as:

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

In Section III A of 10 CFR 50 Appendix H [Ref. 2], the lower limit of neutron exposure for consideration of radiation induced material damage is specified by a neutron fluence (E > 1.0 MeV) threshold of 1.0E+17 n/cm^2. Each of the materials that is anticipated to experience a neutron exposure that exceeds this fluence threshold must be considered in the overall embrittlement assessments for the pressure vessel.

The existing fluence analysis of the North Anna Units 1 and 2 pressure vessels [Refs. 3 and 4] was limited to an axial range that extended approximately 1.5 foot above and 1 foot below the active fuel stack. This model did not include all the pressure vessel materials that could potentially exceed the 1.0E+17 n/cm^2 (E > 1.0 MeV) fluence threshold defined in 10 CFR 50 Appendix H [Ref. 2]. The purpose of this extended beltline fluence evaluation is to define which materials in the North Anna Units 1 and 2 pressure vessels are projected to exceed the 1.0E+17 n/cm^2 threshold neutron fluence before the End of License Extension (EOLE); and, to project the neutron fluence for each of these specific materials. This will help Dominion to fulfill its commitment with respect to the USNRC Request for Additional Information (RAI) Docket Nos.: 50-338/339 [Ref. 8] in determining whether the neutron fluence exposure (E > 1.0 MeV) of the inlet and outlet nozzle materials would be greater than 1.0E+17 n/cm^2.

In subsequent sections of this report, the methodologies used to perform neutron transport calculations are described in some detail and the results of the plant-specific transport calculations are given for each of the materials located in the traditional and extended beltline regions of the North Anna Units 1 and 2 pressure vessels.

1.1 ERROR IN WCAP-18015-NP, REV. 0

Corrective Action Program (CAP) Issue number IR-2018-8334 [Ref. 13] has identified an issue with the power level used for Unit 2 Cycles 21-23. Revision 0 of this document used a pre-uprate power level of 2893 MWt for Unit 2 Cycles 21-23; however, a power uprate to 2940 MWt has been authorized at North Anna Unit 2 (ADAMS Accession Number ML092250616) and implemented prior to Cycle 21.

This error has been corrected in Revision 1. The impact of this error was limited—the largest change in Table 2-3 is 1.3%, which occurs at the 72 effective-full-power-years (EFPY) projection for the "Inlet Nozzle Forging to Vessel Shell Welds – Lowest Extent: Nozzle 1". At 54 EFPY the error is bounded by 1.0% which occurs at the "¼ T Flaw in Inlet Nozzle 2".

2 CALCULATED NEUTRON FLUENCE

2.1 INTRODUCTION

A discrete ordinates S_N transport analysis was performed for the North Anna Units 1 and 2 reactors to determine the neutron radiation environment within the extended beltline of the reactor pressure vessel. In this analysis, radiation exposure parameters were established on a plant- and fuel-cycle-specific basis.

All of the calculations described in this report were based on nuclear cross-section data derived from ENDF/B-VI. Furthermore, the neutron transport evaluation methodologies follow the guidance of Regulatory Guide 1.190 [Ref. 5]. Additionally, the methods used to develop the calculated pressure vessel fluence are consistent with the NRC-approved methodology described in WCAP-14040-A, Revision 4 [Ref. 1].

Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" [Ref. 5], describes state-of-the-art calculation and measurement procedures that are acceptable to the USNRC staff for determining pressure vessel fluence. Also included in Regulatory Guide 1.190 is a discussion of the steps required to qualify and validate the methodology used to determine the neutron exposure of the pressure vessel wall. One important step in the validation process is the comparison of plant-specific neutron calculations with available measurements. An evaluation of the dosimetry sensor sets from three surveillance capsules withdrawn from North Anna Unit 1 is provided in Reference 3; the evaluation for the three surveillance capsules withdrawn from North Anna Unit 2 is provided in Reference 4. The dosimetry analyses documented in References 3 and 4 showed that the ±20% (1σ) acceptance criteria specified in Regulatory Guide 1.190 [Ref. 5] is met.

The results of the present extended beltline analysis are consistent with those of References 3 and 4. Therefore, the Regulatory Guide 1.190 acceptance criteria continue to be met. The validated calculations form the basis for providing future projections of the neutron exposure of the reactor pressure vessel. In line with References 3 and 4, projections up to 54 EFPY are provided in Section 2.2.4. Extended projections up to 72 EFPY are also provided. In addition, as per Dominion's request, projections corresponding to 80 years of life are provided based on 18-month cycles with an average outage time of 25 days. This was determined to correspond to 70.7 EFPY for North Anna Unit 1 and to 71.9 for North Anna Unit 2.

2.2 DISCRETE ORDINATES ANALYSIS

2.2.1 Method Discussion

In performing the fast neutron exposure evaluations for the North Anna Units 1 and 2 reactor vessels, a series of fuel-cycle-specific forward transport calculations were carried out using the following three-dimensional flux synthesis technique:

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

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

All of the transport calculations supporting this analysis were carried out using the DORT discrete ordinates code [Ref. 6] and the BUGLE-96 cross-section library [Ref. 7]. The BUGLE-96 library provides a 67-group coupled neutron-gamma ray cross-section data set produced specifically for light-water reactor (LWR) applications. In these analyses, anisotropic scattering was treated with a P_5 Legendre expansion and angular discretization was modeled with an S_{16} order of angular quadrature. Energy- and space-dependent core power distributions, as well as system operating temperatures, were treated on a fuel-cycle-specific basis.

2.2.2 Reactor Geometry

The analyses documented in References 3 and 4 formed the basis for the current extended beltline evaluation. In completing the current analysis, the [r,θ] and [r] models from Reference 3 and 4 were retained as is while the [r,z] model was expanded to encompass all axial elevations that were anticipated to experience a neutron fluence greater than 1.0E+17 n/cm^2. The [r,z] model was expanded by about 4.5 feet in the +Z direction (relative to the core midplane) to encompass these axial elevations.

For the North Anna Units 1 and 2 transport calculations, the [r,θ] model depicted in Figure 2-1 was utilized because the reactor is octant symmetric. This [r,θ] model includes the core, the reactor internals, the thermal shield – including explicit representations of the surveillance capsules at 15°, 25°, 35°, and 45° – the pressure vessel cladding and vessel wall, the insulation external to the pressure vessel, and the water-filled shield tank. The symmetric [r,θ] model was utilized to perform both the surveillance capsule dosimetry evaluations with subsequent comparisons with calculated results [Refs. 3 and 4], and to generate the maximum fluence levels at the pressure vessel wall. In developing this analytical model, nominal design dimensions were employed for the various structural components. Likewise, water temperatures, and hence, coolant densities in the reactor core, bypass and downcomer regions of the

reactor were taken to be representative of full-power operating conditions. The coolant densities were treated on a fuel-cycle-specific basis (see Section 2.2.3). The reactor core itself was treated as a homogeneous mixture of fuel, cladding, water, and miscellaneous core structures such as fuel assembly grids, guide tubes, etc. The geometric mesh description of the [r,θ] reactor model consisted of 156 radial by 83 azimuthal intervals. Mesh sizes were chosen to assure that proper convergence of the inner iterations was achieved on a pointwise basis. The pointwise inner iteration flux convergence criterion utilized in the [r,θ] calculations was set at a value of 0.001.

The [r,z] model used for the North Anna Units 1 and 2 calculations is shown in Figure 2-2. The model extends radially from the centerline of the reactor core out to a location interior to the water filled shield tank and over an axial span from an elevation approximately 1 foot below to 4 feet above the active fuel. As in the case of the [r,θ] models, nominal design dimensions and full-power coolant densities were employed in the calculations. In this case, the homogenous core region was treated as an equivalent cylinder with a volume equal to that of the active core zone. The stainless steel former plates located between the core baffle and core barrel regions were also explicitly included in the model. The [r,z] geometric mesh description of these reactor models consisted of 148 radial by 148 axial intervals. As in the case of the [r,θ] calculations, mesh sizes were chosen to assure that proper convergence of the inner iterations was achieved on a pointwise basis. The pointwise inner iteration flux convergence criterion utilized in the [r,z] calculations was also set at a value of 0.001.

The one-dimensional radial model used in the synthesis procedure consisted of the same 148 radial mesh intervals included in the [r,z] model. Thus, radial synthesis factors could be determined on a meshwise basis throughout the entire geometry.

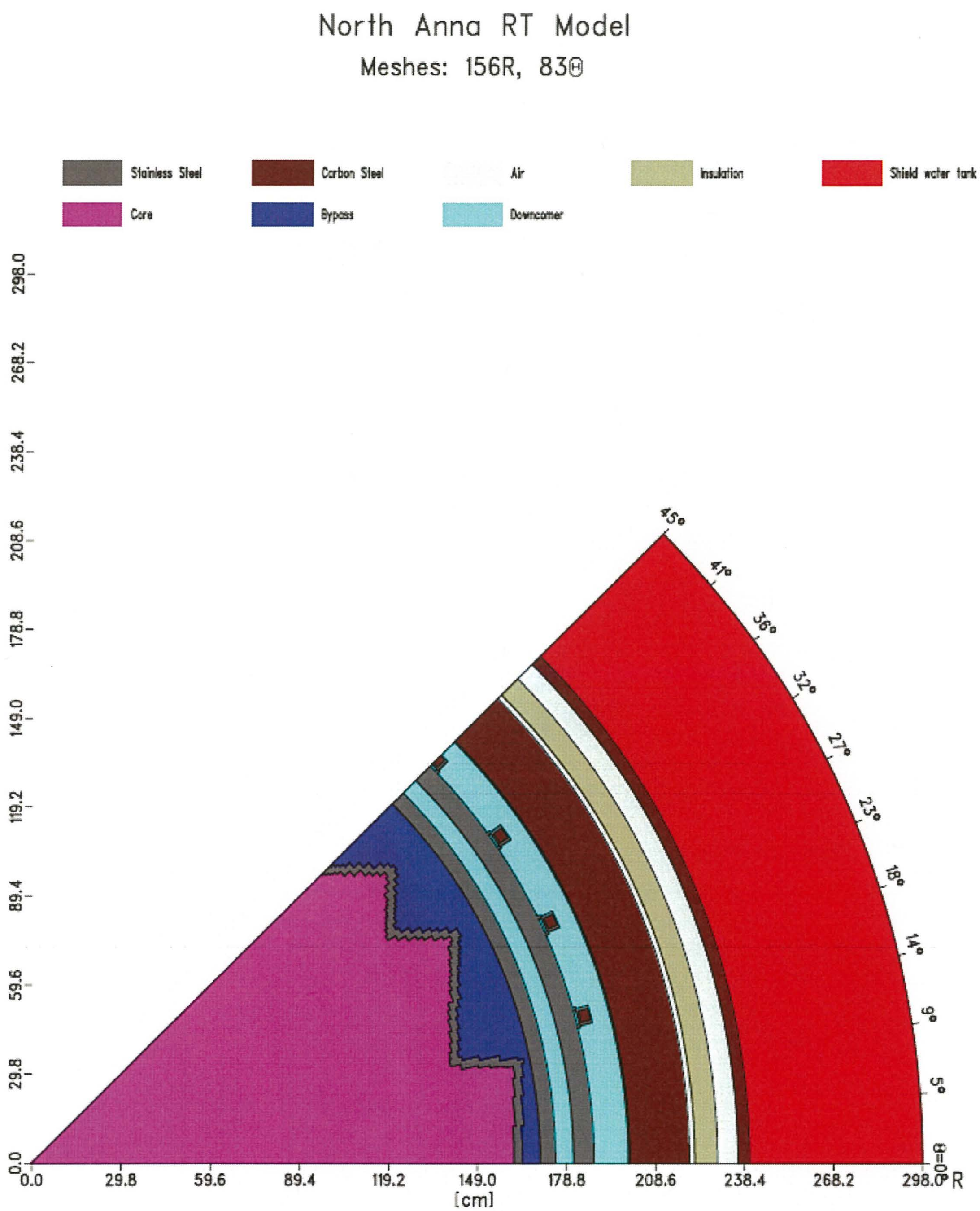

Figure 2-1: North Anna [r,θ] Reactor Geometry at Core Midplane

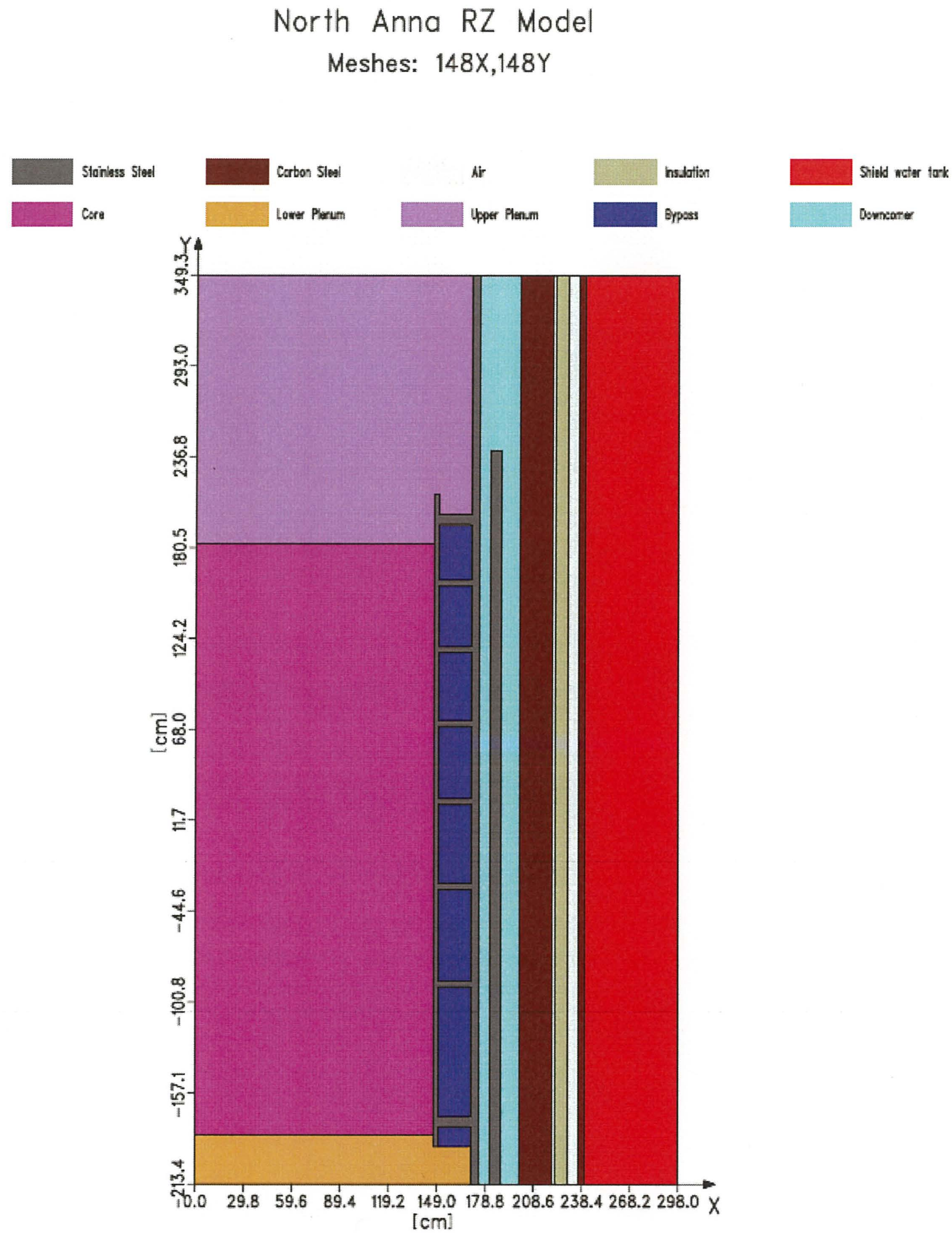

Figure 2-2: North Anna [r,z] Reactor Geometry

2.2.3 Cycle-Specific Information

Because the analyses presented in References 3 and 4 represent the basis for the current evaluation, most of the core design data and operating parameters were taken directly from those analyses. In particular, Reference 3 used Unit 1 cycle-specific core design information for Cycles 1 through 19, whereas Reference 4 used Unit 2 cycle-specific core design information for Cycles 1 through 18. Projections were used beyond that point. Since then, Cycles 20 through 24 were implemented at Unit 1 and Cycles 19 through 23 at Unit 2. The cycle-specific core design data for these latter cycles were taken from Reference 9.

The future projections were based on the assumption that the core power distribution and associated plant operating characteristics from the latest implemented cycle were representative of future plant operation. Therefore, for Unit 1, projections for Cycles 25 and beyond were based on Cycle 24 while for Unit 2, projections for Cycles 24 and beyond were based on Cycle 23.

The data utilized for the core power distributions in plant-specific transport analyses included cycle-dependent fuel assembly initial enrichments, burnups, and axial power distributions. This information was used to develop spatial- and energy-dependent core source distributions averaged over each individual fuel cycle for use in the [r,θ], [r,z], and [r] discrete ordinates transport calculations. Therefore, the results from the neutron transport calculations provided data in terms of fuel cycle-averaged neutron flux, which when multiplied by the appropriate fuel cycle length, generated the incremental fast neutron exposure for each fuel cycle. The cycle length was taken from References 3 and 4 up until Unit 1 Cycle 20 and Unit 2 Cycle 19.

In constructing these core source distributions, the Westinghouse generic approach was used. In this approach, the source term originates from the fission of six nuclides: ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu and ^{242}Pu. Generic values are used including the fission spectra, fission sharing, energy released per fission and average number of neutrons per fission. The relative pin power distributions are taken from the Westinghouse Core Radiation Source Data (CRSD).

Water densities in the core, bypass, and downcomer regions as well as in the upper and lower core plena regions were determined on a fuel cycle-specific basis consistent with the average temperature rise in the 36 fuel assemblies located on the periphery of the reactor core. Because the neutron fluence at the pressure vessel is dominated by leakage from the peripheral fuel assemblies, the use of the peripheral water density in the analytical models is justified. The normal operating condition temperatures were taken from References 3 and 4 up until Unit 1 Cycle 19 and Unit 2 Cycle 18. Beyond that point, the data were taken from Reference 9.

2.2.4 Results

In Table 2-1, locations of the North Anna Units 1 and 2 vessel welds and plates are provided. The axial position of each material is indexed to z = 0.0 cm, which corresponds to the mid-plane of the active fuel stack.

Selected results from the neutron transport analyses are provided in Table 2-2 and Table 2-3 for Units 1 and 2 respectively. Calculated fast neutron (E > 1.0 MeV) fluence for reactor vessel materials, on the pressure vessel clad/base metal interface, is provided for the nominal end of Cycle 24 for Unit 1 (29.7 EFPY) and nominal end of Cycle 23 for Unit 2 (28.1 EFPY). In line with References 3 and 4, projections up to 54 EFPY are provided. Extended projections up to 72 EFPY are also provided. In addition, in Revision 0, as per Dominion's request, projections corresponding to 80 years of life are provided based on 18-month cycles with an average outage time of 25 days. North Anna Unit 1 will reach its 80-years EOLE on April 1, 2058, whereas Unit 2 will reach it on December 14, 2060. Assuming that Unit 1 Cycle 25 started on March 30, 2015 and that Unit 2 Cycle 24 started on October 10, 2014, it was determined that 80-years of life correspond to 70.7 EFPY for Unit 1 and 71.9 EFPY for Unit 2. In Revision 1, Dominion requested that 72 EFPY be used for the 80-years EOLE [Ref. 12].

Figure 2-3 and Figure 2-4 show the relevant weld locations. For the regions beyond the upper circumferential weld, Figure 2-3 shows the axial boundary of the 1.0E+17 n/cm^2 fluence threshold (at 50.3 and 72 EFPY) as a function of azimuthal position (Z versus θ) for Unit 1, whereas Figure 2-4 shows the information (at 52.3 and 72 EFPY) for Unit 2. It is noted that the nozzle materials located above the nozzle centerline remain below 1.0E+17 n/cm^2 through EOLE. Likewise, the lower shell to lower head circumferential weld remains out of the beltline region through EOLE. The data used to generate Figure 2-3 and Figure 2-4 is tabulated in Appendix A and Appendix B, respectively.

The capsule lead factors for the in-vessel surveillance capsules are provided in Table 2-4 and Table 2-5 for Units 1 and 2, respectively. The fast neutron (E > 1.0 MeV) fluence at the clad/base-metal interface and for each position corresponding to the surveillance capsules is provided in Table 2-6 and Table 2-7 for North Anna Units 1 and 2, respectively.

Table 2-1 North Anna 1 & 2 - Pressure Vessel Material Locations

Material	Axial Location* [cm]	Azimuthal Location [degrees]
1/4 T Flaw in Outlet Nozzle		
Nozzle 1	276.54	25
Nozzle 2	276.54	145
Nozzle 3	276.54	265
1/4 T Flaw in Inlet Nozzle		
Nozzle 1	268.31	95
Nozzle 2	268.31	215
Nozzle 3	268.31	335
Outlet Nozzle Forging to Vessel Shell Welds – Lowest Extent		
Nozzle 1	264.82	25
Nozzle 2	264.82	145
Nozzle 3	264.82	265
Inlet Nozzle Forging to Vessel Shell Welds – Lowest Extent		
Nozzle 1	254.52	95
Nozzle 2	254.52	215
Nozzle 3	254.52	335
Nozzle Shell to Intermediate Shell Circumferential Weld	217.42 to 219.42	0 to 360
Intermediate Shell	-42.78 to 217.42	0 to 360
Intermediate Shell to Lower Shell Circumferential Weld	-44.78 to -42.78	0 to 360
Lower Shell	-307.78 to -44.78	0 to 360
Lower Shell to Lower Vessel Head Circumferential Weld	-307.78	0 to 360

* Axial elevations are indexed to Z = 0.0 at the midplane of the active fuel stack.

Table 2-2 North Anna Unit 1 - Maximum Fast Neutron (E > 1.0 MeV) Fluence Experienced by Pressure Vessel Materials in the Extended Beltline

Material	Neutron Fluence [n/cm^2]				
	29.7 EFPY	**50.3 EFPY**	**54 EFPY**	**70.7 EFPY**	**72 EFPY[(a)]**
1/4 T Flaw in Outlet Nozzle					
Nozzle 1	1.35E+16	2.33E+16	2.50E+16	3.29E+16	3.35E+16
Nozzle 2	9.74E+15	1.72E+16	1.86E+16	2.46E+16	2.51E+16
Nozzle 3	3.62E+16	6.12E+16	6.57E+16	8.60E+16	8.75E+16
1/4 T Flaw in Inlet Nozzle					
Nozzle 1[(b)]	6.13E+16	1.04E+17	1.11E+17	1.46E+17	1.48E+17
Nozzle 2	1.65E+16	2.92E+16	3.15E+16	4.17E+16	4.25E+16
Nozzle 3	2.29E+16	3.94E+16	4.24E+16	5.57E+16	5.68E+16
Outlet Nozzle Forging to Vessel Shell Welds – Lowest Extent					
Nozzle 1	2.82E+16	4.84E+16	5.20E+16	6.84E+16	6.97E+16
Nozzle 2	2.03E+16	3.59E+16	3.87E+16	5.12E+16	5.22E+16
Nozzle 3[(c)]	7.53E+16	1.27E+17	1.37E+17	1.79E+17	1.82E+17
Inlet Nozzle Forging to Vessel Shell Welds – Lowest Extent					
Nozzle 1[(d)]	1.30E+17	2.19E+17	2.35E+17	3.07E+17	3.13E+17
Nozzle 2	3.50E+16	6.17E+16	6.65E+16	8.81E+16	8.98E+16
Nozzle 3[(e)]	4.85E+16	8.33E+16	8.95E+16	1.18E+17	1.20E+17
Nozzle Shell	1.30E+18	2.15E+18	2.30E+18	2.99E+18	3.04E+18
Nozzle Shell to Intermediate Shell Circumferential Weld	1.50E+18	2.48E+18	2.66E+18	3.45E+18	3.51E+18
Intermediate shell	3.11E+19	5.03E+19	5.39E+19	6.95E+19	7.07E+19
Intermediate Shell to Lower Shell Circumferential Weld	3.09E+19	5.02E+19	5.36E+19	6.92E+19	7.04E+19
Lower Shell	3.16E+19	5.13E+19	5.48E+19	7.07E+19	7.20E+19
Lower Shell to Lower Vessel Head Circumferential Weld	< 1E+17	< 1E+17	< 1E+17	< 1E+17	< 1E+17

(a) Corresponds to 80 years of life
(b) 1/4 T Flaw in Inlet Nozzle Inlet 1 is projected to reach 1.0E+17 n/cm^2 at approximately 48.5 EFPY; which corresponds to December 26, 2034[(f)].
(c) Outlet Nozzle 3 is projected to reach 1.0E+17 n/cm^2 at approximately 39.5 EFPY; which corresponds to June 6, 2025[(f)].
(d) Inlet Nozzle 1 reached 1.0E+17 n/cm^2 at approximately 22.4 EFPY, which occurred during Cycle 19.
(e) Inlet Nozzle 3 is projected to reach 1.0E+17 n/cm^2 at approximately 60.3 EFPY; which corresponds to May 1, 2047[(f)].
(f) Note these dates are crude approximations based on an 18 month cycle an average outage time of 25 days.

Table 2-3 North Anna Unit 2 - Maximum Fast Neutron (E > 1.0 MeV) Fluence Experienced by Pressure Vessel Materials in the Extended Beltline

Material	Neutron Fluence [n/cm²]				
	28.1 EFPY	**52.3 EFPY**	**54 EFPY**	**71.9 EFPY**	**72 EFPY[(a)]**
1/4 T Flaw in Outlet Nozzle[(h)]					
Nozzle 1	1.28E+16	2.39E+16	2.47E+16	3.29E+16	3.30E+16
Nozzle 2	9.18E+15	1.68E+16	1.73E+16	2.30E+16	2.30E+16
Nozzle 3	3.36E+16	6.33E+16	6.54E+16	8.73E+16	8.75E+16
1/4 T Flaw in Inlet Nozzle[(g)]					
Nozzle 1[(b)]	5.69E+16	1.07E+17	1.11E+17	1.48E+17	1.48E+17
Nozzle 2	1.56E+16	2.85E+16	2.94E+16	3.89E+16	3.90E+16
Nozzle 3	2.17E+16	4.05E+16	4.19E+16	5.58E+16	5.59E+16
Outlet Nozzle Forging to Vessel Shell Welds – Lowest Extent[(g)]					
Nozzle 1	2.67E+16	4.98E+16	5.14E+16	6.86E+16	6.87E+16
Nozzle 2	1.91E+16	3.50E+16	3.61E+16	4.79E+16	4.79E+16
Nozzle 3[(c)]	6.99E+16	1.32E+17	1.36E+17	1.82E+17	1.82E+17
Inlet Nozzle Forging to Vessel Shell Welds – Lowest Extent[(g)]					
Nozzle 1[(d)]	1.21E+17	2.27E+17	2.35E+17	3.13E+17	3.14E+17
Nozzle 2	3.29E+16	6.03E+16	6.22E+16	8.24E+16	8.26E+16
Nozzle 3[(e)]	4.60E+16	8.58E+16	8.86E+16	1.18E+17	1.18E+17
Nozzle Shell[(g)]	1.20E+18	2.23E+18	2.30E+18	3.07E+18	3.07E+18
Nozzle Shell to Intermediate Shell Circumferential Weld[(g)]	1.38E+18	2.58E+18	2.66E+18	3.55E+18	3.55E+18
Intermediate Shell[(g)]	2.87E+19	5.25E+19	5.42E+19	7.19E+19	7.20E+19
Intermediate Shell to Lower Shell Circumferential Weld[(g)]	2.86E+19	5.24E+19	5.41E+19	7.17E+19	7.18E+19
Lower Shell[(g)]	2.92E+19	5.36E+19	5.53E+19	7.33E+19	7.34E+19
Lower Shell to Lower Vessel Head Circumferential Weld	< 1.00E+17	< 1.00E+17	< 1.00E+17	< 1.00E+17	< 1.00E+17

(a) Corresponds to 80 years of life

(b) 1/4 T Flaw in Inlet Nozzle Inlet 1 is projected to reach 1.0E+17 n/cm² at approximately 48.8 EFPY; which corresponds to May 27, 2036[(f, h)].

(c) Outlet Nozzle 3 is projected to reach 1.0E+17 n/cm² at approximately 39.8 EFPY[(h)]; which corresponds to February 4, 2027[(f)].

(d) Inlet Nozzle 1 reached 1.0E+17 n/cm² at approximately 23.1 EFPY[(h)], which occurred during Cycle 20.

(e) Inlet Nozzle 3 is projected to reach 1.0E+17 n/cm² at approximately 60.9 EFPY[(h)]; which corresponds to February 12, 2049[(f)].

(f) Note these dates are crude approximations based on an 18 month cycle an average outage time of 25 days.

(g) Several values have changed in Rev. 1 due to the use of the pre-uprate power for the projection cycles in Rev. 0.

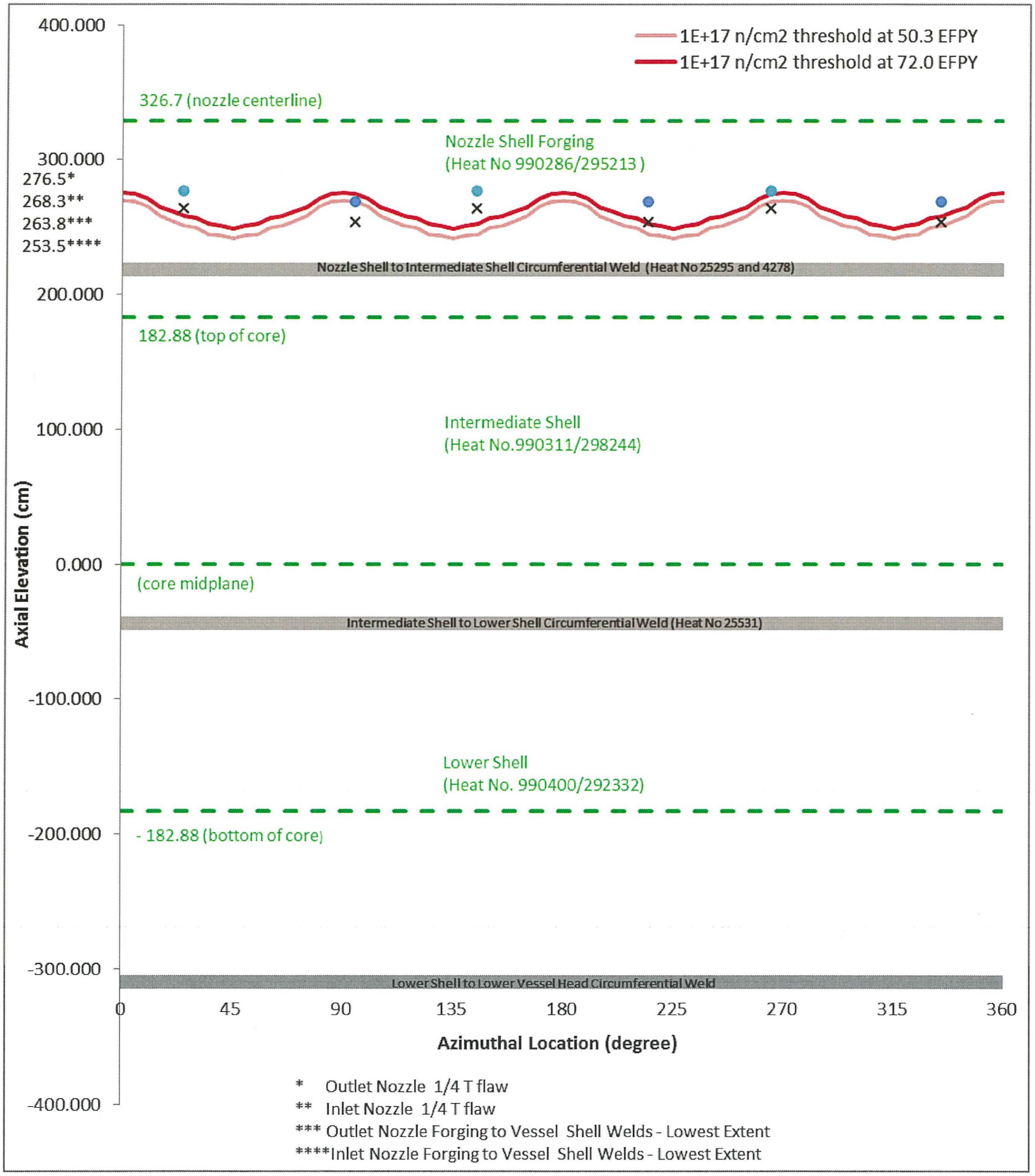

Figure 2-3 North Anna Unit 1 - Axial Boundary of the 1.0E+17 n/cm^2 Fluence Threshold in the +Z Direction (at 50.3 and 72 EFPY)

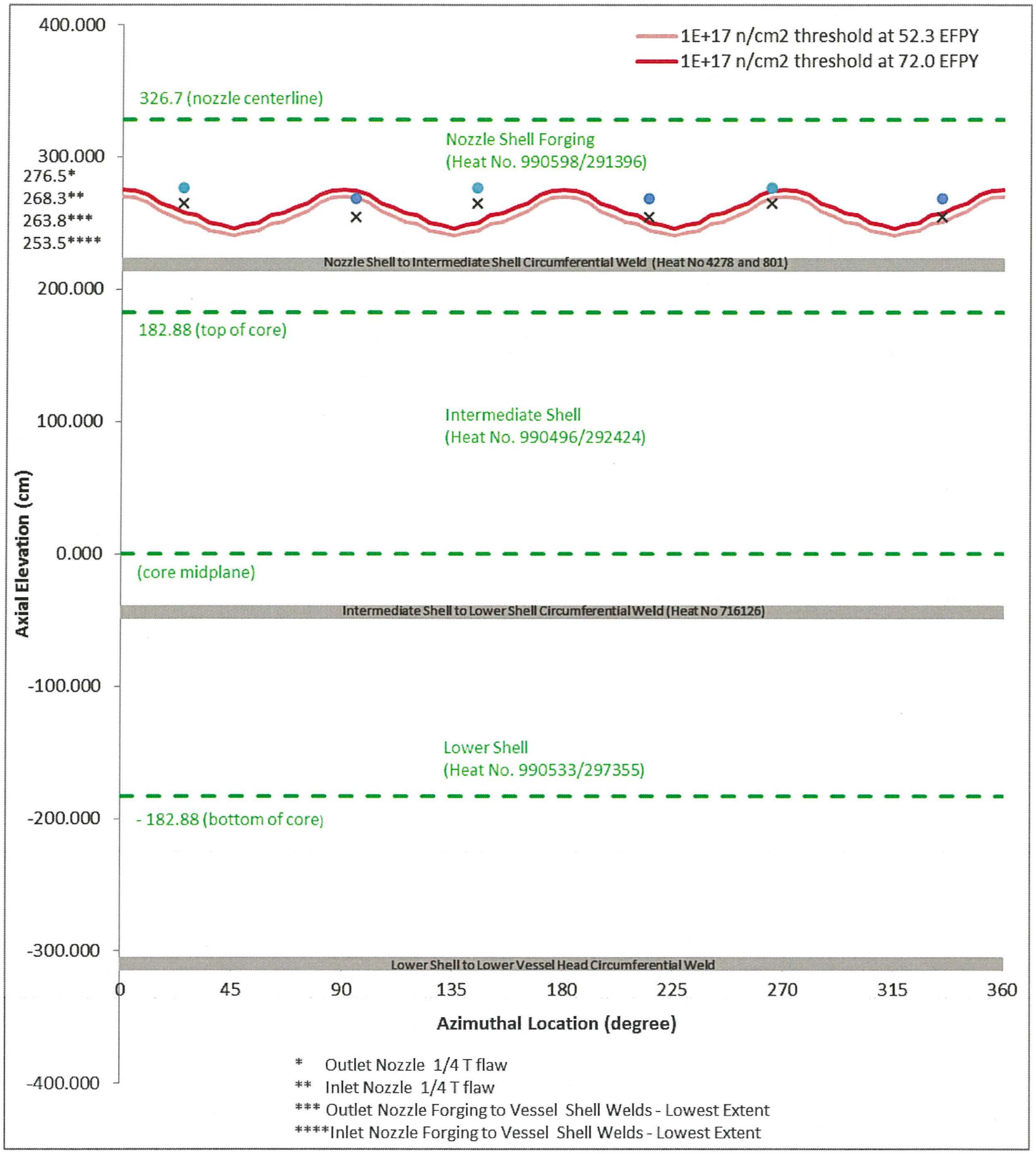

Figure 2-4 North Anna Unit 2 - Axial Boundary of the 1.0E+17 n/cm^2 Fluence Threshold in the +Z Direction (at 52.3 and 72 EFPY)

Table 2-4 Surveillance Capsule Lead Factors for North Anna Unit 1

Cycle	Cumulative Time [EFPY]	Capsules (position in first octant) V (15°)	U (25°)	W (25°)	Z (35°/15°)*	T (35°/25°)*	Y (25°)	S (45°)	X (15°)
1	1.1	1.61	1.05	1.05	0.72	0.72	1.05	0.56	1.61
2	1.9	---	1.04	1.04	0.70	0.70	1.04	0.54	1.61
3	2.9	---	1.05	1.05	0.70	0.70	1.05	0.54	1.60
4	3.8	---	1.01	1.01	0.67	0.67	1.01	0.51	1.59
5	4.8	---	1.02	1.02	0.68	0.68	1.02	0.53	1.59
6	5.9	---	1.04	1.04	0.69	0.69	1.04	0.54	1.61
7	7.1	---	---	1.07	0.72	0.72	1.07	0.56	1.63
8	8.4	---	---	1.10	0.74	0.74	1.10	0.57	1.65
9	9.8	---	---	1.11	0.75	0.75	1.11	0.58	1.66
10	11.1	---	---	1.13	0.76	0.76	1.13	0.60	1.68
11	12.4	---	---	1.14	0.78	0.78	1.14	0.61	1.68
12	13.5	---	---	1.15	0.78	0.78	1.15	0.61	1.69
13	14.8	---	---	1.16	0.79	0.79	1.16	0.62	1.70
14	16.2	---	---	---	0.80	0.80	1.17	0.63	1.71
15	17.5	---	---	---	0.87	0.84	1.19	0.63	1.72
16	18.9	---	---	---	0.93	0.87	1.20	0.64	1.73
17	20.2	---	---	---	0.98	0.90	1.21	0.65	1.74
18	21.6	---	---	---	1.03	0.92	1.22	0.66	1.75
19	23.0	---	---	---	1.07	0.94	1.22	0.66	1.75
20	24.4	---	---	---	1.11	0.96	1.23	0.66	1.76
21	25.8	---	---	---	1.15	0.98	1.24	0.67	1.76
22	26.9	---	---	---	1.17	1.00	1.24	0.68	1.77
23	28.3	---	---	---	1.19	1.00	1.24	0.68	1.76
24	29.7	---	---	---	1.22	1.02	1.24	0.68	1.77
Projected	50.3	---	---	---	1.44	1.13	1.26	0.73	1.78
Projected	51.6	---	---	---	1.44	1.13	1.26	0.73	1.78
Projected	54.0	---	---	---	1.46	1.14	1.27	0.73	1.78
Projected	70.7	---	---	---	1.53	1.17	1.27	0.75	1.78
Projected	72.0	---	---	---	1.54	1.18	1.27	0.75	1.78

* Capsules Z and T were moved from the 35° equivalent positions at end-of-cycle 14 [Ref. 10].

- Capsule Z moved from 305° to 165° (first octant equivalent: 35° to 15°)
- Capsule T moved from 55° to 245° (first octant equivalent: 35° to 25°)

Table 2-5 Surveillance Capsule Lead Factors for North Anna Unit 2

Cycle	Cumulative Time [EFPY]	Capsules (position in first octant) V (15°)	U (25°)	W (25°)	Z* (35°/15°)	T* (35°/25°)	Y (25°)	S (45°)	X (15°)
1	1.0	1.61	1.05	1.05	0.72	0.72	1.05	0.56	1.61
2	1.6	---	1.04	1.04	0.71	0.71	1.04	0.55	1.60
3	2.7	---	1.16	1.16	0.79	0.79	1.16	0.62	1.66
4	3.8	---	1.15	1.15	0.79	0.79	1.15	0.63	1.66
5	5.0	---	1.16	1.16	0.80	0.80	1.16	0.63	1.68
6	6.2	---	1.17	1.17	0.81	0.81	1.17	0.63	1.69
7	7.5	---	---	1.17	0.81	0.81	1.17	0.64	1.70
8	8.7	---	---	1.18	0.82	0.82	1.18	0.64	1.71
9	9.9	---	---	1.19	0.83	0.83	1.19	0.65	1.71
10	11.3	---	---	1.20	0.84	0.84	1.20	0.67	1.72
11	12.5	---	---	1.20	0.84	0.84	1.20	0.67	1.73
12	13.8	---	---	1.20	0.84	0.84	1.20	0.66	1.73
13	15.1	---	---	1.21	0.85	0.85	1.21	0.67	1.74
14	16.5	---	---	---	0.92	0.88	1.22	0.67	1.75
15	17.7	---	---	---	0.97	0.91	1.23	0.68	1.76
16	19.0	---	---	---	1.14	0.93	1.24	0.68	1.77
17	20.3	---	---	---	1.18	0.96	1.25	0.68	1.77
18	21.6	---	---	---	1.22	0.99	1.26	0.68	1.78
19	22.9	---	---	---	1.25	1.01	1.27	0.69	1.79
20	24.3	---	---	---	1.28	1.02	1.27	0.69	1.79
21	25.5	---	---	---	1.31	1.04	1.27	0.69	1.79
22	26.8	---	---	---	1.33	1.05	1.28	0.70	1.80
23	28.1	---	---	---	1.35	1.06	1.28	0.69	1.80
Projected	52.3	---	---	---	1.58	1.15	1.27	0.67	1.82
Projected	54.0	---	---	---	1.58	1.16	1.27	0.67	1.82
Projected	71.9	---	---	---	1.65	1.19	1.27	0.66	1.82
Projected	72.0	---	---	---	1.65	1.19	1.27	0.66	1.82

* Capsules Z and T were moved from the 35° equivalent positions at end-of-cycle 13 [Ref. 10].

- Capsule Z moved from 305° to 165° (first octant equivalent: 35° to 15°)
- Capsule T moved from 55° to 65° (first octant equivalent: 35° to 25°)

Table 2-6 Calculated Neutron Fluence (E > 1.0 MeV) [n/cm^2] at the Surveillance Capsule Center and Maximum at the Pressure Vessel Clad/Base Metal interface for North Anna Unit 1

Cycle	Cumulative Time [EFPY]	Surveillance Capsules 15°	25°	35°	45°	35°/15°(d)	35°/25°(e)	Pressure Vessel Clad/Base Metal Interface
1	1.1	3.06E+18(a)	2.01E+18	1.37E+18	1.07E+18	1.37E+18	1.37E+18	1.90E+18
2	1.9	5.45E+18	3.54E+18	2.38E+18	1.84E+18	2.38E+18	2.38E+18	3.39E+18
3	2.9	7.65E+18	5.01E+18	3.34E+18	2.58E+18	3.34E+18	3.34E+18	4.78E+18
4	3.8	1.00E+19	6.39E+18	4.20E+18	3.24E+18	4.20E+18	4.20E+18	6.31E+18
5	4.8	1.18E+19	7.59E+18	5.03E+18	3.91E+18	5.03E+18	5.03E+18	7.42E+18
6	5.9	1.40E+19	9.14E+18(b)	6.06E+18	4.70E+18	6.06E+18	6.06E+18	8.75E+18
7	7.1	1.64E+19	1.08E+19	7.19E+18	5.58E+18	7.19E+18	7.19E+18	1.01E+19
8	8.4	1.89E+19	1.25E+19	8.41E+18	6.54E+18	8.41E+18	8.41E+18	1.14E+19
9	9.8	2.14E+19	1.42E+19	9.59E+18	7.46E+18	9.59E+18	9.59E+18	1.29E+19
10	11.1	2.37E+19	1.59E+19	1.08E+19	8.40E+18	1.08E+19	1.08E+19	1.41E+19
11	12.4	2.59E+19	1.75E+19	1.19E+19	9.33E+18	1.19E+19	1.19E+19	1.54E+19
12	13.5	2.79E+19	1.90E+19	1.29E+19	1.01E+19	1.29E+19	1.29E+19	1.65E+19
13	14.8	3.02E+19	2.05E+19(c)	1.40E+19	1.09E+19	1.40E+19	1.40E+19	1.77E+19
14	16.2	3.26E+19	2.23E+19	1.52E+19	1.19E+19	1.52E+19	1.52E+19	1.90E+19
15	17.5	3.50E+19	2.41E+19	1.65E+19	1.29E+19	1.76E+19	1.70E+19	2.03E+19
16	18.9	3.74E+19	2.58E+19	1.77E+19	1.38E+19	2.01E+19	1.88E+19	2.16E+19
17	20.2	3.98E+19	2.76E+19	1.89E+19	1.48E+19	2.24E+19	2.05E+19	2.28E+19
18	21.6	4.22E+19	2.94E+19	2.02E+19	1.58E+19	2.49E+19	2.23E+19	2.41E+19
19	23.0	4.47E+19	3.11E+19	2.14E+19	1.68E+19	2.73E+19	2.41E+19	2.55E+19
20	24.4	4.71E+19	3.29E+19	2.27E+19	1.78E+19	2.98E+19	2.58E+19	2.68E+19
21	25.8	4.95E+19	3.47E+19	2.39E+19	1.88E+19	3.21E+19	2.76E+19	2.81E+19
22	26.9	5.13E+19	3.60E+19	2.49E+19	1.96E+19	3.39E+19	2.89E+19	2.90E+19
23	28.3	5.36E+19	3.75E+19	2.60E+19	2.05E+19	3.62E+19	3.04E+19	3.03E+19
24	29.7	5.59E+19	3.92E+19	2.73E+19	2.16E+19	3.86E+19	3.22E+19	3.16E+19
Projected	50.3	9.11E+19	6.48E+19	4.64E+19	3.74E+19	7.37E+19	5.77E+19	5.13E+19
Projected	54.0	9.74E+19	6.94E+19	4.98E+19	4.03E+19	8.00E+19	6.23E+19	5.48E+19
Projected	70.7	1.26E+20	9.01E+19	6.53E+19	5.31E+19	1.08E+20	8.30E+19	7.07E+19
Projected	72.0	1.28E+20	9.17E+19	6.65E+19	5.41E+19	1.11E+20	8.46E+19	7.20E+19

(a) Capsule V was withdrawn at the end-of-cycle 1.
(b) Capsule U was withdrawn at the end-of-cycle 6.
(c) Capsule W was withdrawn at the end-of-cycle 13.
(d) Capsule Z was moved at the end-of-cycle 14.
(e) Capsule T was moved at the end-of-cycle 14.

Table 2-7 Calculated Neutron Fluence (E > 1.0 MeV) [n/cm^2] at the Surveillance Capsule Center and Maximum at the Pressure Vessel Clad/Base Metal interface for North Anna Unit 2

Cycle	Cumulative Time [EFPY]	Surveillance Capsules 15°	25°	35°	45°	35°/15°[(d)]	35°/25°[(e)]	Pressure Vessel Clad/Base Metal Interface
1	1.0	2.86E+18[(a)]	1.87E+18	1.27E+18	9.96E+17	1.27E+18	1.27E+18	1.78E+18
2	1.6	4.68E+18	3.05E+18	2.06E+18	1.61E+18	2.06E+18	2.06E+18	2.92E+18
3	2.7	6.99E+18	4.86E+18	3.34E+18	2.62E+18	3.34E+18	3.34E+18	4.20E+18
4	3.8	9.44E+18	6.53E+18	4.51E+18	3.57E+18	4.51E+18	4.51E+18	5.68E+18
5	5.0	1.19E+19	8.20E+18	5.68E+18	4.48E+18	5.68E+18	5.68E+18	7.07E+18
6	6.2	1.42E+19	9.85E+18[(b)]	6.80E+18	5.35E+18	6.80E+18	6.80E+18	8.43E+18
7	7.5	1.65E+19	1.14E+19	7.88E+18	6.21E+18	7.88E+18	7.88E+18	9.71E+18
8	8.7	1.87E+19	1.30E+19	8.97E+18	7.06E+18	8.97E+18	8.97E+18	1.10E+19
9	9.9	2.09E+19	1.45E+19	1.01E+19	8.00E+18	1.01E+19	1.01E+19	1.22E+19
10	11.3	2.30E+19	1.60E+19	1.12E+19	8.91E+18	1.12E+19	1.12E+19	1.34E+19
11	12.5	2.52E+19	1.75E+19	1.23E+19	9.71E+18	1.23E+19	1.23E+19	1.46E+19
12	13.8	2.75E+19	1.91E+19	1.33E+19	1.06E+19	1.33E+19	1.33E+19	1.59E+19
13	15.1	2.99E+19	2.08E+19[(c)]	1.45E+19	1.15E+19	1.45E+19	1.45E+19	1.71E+19
14	16.5	3.22E+19	2.25E+19	1.57E+19	1.24E+19	1.68E+19	1.62E+19	1.84E+19
15	17.7	3.44E+19	2.41E+19	1.68E+19	1.32E+19	1.91E+19	1.78E+19	1.96E+19
16	19.0	3.65E+19	2.56E+19	1.78E+19	1.41E+19	2.35E+19	1.93E+19	2.07E+19
17	20.3	3.90E+19	2.75E+19	1.91E+19	1.50E+19	2.60E+19	2.12E+19	2.20E+19
18	21.6	4.15E+19	2.93E+19	2.03E+19	1.59E+19	2.85E+19	2.30E+19	2.33E+19
19	22.9	4.37E+19	3.09E+19	2.14E+19	1.68E+19	3.06E+19	2.46E+19	2.44E+19
20	24.3	4.59E+19	3.25E+19	2.26E+19	1.77E+19	3.29E+19	2.62E+19	2.56E+19
21	25.5	4.80E+19	3.41E+19	2.36E+19	1.86E+19	3.50E+19	2.78E+19	2.67E+19
22	26.8	5.03E+19	3.57E+19	2.48E+19	1.95E+19	3.72E+19	2.94E+19	2.80E+19
23	28.1	5.26E+19	3.73E+19	2.59E+19	2.03E+19	3.96E+19	3.10E+19	2.92E+19
Projected	52.3	9.75E+19	6.82E+19	4.63E+19	3.59E+19	8.44E+19	6.19E+19	5.36E+19
Projected	54.0	1.01E+20	7.04E+19	4.78E+19	3.70E+19	8.76E+19	6.41E+19	5.53E+19
Projected	71.9	1.34E+20	9.32E+19	6.29E+19	4.85E+19	1.21E+20	8.69E+19	7.33E+19
Projected	72.0	1.34E+20	9.33E+19	6.30E+19	4.85E+19	1.21E+20	8.70E+19	7.34E+19

(a) Capsule V was withdrawn at the end-of-cycle 1.
(b) Capsule U was withdrawn at the end-of-cycle 6.
(c) Capsule W was withdrawn at the end-of-cycle 13.
(d) Capsule Z was moved at the end-of-cycle 13.
(e) Capsule T was moved at the end-of-cycle 13.

2.2.5 Recommendations

Table 2-2 and Table 2-3 report the maximum fast neutron (E > 1.0 MeV) fluence at specific pressure vessel materials in the extended beltline for North Anna 1 & 2, respectively. The nozzle shell, nozzle shell to intermediate shell circumferential weld, intermediate shell to lower shell circumferential weld, and the lower shell to lower vessel head circumferential weld have a single set of neutron fluence values for each unit. These neutron fluence values would be appropriate for use for P-T limit analyses for these materials. Regarding the inlet and outlet nozzles, two separate fast neutron fluence values are given at two locations for each nozzle. One location represents the lowest extent of the nozzle forging to vessel shell welds, whereas the second location conservatively represents the 1/4 T flaw in the nozzles. Although the fluence results at the nozzle forging to vessel shell welds are more limiting, the 1/4T flaw location is more representative of the fluence for the nozzle at the peak stress location. Therefore, fluence at either location (lowest extent of the nozzle forging to vessel shell welds or at the 1/4T flaw) can be used for the P-T limit analyses for the inlet and outlet nozzles for each unit.

A full three-dimensional discrete ordinates model provides a more geometrically detailed analysis. This more detailed representation can be utilized as a next step for further analysis of the maximum fast neutron fluence analyses for each unit if needed.

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APPENDIX A
DATA USED IN THE GENERATION OF FIGURE 2-3

The table below contains the data used to plot Figure 2-3. All values in this table are Z elevations (in centimeters) indexed by the angle (in degrees).

- "Weld C1" is the Nozzle Shell to Intermediate Shell Circumferential Weld.
- "Weld C2" is the Intermediate Shell to Lower Shell Circumferential Weld.
- "Weld C3" is the Lower Shell to Lower vessel Head Circumferential Weld.
- "Outlet" is the Outlet Nozzle Forging to Vessel Shell Welds-Lowest Extent.
- "Inlet" is the Inlet Nozzle Forging to Vessel Shell Welds-Lowest Extent.
- "1/4 Outlet" is the Outlet Nozzle ¼ T Flaw.
- "1/4 Inlet" is the Inlet Nozzle ¼ T Flaw.

The threshold elevations were obtained by linearly interpolating the synthesis output.

	1E17 n/cm^2 Fluence Threshold, Z (cm)		Top Of Core	Core Midplane	Bottom of Core	Nozzle Centerline	Weld C1	Weld C2	Weld C3			1/4 Outlet	1/4 Inlet
Angle	50.3 EFPY	72 EFPY	(cm)	(cm)	(cm)	(cm)	(cm)	(cm)	(cm)	Outlet (cm)	Inlet (cm)	(cm)	(cm)
0	269.846	275.433	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
5	268.911	274.519	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
10	265.397	271.309	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
15	258.074	264.708	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
20	254.690	261.741	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
25	250.669	258.155	182.88	0	-182.88	328.524	218.42	-43.78	-309.78	263.82		276.54	
30	249.263	256.901	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
35	244.754	252.254	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				

Angle	1E17 n/cm^2 Fluence Threshold, Z (cm) 50.3 EFPY	72 EFPY	Top Of Core (cm)	Core Midplane (cm)	Bottom of Core (cm)	Nozzle Centerline (cm)	Weld C1 (cm)	Weld C2 (cm)	Weld C3 (cm)	Outlet (cm)	Inlet (cm)	1/4 Outlet (cm)	1/4 Inlet (cm)
40	243.608	251.077	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
45	241.725	248.669	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
50	243.608	251.077	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
55	244.754	252.254	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
60	249.263	256.901	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
65	250.669	258.155	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
70	254.690	261.741	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
75	258.074	264.708	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
80	265.397	271.309	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
85	268.911	274.519	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
90	269.846	275.433	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
95	268.911	274.519	182.88	0	-182.88	328.524	218.42	-43.78	-309.78		253.52		268.31
100	265.397	271.309	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
105	258.074	264.708	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
110	254.690	261.741	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
115	250.669	258.155	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
120	249.263	256.901	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
125	244.754	252.254	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
130	243.608	251.077	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
135	241.725	248.669	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
140	243.608	251.077	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
145	244.754	252.254	182.88	0	-182.88	328.524	218.42	-43.78	-309.78	263.82		276.54	
150	249.263	256.901	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
155	250.669	258.155	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
160	254.690	261.741	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
165	258.074	264.708	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				

Angle	1E17 n/cm^2 Fluence Threshold, Z (cm) 50.3 EFPY	72 EFPY	Top Of Core (cm)	Core Midplane (cm)	Bottom of Core (cm)	Nozzle Centerline (cm)	Weld C1 (cm)	Weld C2 (cm)	Weld C3 (cm)	Outlet (cm)	Inlet (cm)	1/4 Outlet (cm)	1/4 Inlet (cm)
170	265.397	271.309	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
175	268.911	274.519	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
180	269.846	275.433	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
185	268.911	274.519	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
190	265.397	271.309	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
195	258.074	264.708	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
200	254.690	261.741	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
205	250.669	258.155	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
210	249.263	256.901	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
215	244.754	252.254	182.88	0	-182.88	328.524	218.42	-43.78	-309.78		253.52		268.31
220	243.608	251.077	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
225	241.725	248.669	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
230	243.608	251.077	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
235	244.754	252.254	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
240	249.263	256.901	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
245	250.669	258.155	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
250	254.690	261.741	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
255	258.074	264.708	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
260	265.397	271.309	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
265	268.911	274.519	182.88	0	-182.88	328.524	218.42	-43.78	-309.78	263.82		276.54	
270	269.846	275.433	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
275	268.911	274.519	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
280	265.397	271.309	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
285	258.074	264.708	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
290	254.690	261.741	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
295	250.669	258.155	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				

Angle	1E17 n/cm^2 Fluence Threshold, Z (cm) 50.3 EFPY	72 EFPY	Top Of Core (cm)	Core Midplane (cm)	Bottom of Core (cm)	Nozzle Centerline (cm)	Weld C1 (cm)	Weld C2 (cm)	Weld C3 (cm)	Outlet (cm)	Inlet (cm)	1/4 Outlet (cm)	1/4 Inlet (cm)
300	249.263	256.901	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
305	244.754	252.254	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
310	243.608	251.077	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
315	241.725	248.669	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
320	243.608	251.077	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
325	244.754	252.254	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
330	249.263	256.901	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
335	250.669	258.155	182.88	0	-182.88	328.524	218.42	-43.78	-309.78		253.52		268.31
340	254.690	261.741	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
345	258.074	264.708	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
350	265.397	271.309	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
355	268.911	274.519	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
360	269.846	275.433	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				

APPENDIX B
DATA USED IN THE GENERATION OF FIGURE 2-4

The table below contains the data used to plot Figure 2-4. All values in this table are Z elevations (in centimeters) indexed by the angle (in degrees).

- "Weld C1" is the Nozzle Shell to Intermediate Shell Circumferential Weld.
- "Weld C2" is the Intermediate Shell to Lower Shell Circumferential Weld.
- "Weld C3" is the Lower Shell to Lower vessel Head Circumferential Weld.
- "Outlet" is the Outlet Nozzle Forging to Vessel Shell Welds-Lowest Extent.
- "Inlet" is the Inlet Nozzle Forging to Vessel Shell Welds-Lowest Extent.
- "1/4 Outlet" is the Outlet Nozzle ¼ T Flaw.
- "1/4 Inlet" is the Inlet Nozzle ¼ T Flaw.

The threshold elevations were obtained by linearly interpolating the synthesis output.

	1E17 n/cm^2 Fluence Threshold, Z (cm)		Top Of Core	Core Midplane	Bottom of Core	Nozzle Centerline	Weld C1	Weld C2	Weld C3	Outlet	Inlet	1/4 Outlet	1/4 Inlet
Angle	52.3 EFPY	72 EFPY	(cm)	(cm)	(cm)	(cm)	(cm)	(cm)	(cm)	(cm)	(cm)	(cm)	(cm)
0	270.275	275.296	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
5	269.445	273.781	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
10	266.139	270.894	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
15	259.023	264.348	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
20	255.602	261.420	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
25	251.325	257.172	182.88	0	-182.88	328.524	218.42	-43.78	-309.78	263.82		276.54	
30	249.407	254.921	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
35	244.397	249.290	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
40	242.808	246.754	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
45	240.778	245.467	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				

Angle	1E17 n/cm^2 Fluence Threshold, Z (cm) 52.3 EFPY	72 EFPY	Top Of Core (cm)	Core Midplane (cm)	Bottom of Core (cm)	Nozzle Centerline (cm)	Weld C1 (cm)	Weld C2 (cm)	Weld C3 (cm)	Outlet (cm)	Inlet (cm)	1/4 Outlet (cm)	1/4 Inlet (cm)
50	242.808	246.754	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
55	244.397	249.290	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
60	249.407	254.921	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
65	251.325	257.172	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
70	255.602	261.420	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
75	259.023	264.348	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
80	266.139	270.894	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
85	269.445	273.781	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
90	270.275	275.296	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
95	269.445	273.781	182.88	0	-182.88	328.524	218.42	-43.78	-309.78		253.52		268.31
100	266.139	270.894	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
105	259.023	264.348	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
110	255.602	261.420	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
115	251.325	257.172	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
120	249.407	254.921	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
125	244.397	249.290	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
130	242.808	246.754	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
135	240.778	245.467	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
140	242.808	246.754	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
145	244.397	249.290	182.88	0	-182.88	328.524	218.42	-43.78	-309.78	263.82		276.54	
150	249.407	254.921	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
155	251.325	257.172	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
160	255.602	261.420	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
165	259.023	264.348	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
170	266.139	270.894	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
175	269.445	273.781	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				

Angle	1E17 n/cm² Fluence Threshold, Z (cm) 52.3 EFPY	72 EFPY	Top Of Core (cm)	Core Midplane (cm)	Bottom of Core (cm)	Nozzle Centerline (cm)	Weld C1 (cm)	Weld C2 (cm)	Weld C3 (cm)	Outlet (cm)	Inlet (cm)	1/4 Outlet (cm)	1/4 Inlet (cm)
180	270.275	275.296	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
185	269.445	273.781	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
190	266.139	270.894	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
195	259.023	264.348	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
200	255.602	261.420	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
205	251.325	257.172	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
210	249.407	254.921	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
215	244.397	249.290	182.88	0	-182.88	328.524	218.42	-43.78	-309.78		253.52		268.31
220	242.808	246.754	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
225	240.778	245.467	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
230	242.808	246.754	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
235	244.397	249.290	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
240	249.407	254.921	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
245	251.325	257.172	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
250	255.602	261.420	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
255	259.023	264.348	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
260	266.139	270.894	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
265	269.445	273.781	182.88	0	-182.88	328.524	218.42	-43.78	-309.78	263.82		276.54	
270	270.275	275.296	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
275	269.445	273.781	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
280	266.139	270.894	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
285	259.023	264.348	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
290	255.602	261.420	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
295	251.325	257.172	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
300	249.407	254.921	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
305	244.397	249.290	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				

Angle	1E17 n/cm² Fluence Threshold, Z (cm) 52.3 EFPY	72 EFPY	Top Of Core (cm)	Core Midplane (cm)	Bottom of Core (cm)	Nozzle Centerline (cm)	Weld C1 (cm)	Weld C2 (cm)	Weld C3 (cm)	Outlet (cm)	Inlet (cm)	1/4 Outlet (cm)	1/4 Inlet (cm)
310	242.808	246.754	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
315	240.778	245.467	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
320	242.808	246.754	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
325	244.397	249.290	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
330	249.407	254.921	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
335	251.325	257.172	182.88	0	-182.88	328.524	218.42	-43.78	-309.78		253.52		268.31
340	255.602	261.420	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
345	259.023	264.348	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
350	266.139	270.894	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
355	269.445	273.781	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				
360	270.275	275.296	182.88	0	-182.88	328.524	218.42	-43.78	-309.78				

Attachment 3

WCAP-14040-NP-A, REVISION 4, "METHODOLOGY USED TO DEVELOP COLD OVERPRESSURE MITIGATING SYSTEM SETPOINTS AND RCS HEATUP AND COOLDOWN LIMIT CURVES"

Virginia Electric and Power Company
(Dominion Energy Virginia)
North Anna Power Station Units 1 and 2

Westinghouse Non-Proprietary Class 3

WCAP-14040-A
Revision 4

May 2004

Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves

Westinghouse

NRC SAFETY EVALUATION

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 27, 2004

Mr. Gordon Bischoff, Manager
Owners Group Program Management Office
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

RECEIVED
MAR 05 2004
WOG PROJECT OFF

SUBJECT: FINAL SAFETY EVALUATION FOR TOPICAL REPORT WCAP-14040, REVISION 3, "METHODOLOGY USED TO DEVELOP COLD OVERPRESSURE MITIGATING SYSTEM SETPOINTS AND RCS HEATUP AND COOLDOWN LIMIT CURVES" (TAC NO. MB5754)

Dear Mr. Bischoff:

On May 23, 2002, the Westinghouse Owners Group (WOG) submitted Topical Report (TR) WCAP-14040, Revision 3, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," to the staff for review. On February 2, 2004, an NRC draft safety evaluation (SE) regarding our approval of WCAP-14040, Revision 3, was provided for your review and comments. By letter dated February 18, 2004, the WOG commented on the draft SE by indicating that the actual provision number of GL 96-03 should be provided in Sections 3.0 and 4.0 of the SE. In addition, minor editorial comments were provided by the WOG. The staff has incorporated the WOG's suggested comments into the final SE enclosed with this letter.

The staff has found that WCAP-14040, Revision 3, is acceptable for referencing in licensing applications for Westinghouse-designed pressurized water reactors to the extent specified and under the limitations delineated in the report and in the enclosed SE. The SE defines the basis for acceptance of the report.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that the WOG publish an accepted version of this TR within three months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed SE between the title page and the abstract. It must be well indexed such that information is readily located. Also, it must contain in appendices historical review information, such as questions and accepted responses, draft SE comments, and original report pages that were replaced. The accepted version shall include a "-A" (designating accepted) following the report identification symbol.

If the NRC's criteria or regulations change so that its conclusions in this letter, that the TR is acceptable, is invalidated, the WOG and/or the licensees referencing the TR will be expected to revise and resubmit its respective documentation, or submit justification for the continued applicability of the TR without revision of the respective documentation.

Sincerely,

Robert A Gramm for/

Herbert N. Berkow, Director
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 694

Enclosure: Safety Evaluation

cc w/encl:
Mr. James A. Gresham, Manager
Regulatory Compliance and Plant Licensing
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

WCAP-14040, REVISION 3, "METHODOLOGY USED TO DEVELOP COLD OVERPRESSURE MITIGATING SYSTEM SETPOINTS AND REACTOR COOLANT SYSTEM HEATUP AND COOLDOWN LIMIT CURVES"

WESTINGHOUSE OWNERS GROUP

PROJECT NO. 694

1.0 INTRODUCTION

By letter dated May 23, 2002, the Westinghouse Owners Group (WOG) submitted Topical Report (TR) WCAP-14040, Revision 3, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," for NRC staff review and approval. This TR was developed to define a methodology for reactor pressure vessel (RPV) pressure-temperature (P-T) limit curve development and, consistent with the guidance provided in NRC Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," for the development of plant-specific Pressure-Temperature Limit Reports (PTLRs). A prior revision, WCAP-14040, Revision 2, had been approved as a PTLR methodology by the NRC staff's safety evaluation dated October 16, 1995. WCAP-14040, Revision 3, was submitted for NRC staff approval to reflect recent changes in the WOG methodology. Given the scope of the changes incorporated in WCAP-14040, Revision 3, and a significant amount of rewriting which was done to improve clarity of some sections, the NRC staff reviewed the TR in its entirety. Based on questions posed by the NRC staff necessitating clarification of statements or editorial changes, the WOG revised WCAP-14040, Revision 3, and submitted revisions to the TR for NRC staff review and approval by letter dated October 20, 2003.

On February 2, 2004, an NRC draft safety evaluation (SE) regarding our approval of WCAP-14040, Revision 3, was provided for your review and comments. By letter dated February 18, 2004, the WOG commented on the draft SE by indicating that the actual provision number of GL 96-03 should be provided in Sections 3.0 and 4.0 of the SE. In addition, minor editorial comments were provided by the WOG. The staff has incorporated the WOG's comments.

2.0 REGULATORY EVALUATION

Four specific topics are addressed in the context of the development of a PTLR methodology: (1) the calculation of neutron fluences for the RPV and RPV surveillance capsules; (2) the evaluation of RPV material properties due to changes caused by neutron radiation; (3) the development of appropriate P-T limit curves based on these RPV material properties and the establishment of cold overpressure mitigating system (COMS) setpoints to protect the RPV

from brittle failure; and (4) the development of an RPV material surveillance program to monitor changes in RPV material properties due to radiation. Regulatory requirements related to the four topics noted above are addressed in Appendices G and H to Title 10 of the Code of Federal Regulations Part 50 (10 CFR Part 50). Appendix G to 10 CFR Part 50 provides requirements related to RPV P-T limit development and directly or indirectly addresses topics (1) through (3) above. Appendix H to 10 CFR Part 50 defines regulatory requirements related to RPV material surveillance programs and addresses topic (4) above.

For the staff's review of WCAP-14040, Revision 3, several additional guidance documents were used. NRC Standard Review Plan (SRP) Sections 5.2.2, "Overpressure Protection," 5.3.1, "Reactor Vessel Materials," and 5.3.2, "Pressure-Temperature Limits," provide specific review guidance related to RPV material property determination, P-T limit development, and COMS performance. Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," describes analysis procedures acceptable to the NRC staff for the purpose of assessing RPV material property changes due to radiation. RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," addresses NRC staff expectations for an acceptable fluence calculation methodology. American Society for Testing and Materials (ASTM) Standard Practice E 185, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," provides guidance on the establishment of RPV material surveillance programs and editions of ASTM E 185 are incorporated by reference into Appendix H to 10 CFR Part 50. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI, Appendix G provides specific requirements regarding the development of P-T limit curves.

Finally, specific guidance regarding topics and the level of detail to which they must be addressed as part of an acceptable PTLR methodology is given in GL 96-03.

3.0 TECHNICAL EVALUATION

The technical requirements to be addressed in an acceptable PTLR methodology are provided under the column heading "Minimum Requirements to be Included in Methodology" in the table entitled "Requirements for Methodology and PTLR" in Attachment 1 to GL 96-03. Summarized versions of the seven requirements are given below, along with the staff's technical evaluation of information in WCAP-14040, Revision 3, related to each requirement.

Requirement 1: Regarding the reactor vessel material surveillance program, the methodology should briefly describe the surveillance program. The methodology should clearly reference the requirements of Appendix H to 10 CFR Part 50.

The provisions of the methodology described in WCAP-14040, Revision 3, do not specify how the plant-specific RPV surveillance programs should be maintained in order to be in compliance with Appendix H to 10 CFR Part 50. Licensees who wish to use WCAP-14040, Revision 3, as their PTLR methodology must submit additional information to address the methodology requirements discussed in provision 2 in the table of Attachment 1 to GL 96-03 related to the RPV material surveillance program.

Requirement 2: Regarding the calculation of RPV materials' adjusted reference temperatures (ART) values, the methodology should describe the method for calculating material ART values using RG 1.99, Revision 2.

Information regarding how material ARTs are to be determined within the WCAP-14040, Revision 3, PTLR methodology is provided in Sections 2.3 and 2.4 of the TR. In Section 2.3, the determination of initial, unirradiated material properties from Charpy V-notch impact tests and/or nil-ductility drop weight tests is clearly defined. The methodology specified in Section 2.3 accurately incorporates the guidance found in ASME Code Section III, paragraph NB-2331 and additional information in SRP Section 5.3.1.

In Section 2.4 of the TR, the determination of changes in material properties due to irradiation is addressed, along with the determination of margins necessary to account for uncertainties in initial properties and irradiation damage assessment. The methodology specified in Section 2.4 accurately incorporates the guidance found in RG 1.99, Revision 2.

The NRC staff, therefore, determined that the methodology described for determining material ART values in WCAP-14040, Revision 3, was consistent with the guidance provided in the ASME Code, SRP Section 5.3.1, and RG 1.99, Revision 2, and was, therefore, acceptable.

Requirement 3: Regarding the development of RPV P-T limit curves, the methodology should describe the application of fracture mechanics-based calculations in constructing P-T limit curves based on the provisions of Appendix G to Section XI of the ASME Code and SRP Section 5.3.2.

Basic and optional elements of the methodology for RPV P-T limit curve development in WCAP-14040, Revision 3, are given in Sections 2.5, 2.6, 2.7, 2.8, and Appendix A of the TR.

In Section 2.5 of the TR, the fracture toughness-based guidelines from Appendix G to Section XI of the ASME Code are specified (based on the 1995 Edition through 1996 Addenda of the ASME Code). Notably, specific reference is made to the use of: (1) the ASME Code lower bound dynamic crack initiation/crack arrest (K_{IA}) fracture toughness curve; (2) the use of a postulated flaw that has a depth of one-quarter of the wall thickness and a 6:1 aspect ratio; and (3) the use of a structural factor of 2 on primary membrane stress intensities (K_{IM}) when evaluating normal heatup and cooldown and a structural factor of 1.5 on K_{IM} when evaluating hydrostatic/leak test conditions.

Optional guidelines for P-T limit curve development are also addressed in WCAP-14040, Revision 3. The option of using the ASME Code static crack initiation fracture toughness curve (K_{IC}), as given in ASME Code Case N-640, is addressed in Sections 2.5 and 2.8. The option of using ASME Code Case N-588, which enables the postulation of a circumferentially-oriented flaw (with appropriate stress magnification factors) when evaluating a circumferential weld, is addressed in Section 2.8. WCAP-14040, Revision 3, notes, however, that licensee use of the provisions of either ASME Code Case N-640 or N-588 requires, in accordance with 10 CFR 50.60(b), an exemption if the provisions of the Code Case are not contained in the edition of the ASME Code included in a facility's licensing basis. Appendix A to WCAP-14040, Revision 3, provides additional details regarding the application of optional ASME Code Cases and includes

copies of ASME Code Case N-588, N-640, and N-641 (which effectively combines the provisions of N-588 and N-640 into a single Code Case).

A detailed discussion of the calculational methodology for P-T limit curve generation is given in Section 2.6 of the TR. Specific equations are given for the determination of primary membrane stresses due to internal pressure and membrane and bending stresses due to thermal gradients. Equations related to the generation of P-T limit curves for steady-state conditions, finite heatup rates, finite cooldown rates, and hydrostatic/leak test conditions are given. The equations given in WCAP-14040, Revision 3, are equivalent to those provided in Section XI of the ASME Code and consistent with the guidance given in SRP Section 5.3.2.

Therefore, the NRC staff has concluded that the basic methodology specified in WCAP-14040, Revision 3, for establishing P-T limit curves meets the regulatory requirements of Appendix G to 10 CFR Part 50 and the guidance provided in SRP Section 5.3.2. However, the NRC staff has concluded that the discussion provided in WCAP-14040, Revision 3, regarding the use of optional guidelines for the development of P-T limit curves, including the use of ASME Code Cases N-588, N-640, and N-641 is not acceptable. The NRC staff has concluded, based on guidance provided by the NRC's Office of the General Counsel, that licensees do not need to obtain exemptions to use the provisions of ASME Code Case N-588, N-640, or N-641. The basis for this decision is as follows. Appendix G to 10 CFR Part 50 references the use of ASME Code Section XI, Appendix G and defines the acceptable Editions and Addenda of the Code by reference to those endorsed in 10 CFR 50.55a. The 2003 Edition of 10 CFR Part 50, 10 CFR 50.55a, endorses editions and addenda of ASME Section XI up through the 1998 Edition and 2000 Addenda. The provisions of N-588, N-640, and N-641 have been directly incorporated into the Code in the 2000 Addenda version of ASME Section XI, Appendix G. Therefore, licensees may freely make use of the provisions in Code Cases N-588, N-640, and N-641 by using the methodology in the 2000 Addenda version of ASME Section XI without the need for an exemption. When published, the approved revision of TR WCAP-14040 should be modified to reflect this NRC staff conclusion.

Requirement 4: Regarding the development of RPV P-T limit curves, the methodology should describe how the minimum temperature requirements in Appendix G to 10 CFR Part 50 are applied when constructing P-T limit curves.

Minimum temperature requirements regarding the material in the highly stressed region of the RPV flange are given in Appendix G to 10 CFR Part 50. Information provided in Sections 2.9 and 2.10 of the TR addresses the incorporation of minimum temperature requirements into the development of P-T limit curves. In Section 2.9, the 10 CFR Part 50, Appendix G requirements are cited. WCAP-14040, Revision 3, goes on to note that there is an effort underway to revise or eliminate these requirements based on information contained in WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants." However, WCAP-14040, Revision 3, states that until Appendix G to 10 CFR Part 50 is revised to modify/eliminate the existing RPV flange minimum temperature requirements or an exemption request to modify/eliminate these requirements is approved by the NRC for a specific facility, the stated minimum temperature must be incorporated into a facility's P-T limit curves.

WCAP-14040, Revision 3, provides supplemental information in Section 2.10 regarding the establishment of RPV boltup temperature, specifically that the minimum boltup temperature should be 60 °F or equal to the highest material reference temperature in the highly stressed RPV flange region, whichever is higher (i.e., more conservative). Although no specific requirements related to boltup temperature are provided in Appendix G to 10 CFR Part 50, the information in WCAP-14040, Revision 3, is consistent with other, related requirements in Appendix G to 10 CFR Part 50 and in Appendix G to Section XI of the ASME Code.

The NRC staff concludes that the methodology specified in WCAP-14040, Revision 3, addresses RPV minimum temperature requirements in a way which is consistent with Appendix G to 10 CFR Part 50 and Appendix G to Section XI of the ASME Code and is, therefore, acceptable.

Requirement 5: Regarding the calculation of RPV materials' ARTs, the methodology should describe how the data from multiple surveillance capsules may be used in ART calculations.

Requirement 2 of Section 2.4 of WCAP-14040, Revision 3, addresses the determination of changes in material properties due to irradiation. This information includes a description of how surveillance capsule test results may be used to calculate RPV material properties in a manner which is consistent with Section C.2.1 of RG 1.99, Revision 2, and other NRC staff guidance.

The NRC staff has reviewed the information in Section 2.4 of the TR and determined that it is consistent with NRC staff guidance, including RG 1.99, Revision 2, and is, therefore, acceptable.

Requirement 6: Regarding the calculation of the neutron fluence, the methodololgy should describe how the neutron fluence is calculated.

Neutron Fluence Methodology

WCAP-14040, Revision 3, includes a revised Section 2.2. The revised section includes plant-specific transport calculations and the validity of the calculations. For the neutron transport calculations, the applicant is using the two-dimensional discrete ordinates code, DORT (Reference 1) with the BUGLE-96 cross section library (Reference 2). Approximations include a P_5 Legendre expansion for anisotropic scattering and a S_{16} order of angular quadrature. Space and energy dependent core power (neutron source) distributions and associated core parameters are treated on a fuel cycle specific basis. Two dimensional flux solutions $\Phi(r, \theta, z)$ are constructed using (r,θ) and (r,z) distributions. Extreme cases, with respect to power distribution arising from part-length fuel assemblies, use the three-dimensional TORT Code (Reference 1) with the BUGLE-96 cross section library. Source distribution is obtained from a burn-up weighted average of the power distributions of individual fuel cycles. The method accounts for source energy spectral effects and neutrons/fission due to burnup by tracking the concentration of U-235, U-238, Pu-239, Pu-240, and Pu-241. Mesh spacing accounts for flux gradients and material interfaces.

The proposed methodology, as outlined above, adheres to the guidance of RG 1.190, and therefore, is acceptable.

Validation of Transport Calculations

The Westinghouse validation is structured in four parts:

- comparison to pool critical assembly (PCA) simulator results (Reference 3),
- comparison to calculations in the H. B. Robinson benchmark (Reference 4),
- comparison to a measurement database from pressurized water reactor (PWR) surveillance capsules, and
- an analytical sensitivity study addressing the uncertainty components of the transport calculations.

Comparisons of calculated results to the corresponding PCA measured quantities establish the adequacy of the basic transport calculation and the associated cross sections. Comparison to the H.B. Robinson benchmark addresses uncertainties related to the method and generally to the neutron exposure. Comparisons to the PWR database provide an indication of the presence of a bias and of the uncertainty of the calculated value with respect to the corresponding measured values. Finally, the analytical sensitivity study validates the overall uncertainties whether from the methodology or the lack of precise knowledge of the input parameters.

Comparison of the measured data to the calculations was performed on the basis of measured/calculated (M/C) ratios, and with best estimate values calculated using least squares adjusted measured values. The least squares adjustment is based on weighing individual measurements based on spectral coverage. Comparisons are done before and after spectral adjustments. This method is addressed in RG 1.190, as well as in the ASTM Standard E944-96.

The NRC staff requested that the WOG address the completeness of its database. By letter dated October 20, 2003, the WOG responded by indicating that all of the surveillance capsules analyzed with the proposed methodology (DORT and BUGLE-96) are included in the database. The NRC staff found the response acceptable.

The NRC staff concludes that the proposed benchmarking methodology adheres to the guidance in RG 1.190 and to ASTM standards, and therefore, is acceptable.

Requirement 7: Regarding the low temperature overpressure protection/cold overpressure mitigating system, the lift setting limits for the power operated relief valves should be developed using NRC-approved methodologies.

The method in this section is identical to the existing method in the approved Revision 2 of WCAP-14040. The thermal hydraulics analysis for the mass and heat input transients is using the same specialized version of LOFTRAN, which was approved in Revision 2.

The cold overpressure mitigating system is the same as in the approved version, and therefore, the NRC staff finds it acceptable.

4.0 CONCLUSION

The NRC staff has reviewed the information provided in WCAP-14040, Revision 3, related to the requirements of GL 96-03, as cited in Section 3.0 of this SE, and finds WCAP-14040, Revision 3, to be acceptable for referencing as a PTLR methodology, subject to the following conditions:

a. Licensees who wish to use WCAP-14040, Revision 3, as their PTLR methodology must provide additional information to address the methodology requirements discussed in provision 2 in the table of Attachment 1 to GL 96-03 related to the RPV material surveillance program.

b. Contrary to the information in WCAP-14040, Revision 3, licensee use of the provisions of ASME Code Cases N-588, N-640, or N-641 in conjunction with the basic methodology in WCAP-14040, Revision 3, does not require an exemption since the provisions of these Code Cases are contained in the edition and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a. When published, the approved revision (Revision 4) of TR WCAP-14040 should be modified to reflect this NRC staff conclusion.

c. As stated in WCAP-14040, Revision 3, until Appendix G to 10 CFR Part 50 is revised to modify/eliminate the existing RPV flange minimum temperature requirements or an exemption request to modify/eliminate these requirements is approved by the NRC for a specific facility, the stated minimum temperature must be incorporated into a facility's P-T limit curves.

5.0 REFERENCES

1. "DOORS 3.1, One-, Two- and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System," Radiation Shielding Information Center (RSIC) Computer Code Collection CCC-650, Oak Ridge National Laboratory, August 1996.

2. "Bugle-96, Coupled 47 Neutron, 20 Gamma-Ray Group Cross-Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications," RSICL Data Library Collection DLC-185, Oak Ridge National Laboratory, March 1996.

3. NUREG/CR-6454 (ORNL/TM-13205), "Pool Critical Assembly Pressure Vessel Benchmark," by I. Remek and F.B.K. Kam, Oak Ridge National Laboratory, July 1997.

4. NUREG/CR-6453 (ORNL/TM-13204), "H.B. Robinson Pressure Vessel Benchmark," by I. Remek and F.B.K. Kam, Oak Ridge National Laboratory, February 1998.

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WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-14040-A
Revision 4

Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves

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1.0 INTRODUCTION

1.1 BACKGROUND

The concept of a Pressure and Temperature Limits Report (PTLR) was introduced into the Technical Specifications during the development of NUREG 1431[1], Standard Technical Specifications for Westinghouse PWRs and is consistent with the philosophy of NRC Generic Letter 88-16[2]. The PTLR is similar to the Core Operating Limits Report (COLR), which is currently licensed for several plants and also contained in NUREG 1431. The COLR contains core related limit values which may change from cycle to cycle as they are related to a cycle specific core design. In the same way, a PTLR contains reactor vessel material related limits which may change every fluence cycle as they are related to reactor vessel material and strength. Implementation of the PTLR will allow licensees to relocate their RCS heatup and cooldown curves and COMS setpoints currently contained in the Technical Specifications to the PTLR. Additionally, the Vessel Fluence and Materials tables contained in the Technical Specifications or Bases can be relocated to licensee controlled documents. This process will allow changes to these tables, figures and values to be made without making a License Amendment Request (LAR). These figures are typically revised due to changes in the nil ductility reference temperature (RT_{NDT}), regulations and surveillance capsule withdrawal.

1.2 PURPOSE OF TOPICAL REPORT

In order to implement the PTLR, the analytical methods used to develop the pressure and temperature limits must be consistent with those previously reviewed and approved by the NRC and must be referenced in the Administrative Controls section of the Technical Specifications. The purpose of this report is to provide the current Westinghouse methodology for developing the RCS heatup and cooldown curves and COMS setpoints. When approved by the NRC, this methodology may be referenced by licensees to implement the PTLR.

This topical report does not provide all of the methodologies which can be used to develop RCS heatup and cooldown curves and COMS setpoints, but rather methodologies that can be referenced by licensees when approved by the NRC to license the PTLR concept.

1.3 CONTENT OF TOPICAL REPORT

This report contains the methodology used to develop the RCS heatup and cooldown curves in Section 2.0 and the methodology used to develop the COMS setpoints in Section 3.0. The methodology used to develop the COMS enable temperature is also discussed in Section 3.0.

2.0 PRESSURE-TEMPERATURE LIMIT CURVES

2.1 INTRODUCTION

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} (reference nil-ductility transition temperature) corresponding to the limiting material in the beltline region of the reactor vessel. The most limiting RT_{NDT} of the material in the core (beltline) region of the reactor vessel is determined by using the unirradiated reactor vessel material fracture toughness properties and estimating the irradiation-induced shift (ΔRT_{NDT}). The unirradiated RT_{NDT} is defined as the higher of either the drop weight nil-ductility transition temperature (NDTT) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (both normal to the major working direction) minus 60°F.

RT_{NDT} increases as the material is exposed to fast-neutron irradiation. Therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the original unirradiated RT_{NDT}. The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and nickel) present in reactor vessel steels. The Nuclear Regulatory Commission (NRC) has published a method for predicting radiation embrittlement in Regulatory Guide 1.99, Revision 2 (Radiation Embrittlement of Reactor Vessel Materials)[3]. Regulatory Guide 1.99, Revision 2, is used for the calculation of adjusted reference temperature (ART) values (irradiated RT_{NDT} with margins for uncertainties) at 1/4t and 3/4t locations. "t" is the thickness of the vessel at the beltline region measured from the clad/base metal interface (Note, thickness of cladding is neglected as specified in the ASME Code, Section III, paragraph NB-3122.3). Using the adjusted reference temperature values, pressure-temperature limit curves are determined in accordance with the requirements of Appendix G, 10 CFR Part 50[4], as augmented by Appendix G, Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME B&PV) Code[5]. The procedure for establishing the pressure-temperature limits is entirely deterministic. The conservatisms included in the limits are (but not limited to):

- An assumed flaw in the wall of the reactor vessel has a depth equal to 1/4 of the thickness of the vessel wall and a length equal to 1-1/2 times the vessel wall thickness,
- A factor of 2 is applied to the membrane stress intensity factor (K_{IM}),
- 2-sigma margins are applied in determining the adjusted reference temperature (ART), and
- The limiting toughness is based upon a reference value [K_{Ia}, which is a lower bound of the dynamic crack initiation or arrest toughnesses, or K_{Ic}, which is a lower bound of static feature toughness].

This section describes the methodology used by Westinghouse to develop the allowable pressure-temperature relationships for normal plant heatup and cooldown rates that are included in the Pressure-Temperature Limits Report (PTLR). First, the methodology describing how the neutron fluence is calculated for the reactor vessel beltline materials is provided. Next, sections describing fracture toughness properties, adjusted reference temperature calculation, criteria for allowable pressure-temperature relationships, and pressure-temperature curve generation are provided.

2.2 NEUTRON FLUENCE METHODOLOGY

The methodology used to provide neutron exposure evaluations for the reactor pressure vessel is based on the requirements provided in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence."[6]. The vessel exposure projections are based on the results of plant specific neutron transport calculations that are validated by benchmarking of the analytical approach, comparison with industry wide power reactor data bases, and finally, by comparison to plant specific surveillance capsule and reactor cavity dosimetry data. In the validation process, the measurement data are used solely to confirm the accuracy of the transport calculations. The measurements are not used in any way to modify the results of the transport calculations.

2.2.1 Plant Specific Transport Calculations

In the application of the methodology to the fast neutron exposure evaluations for the surveillance capsules and reactor vessel, plant specific forward transport calculations are carried out on a fuel cycle specific basis using the following three-dimensional flux synthesis technique:

$$\phi(r,\theta,z) = [\phi(r,\theta)] * [\phi(r,z)]/[\phi(r)]$$

where:

> $\phi(r,\theta,z)$ is the synthesized three-dimensional neutron flux distribution,
>
> $\phi(r,\theta)$ is the transport solution in r,θ geometry,
>
> $\phi(r,z)$ is the two-dimensional solution for a cylindrical reactor model using the actual axial core power distribution, and
>
> $\phi(r)$ is the one-dimensional solution for a cylindrical reactor model using the same source per unit height as that used in the r,θ two-dimensional calculation.

All of the transport calculations are carried out using the DORT discrete ordinates code Version 3.1[7] and the BUGLE-96 cross-section library[11]. The BUGLE-96 library provides a 67 group coupled neutron-gamma ray cross-section data set produced specifically for light water reactor application. In these analyses, anisotropic scattering is treated with a P_5 legendre expansion and the angular discretization is modeled with an S_{16} order of angular quadrature. Energy and space dependent core power distributions as well as system operating temperatures are treated on a fuel cycle specific basis. The synthesis procedure combining the $\phi(r,\theta)$, $\phi(r,z)$, and $\phi(r)$ transport solutions into the three-dimensional flux/fluence maps within the reactor geometry is accomplished by post-processing the output files generated by the $[r,\theta]$, $[r,z]$, and $[r]$ DORT calculations.

In some extreme cases where part length poisons or shielded fuel assemblies have been inserted into the reactor core to reduce the fluence locally in the vicinity of key vessel materials, the calculational approach may be modified to use either a multi-channel synthesis approach or a fully three-dimensional technique. For the full three-dimensional analysis, the TORT[7] three-dimensional discrete ordinates transport code is used in conjunction with either the BUGLE-96 ENDF/B-VI based library to provide a complete solution without recourse to the use of flux synthesis techniques.

In developing an analytical model of the reactor geometry, nominal design dimensions are normally employed for the various structural components. In some cases as-built dimensions are available; and, in those instances, the more accurate as-built data are used for model development. However, for the most part, as built dimensions of the components in the beltline region of the reactor are not available, thus, dictating the use of design dimensions. Likewise, water temperatures and, hence, coolant density in the reactor core and downcomer regions of the reactor are normally taken to be representative of full power operating conditions. The reactor core itself is treated as a homogeneous mixture of fuel, cladding, water, and miscellaneous core structures such as fuel assembly grids, guide tubes, etc.

The spatial mesh description used in the transport models depends on the overall size of the reactor and on the complexity required to model the core periphery, the in-vessel surveillance capsules, and the details of the reactor cavity. Mesh sizes are chosen to assure that proper convergence of the inner iterations is achieved on a pointwise basis. The pointwise inner iteration flux convergence criterion utilized in the r,θ calculations is set at a value of 0.001.

The mesh selection process results in a smaller spatial mesh in regions exhibiting steep gradients, in material zones of high cross-section (Σ_t), and at material interfaces. In the modeling of in-vessel surveillance capsules, a minimum set of 3 radial by 3 azimuthal mesh are employed within the test specimen array to assure that sufficient information is produced for use in the assessment of fluence gradients within the materials test specimens, as well as in the determination of gradient corrections for neutron sensors. Additional radial and azimuthal mesh are employed to model the capsule structure surrounding the materials test specimen array. In modeling the stainless steel baffle region at the periphery of the core, a relatively fine spatial mesh is required to adequately describe this rectilinear component in r,θ geometry. In performing this x,y to r,θ transition, care is taken to preserve both the thickness and volume of the steel region in order to accurately address the shielding effectiveness of the component.

The spatial variation of the neutron source is generally obtained from a burnup weighted average of the respective power distributions from individual fuel cycles. These spatial distributions include pinwise gradients for all fuel assemblies located at the periphery of the core and typically include a uniform or flat distribution for fuel assemblies interior to the core. The spatial component of the neutron source is transposed from x,y to [r,θ], [r,z], and [r] geometry by overlaying the mesh schematic to be used in the transport calculation on the pin by pin array and then computing the appropriate relative source applicable to each spatial interval within the reactor core.

These x,y to [r,θ], [r,z], and [r] transpositions are accomplished by first defining a fine mesh working array. The sizes of the fine mesh are usually chosen so that there is at least a 10x10 array of fine mesh over the area of each fuel pin at the core periphery. The coordinates of the center of each fine mesh interval and its associated relative source strength are assigned to the fine mesh based on the pin that is coincident with the center of the fine mesh. In the limit as the sizes of the fine mesh approach zero, this technique becomes an exact transformation.

Each space mesh in the transport geometry is checked to determine if it lies totally within the area of a particular fine working mesh. If it does, the relative source of that fine mesh is assigned to the transport space mesh. If, on the other hand, the transport space mesh covers a part of one or more fine mesh, then the relative source assigned to the transport mesh is determined by an area weighting process as follows:

$$P_m = \frac{\sum_i A_i P_i}{\sum_i A_i}$$

where:

P_m = the relative source assigned to transport mesh m.

A_i = the area of fine working mesh i within transport mesh m.

P_i = the relative source within fine working mesh i.

The energy distribution of the source is determined on a fuel assembly specific basis by selecting a fuel assembly burnup representative of conditions averaged over each fuel cycle and an initial enrichment characteristic for each assembly. From this average burnup and initial enrichment, a fission split by isotope including ^{235}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, and ^{241}Pu is derived; and, from that fission split, composite values of energy release per fission, neutron yield per fission, and fission spectrum are determined for each fuel assembly. These composite values are then combined with the spatial distribution to produce the overall absolute neutron source for use in the transport calculations.

2.2.2 Validation of the Transport Calculations

The validation of the methodology described in Section 2.2.1 is based on the guidance provided in Regulatory Guide 1.190. In particular, the validation consists of the following stages:

1. Comparisons of calculations with benchmark measurements from the Pool Critical Assembly (PCA) simulator at the Oak Ridge National Laboratory (ORNL)[12].

2. Comparisons of calculations with surveillance capsule and reactor cavity measurements from the H. B. Robinson power reactor benchmark experiment[22].

3. An analytical sensitivity study addressing the uncertainty components resulting from important input parameters applicable to the plant specific transport calculations used in the exposure assessments.

4. Comparisons of calculations with a measurements data base obtained from a large number of surveillance capsules withdrawn from a variety of pressurized water reactors.

At each subsequent application of the methodology, comparisons are made with plant specific dosimetry results to demonstrate that the plant specific transport calculations are consistent with the uncertainties derived from the methods qualification.

The first stage of the methods validation addresses the adequacy of basic transport calculation and dosimetry evaluation techniques and associated cross-sections. This stage, however, does not test the accuracy of commercial core neutron source calculations nor does it address uncertainties in operational or geometric variables that impact power reactor calculations. The second stage of the validation addresses uncertainties that are primarily methods related and would tend to apply generically to all fast neutron exposure evaluations. The third stage of the validation identifies the potential uncertainties introduced into the overall evaluation due to calculational methods approximations, as well as to a lack of knowledge relative to various plant specific parameters. The overall calculational uncertainty is established from the results of these three stages of the validation process.

The following summarizes the uncertainties determined from the results of the first three stages of the validation process:

PCA Benchmark Comparisons	3%
H. B. Robinson Benchmark Comparisons	3%
Analytical Sensitivity Studies	11%
Internals Dimensions	3%
Vessel Inner Radius	5%
Water Temperature	4%
Peripheral Assembly Source Strength	5%
Axial Power Distribution	5%
Peripheral Assembly Burnup	2%
Spatial Distribution of the Source	4%
Other Factors	5%

The category designated "Other Factors" is intended to attribute an additional uncertainty to other geometrical or operational variables that individually have an insignificant impact on the overall uncertainty, but collectively should be accounted for in the assessment.

The uncertainty components tabulated above represent percent uncertainty at the 1σ level. In the tabulation, the net uncertainty of 11% from the analytical sensitivity studies has been broken down into its individual components. When the four uncertainty values listed above (3%, 3%, 11%, and 5%) are combined in quadrature, the resultant overall 1σ calculational uncertainty is estimated to be 13%.

To date the methodology described in Section 2.2.1 coupled with the BUGLE-96 cross-section library has been used in the evaluation of dosimetry sets from 82 surveillance capsules from 23 pressurized water reactors. These capsule withdrawals included 2-5 capsules from individual reactors. The comparisons of the plant specific calculations with the results of the capsule dosimetry are used to further validate the calculational methodology within the context of a 1σ calculational uncertainty of 13%.

This 82 capsule data base includes all surveillance capsule dosimetry sets analyzed by Westinghouse using the Bugle-96 cross-section library and the synthesis approach described in Section 2.2.1. No surveillance capsule dosimetry sets were excluded from the M/C data base. As additional capsules are

analyzed using the synthesis approach with the BUGLE-96 cross-section library the M/C comparisons will be added to the database.

The comparisons between the plant specific calculations and the data base measurements are provided on two levels. In the first instance, measurement to calculation (M/C) ratios for each fast neutron sensor reaction rate from the surveillance capsule irradiations are listed. This tabulation provides a direct comparison, on an absolute basis, of measurement and calculation. The results of this comparison for the surveillance capsule data base are as follows:

REACTION	M/C	STD DEV
$^{63}Cu(n\alpha)^{60}Co$	1.09	7.9%
$^{54}Fe(n,p)^{54}Mn$	0.99	8.4%
$^{58}Ni(n,p)^{58}Co$	0.99	8.9%
$^{238}U(n,f)^{137}Cs$	1.01	11.8%
$^{237}Np(n,f)^{137}Cs$	1.06	11.3%
Linear Average	1.03	9.8%

These comparisons show that the calculations and measurements for the surveillance capsule data base fall well within the 13% calculational uncertainty for all of the fast neutron reactions.

The second comparison of calculations with the data base is based on the least squares adjustment of the individual surveillance capsule data sets. The least squares adjustment procedure provides a weighting of the individual sensor measurements based on spectral coverage and allows a comparison of the neutron flux (E > 1.0 MeV) before and after adjustment. The neutron flux/fluence (E > 1.0 MeV) is the primary parameter of interest in the overall pressure vessel exposure evaluations.

The least squares evaluations of the 82 surveillance capsule dosimetry sets followed the guidance provided in Section 1.4.2 of Regulatory Guide 1.190 and in ASTM Standard E944-96, "Standard Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance."

The application of the least squares methodology requires the following input:

1. The calculated neutron energy spectrum and associated uncertainties at the measurement location.

2. The measured reaction rates and associated uncertainty for each sensor contained in the multiple foil set.

3. The energy dependent dosimetry reaction cross-sections and associated uncertainties for each sensor contained in the multiple foil sensor set.

For the data base comparisons, the calculated neutron spectra were obtained from the results of plant specific neutron transport calculations applicable to each of the 82 surveillance capsules. The sensor reaction rates and dosimetry cross-sections were the same as those used in the direct M/C comparisons noted above.

The results of this latter comparison expressed in terms of the ratio of adjusted flux to calculated flux (A/C) are summarized as follows for the 82 capsule data base:

PARAMETER	A/C	STD DEV
$\phi(E > 1.0\ MeV)$	1.00	7.3%

As with the comparisons based on the linear average of reaction rate M/C ratios, the comparisons of the least squares adjusted results with the plant specific transport calculations demonstrate that the calculated results are essentially unbiased with an uncertainty well within the 20% criterion established in Regulatory Guide 1.190.

2.3 FRACTURE TOUGHNESS PROPERTIES

The fracture toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the requirements of Appendix G, 10 CFR Part 50[4], as augmented by the additional requirements in subsection NB-2331 of Section III of the ASME B&PV Code[8]. These fracture toughness requirements are also summarized in Branch Technical Position MTEB 5-2 ("Fracture Toughness Requirements")[9] of the NRC Regulatory Standard Review Plan.

These requirements are used to determine the value of the reference nil-ductility transition temperature (RT_{NDT}) for unirradiated material (defined as initial RT_{NDT}, IRT_{NDT}) and to calculate the adjusted reference temperature (ART) as described in Section 2.4. Two types of tests are required to determine a material's value of IRT_{NDT}: Charpy V-notch impact (C_v) tests and drop-weight tests. The procedure is as follows:

1. Determine a temperature T_{NDT} that is at or above the nil-ductility transition temperature by drop weight tests.

2. At a temperature not greater than T_{NDT} + 60°F, each specimen of the C_v test shall exhibit at least 35 mils lateral expansion and not less than 50 ft-lb absorbed energy. When these requirements are met, T_{NDT} is the reference temperature RT_{NDT}.

3. If the requirements of (2) above are not met, conduct additional C_v tests in groups of three specimens to determine the temperature T_{Cv} at which they are met. In this case the reference temperature $RT_{NDT} = T_{Cv}$ - 60°F. Thus, the reference temperature RT_{NDT} is the higher of T_{NDT} and (T_{Cv} - 60°F).

4. If the C_v test has not been performed at T_{NDT} + 60°F, or when the C_v test at T_{NDT} + 60°F does not exhibit a minimum of 50 ft-lb and 35 mils lateral expansion, a temperature representing a minimum of 50 ft-lb and 35 mils lateral expansion may be obtained from a full C_v impact curve developed from the minimum data points of all the C_v tests performed as shown in Figure 2.1.

Plants that do not follow the fracture toughness guidelines in Branch Technical Position MTEB 5-2 to determine IRT_{NDT} can use alternative procedures. However, sufficient technical justification and special circumstances per the criteria of 10CFR50.12(a)(2) must be provided for an exemption from the regulations to be granted by the NRC.

2.4 CALCULATION OF ADJUSTED REFERENCE TEMPERATURE

The adjusted reference temperature (ART) for each material in the beltline region is calculated in accordance with Regulatory Guide 1.99, Revision 2[3]. The most limiting ART values (i.e., highest value at 1/4t and 3/4t locations) are used in determining the pressure-temperature limit curves. ART is calculated by the following equation:

$$ART = IRT_{NDT} + \Delta RT_{NDT} + \text{Margin} \qquad (2.4\text{-}1)$$

IRT_{NDT} is the reference temperature for the unirradiated material as defined in paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code[8] and calculated per Section 2.3. If measured values of IRT_{NDT} are not available for the material in question, generic mean values for that class of material can be used if there are sufficient test results to establish a mean and standard deviation for the class.

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

$$\Delta RT_{NDT} = CF\ f^{(0.28 - 0.10 \log f)} \qquad (2.4\text{-}2)$$

CF (°F) is the chemistry factor and is a function of copper and nickel content. CF is given in Table 1 of Reference 3 for weld metal and in Table 2 in Reference 3 for base metal (Position 1.1 of Regulatory Guide 1.99, Revision 2). In Tables 1 and 2 of Reference 3 "weight-percent copper" and "weight-percent nickel" are the best-estimate values for the material and linear interpolation is permitted. When two or more credible surveillance data sets (as defined in Regulatory Guide 1.99, Revision 2, Paragraph B.4) become available they may be used to calculate the chemistry factor per Position 2.1 of Regulatory Guide 1.99, Revision 2, as follows:

$$CF = \frac{\sum_{i=1}^{n} [A_i\ f_i^{(0.28-0.10\log f_i)}]}{\sum_{i=1}^{n} [f_i^{\ (0.28-0.10\log f_i)}]^2} \qquad (2.4\text{-}3)$$

Where "n" is the number of surveillance data points, "A_i" is the measured shift in the Charpy V-notch 30 ft-lb energy level between the unirradiated condition and the irradiated condition, "f_i." Where "f_i" is the fluence for each surveillance data point.

If Position 2.1 of Regulatory Guide 1.99, Revision 2, results in a higher value of ART than Position 1.1 of Regulatory Guide 1.99, Revision 2, the ART calculated per Position 2.1 must be used. However, if Position 2.1 of Regulatory Guide 1.99, Revision 2, results in a lower value of ART than Position 1.1 of Regulatory Guide 1.99, Revision 2, either value of ART may be used.

To calculate ΔRT_{NDT} at any depth (e.g., at 1/4t or 3/4t), the following formula is used to attenuate the fast neutron fluence (E> 1 MeV) at the specified depth.

$$f = f_{surface}\, e^{(-0.24x)} \tag{2.4-4}$$

where $f_{surface}$ 10^{19} n/cm^2, E > 1 MeV) is the value, calculated per Section 2.2, of the neutron fluence at the base metal surface of the vessel at the location of the postulated defect, and x (in inches) is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then put into equation (2.4-2) to calculate ΔRT_{NDT} at the specified depth.

When two or more credible surveillance capsules have been removed, the measured increase in reference temperature (ΔRT_{NDT}) must be compared to the predicted increase in RT_{NDT} for each surveillance material. The predicted increase in RT_{NDT} is the mean shift in RT_{NDT} calculated by equation (2.4-2) plus two standard deviations ($2\sigma_\Delta$) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value ($\Delta RT_{NDT} + 2\sigma_\Delta$), a supplement to the PTLR must be submitted for NRC review and approval to demonstrate how the results affect the approved methodology.

Margin is the temperature value that is included in the ART calculations to obtain conservative, upper-bound values of ART for the calculations required by Appendix G to 10 CFR Part 50[(4)]. Margin is calculated by the following equation:

$$\text{Margin} = 2\,[(\sigma_I^2 + \sigma_\Delta^2)]^{0.5} \tag{2.4-5}$$

σ_I, is the standard deviation for IRT_{NDT} and σ_Δ is the standard deviation for ΔRT_{NDT}. If IRT_{NDT} is a measured value, σ_I, is estimated from the precision of the test method ($\sigma_I = 0$ for a measured IRT_{NDT} of a single material). If IRT_{NDT} is not a measured value and generic mean values for that class of material are used, σ_I is the standard deviation obtained from the set of data used to establish the mean. Per Regulatory Guide 1.99, σ_Δ is 28°F for welds and 17°F for base metal. When surveillance data is used to calculate ΔRT_{NDT}, σ_Δ values may be reduced by one-half. In all cases, σ_Δ need not exceed half of the mean value of ΔRT_{NDT}.

2.5 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

The ASME Code requirements[(5)] for calculating the allowable pressure-temperature limit curves for various heatup and cooldown rates specify 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, the fracture toughness for the metal temperature at that time. Two values of fracture toughness may be used, K_{Ia} or K_{Ic}.

K_{Ia} is obtained from the reference fracture toughness curve, defined in Appendix G, to Section XI of the ASME Code (1995 Edition through the 1996 Addenda). (Note that in Appendix G, to Section III of the ASME Code, the reference fracture toughness is denoted as K_{IR}, whereas in Appendix G of Section XI, the reference fracture toughness is denoted as K_{Ia}. However, the K_{IR} and K_{Ia} curves are identical and are defined with the identical functional form.) The K_{Ia} curve is given by the following equation:

$$K_{Ia} = 26.78 + 1.223 \exp[0.0145\,(T - RT_{NDT} + 160)] \tag{2.5-1}$$

where,

K_{Ia} = lower bound of dynamic and crack arrest toughness as a function of the metal temperature T and the metal reference nil-ductility transition temperature RT_{NDT}, (ksi $\sqrt{in}$). The value of RT_{NDT} is the adjusted reference temperature (ART) of Section 2.4.

K_{Ic} is also obtained from Section XI of the ASME Code, for example in Appendix A, and is a lower bound of static fracture toughness. Since heatup and cooldown is a slow process, static properties are appropriate. The K_{Ic} curve is given by the following expression:

$$K_{Ic} = 33.20 + 20.734 \exp [0.0200 (T - RT_{NDT})] \quad (2.5\text{-}2)$$

The use of the K_{Ic} curve (Section XI, Appendix A) as a basis for developing P-T limit curves is currently contained in ASME Code Case N640. Use of the K_{Ic} fracture toughness will yield less limiting P-T curves, which is clearly a benefit.

However, the use of Code Case 640 presently includes a restriction on the setpoints for the Cold Overpressure Mitigation System (COMS). This maximum pressure for the COMS system is 100% of the pressure allowed by the P-T limit curves. This essentially disallows the use of Code Case N514 in these circumstances, meaning that the COMS system must protect to the actual P-T limit curve, rather than 110 percent, as allowed by Code Case N514.

The governing equation for generating pressure-temperature limit curves is defined in Appendix G of the ASME Code[5] as follows:

$$C\, K_{IM} + K_{It} < \text{Reference Fracture Toughness} \quad (2.5\text{-}3)$$

where,

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

K_{It} = stress intensity factor caused by the thermal gradients through the vessel wall,

C = 2.0 for Level A and Level B service limits (for heatup and cooldown),

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

Reference Fracture Toughness = K_{Ia} or K_{Ic}, as discussed above

(Note: K_{It} is set to zero for hydrostatic and leak test calculations since these tests are performed at isothermal conditions).

At specific times during the heatup or cooldown transient, the reference fracture toughness is determined by the metal temperature at the tip of the postulated flaw (the postulated flaw has a depth of one-fourth of the section thickness and a length of 1.5 times the section thickness per ASME Code, Section XI, paragraph G-2120), the appropriate value for RT_{NDT} at the same location, and the reference fracture toughness equation (2.5-1 or 2.5-2). The thermal stresses resulting from the temperature gradients

through the vessel wall and the corresponding (thermal) stress intensity factor, K_{It}, for the reference flaw are calculated as described in Section 2.6. From Equation (2.5-3), the limiting pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated as described in Section 2.6.

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

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

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

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4t flaw at the inside of the wall. The heatup results in compressive stresses at the inside surface that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the reference fracture toughness for the inside 1/4t flaw during heatup is lower than the reference fracture toughness for the same flaw during steady-state conditions at the same coolant temperature. However, conditions may exist so that the effects of compressive thermal stresses and lower reference fracture toughness do not offset each other and the pressure-temperature curve based on finite heatup rates could become limiting. Therefore, both cases have to be analyzed in order to ensure that at any coolant temperature, the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained for the inside 1/4t flaw.

The third portion of the heatup analysis concerns the calculation of the pressure-temperature limitations for the case of a 1/4t outside surface flaw. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and therefore tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the

rate of heatup and coolant temperature during the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate is analyzed on an individual basis.

Following the generation of the three pressure-temperature curves, the final limit curves are produced by constructing a composite curve based on a point-by-point comparison of the steady-state data and finite heatup rate data for both inside and outside surface flaws. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration. The use of the composite curve is necessary to set conservative heatup limitations because it is not possible to predict which condition is most limiting because of local differences in irradiation (RT_{NDT}), metal temperature and thermal stresses. With the composite curve, the pressure limit is at all times based on analysis of the most critical situation.

Finally, the 1983 Amendment to 10CFR50[4] has a rule which addresses the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure flange regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation and 90°F for hydrostatic pressure tests and leak tests when the pressure exceeds 20 percent of the preservice hydrostatic test pressure. In addition, when the core is critical, the pressure-temperature limits for core operation (except for low power physics tests) require that the reactor vessel be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum permissible temperature in the corresponding pressure-temperature curve for heatup and cooldown. These limits are incorporated into the pressure-temperature limit curves wherever applicable.

A petition for rulemaking to eliminate the flange requirement contained in 10CFR50 Appendix G was submitted to the NRC by Westinghouse in November 1999. Until 10CFR50 Appendix G is revised to eliminate the flange requirement, it must be included in the P-T limits, unless an exemption request is submitted and approved by the NRC.

Figure 2.2 shows an example of a heatup curve using a heatup rate of 60°F/Hr applicable for the first 16 EFPY. Figure 2.3 shows an example of cooldown curves using rates of 0°, 20°, 40°, 60°, and 100°F/Hr applicable for the first 16 EFPY. Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown in Figures 2.2 and 2.3. Note that the step in these curves are due to the previously described flange requirements [4].

2.6 PRESSURE-TEMPERATURE CURVE GENERATION METHODOLOGY

2.6.1 Thermal and Stress Analyses

The time-dependent temperature solution utilized in both the heatup and cooldown analysis is based on the one-dimensional transient heat conduction equation:

$$\rho C \frac{\partial T}{\partial t} = K \left[\frac{\partial^2 T}{\partial r^2} + \frac{1}{r}\frac{\partial T}{\partial r} \right] \qquad (2.6.1\text{-}1)$$

with the following boundary conditions applied at the inner and outer radii of the reactor vessel,

$$\text{at } r = r_i, \quad -K\frac{\partial T}{\partial r} = h(T - T_c) \tag{2.6.1-2}$$

$$\text{at } r = r_o, \quad \frac{\partial T}{\partial r} = 0 \tag{2.6.1-3}$$

where,

r_i = reactor vessel inner radius

r_o = reactor vessel outer radius

ρ = material density

C = material specific heat

K = material thermal conductivity

T = local temperature

r = radial location

t = time

h = heat transfer coefficient between the coolant and the vessel wall

T_c = coolant temperature

These equations are solved numerically to generate the position and time-dependent temperature distributions, T(r,t), for all heatup and cooldown rates of interest.

With the results of the heat transfer analysis as input, position and time-dependent distributions of hoop thermal stress are calculated using the formula for the thermal stress in a hollow cylinder given by Timoshenko[14].

$$\sigma_\theta(r,t) = \frac{E\alpha}{1-v}\frac{1}{r^2}\left[\frac{r^2 + r_i^2}{r_o^2 - r_i^2}\int_{r_i}^{r_o} T(r,t)r\,dr + \int_{r_i}^{r} T(r,t)r\,dr - T(r,t)r^2\right] \tag{2.6.1-4}$$

where,

$\sigma_\theta(r,t)$ = hoop stress at location and time t

E = modulus of elasticity

α = coefficient of linear expansion

V = Poisson's ratio

The quantities E and α are temperature-dependent properties. However, to simplify the analysis, E and α are evaluated at an equivalent wall temperature at a given time:

$$T_{eqv} = \frac{2\int_{r_i}^{r_o} T(r)r\,dr}{r_o^2 - r_i^2} \tag{2.6.1-5}$$

E and α are calculated as a function of this equivalent temperature and the Eα product in equation (2.6.1-4) is treated as a constant in the computation of hoop thermal stress.

The linear bending (σ_b) and constant membrane (σ_m) stress components of the thermal hoop stress profile are approximated by the linearization technique presented in Appendix A, to Section XI of the ASME Code[15]. These stress components are used for determining the thermal stress intensity factors, K_{It}, as described in subsections 2.6.3 and 2.6.4.

2.6.2 Steady-State Analyses

Using the calculated beltline metal temperature and the metal reference nil-ductility transition temperature, the reference stress intensity factor (K_{Ia}) is determined in Equation (2.5-1) at the 1/4t location where "t" represents the vessel wall thickness. At the 1/4t location, a 1/4 thickness flaw is assumed to originate at the vessel inside radius.

The allowable pressure $P(T_c)$ is a function of coolant temperature, and the pressure temperature curve is calculated for the steady state case at the assumed 1/4t inside surface flaw. First, the maximum allowable membrane (pressure) stress intensity factor is determined using the factor of 2.0 from equation (2.5-2) and the following equation:

$$K_{IM(max)} = \frac{K_I * (T - RT_{NDT})_{1/4t}}{2.0} \tag{2.6.2-1}$$

where,

$K_{Ia}(T\text{-}RT_{NDT})$ = allowable reference stress intensity factor as a function of $T\text{-}RT_{NDT}$ at 1/4t. (See Sections 2.7 and 2.8 for the new approach using Code Cases N640 and N588.)

Next, the maximum allowable pressure stress is determined using an iterative process and the following three equations:

$$Q = \varphi^2 - 0.212\left(\frac{\sigma_P}{\sigma_y}\right)^2 \tag{2.6.2-2}$$

$$\sigma_P = \frac{K_{IM(max)}}{1.1\,M_K\sqrt{\frac{\pi a}{Q}}} \tag{2.6.2-3}$$

$$K_{IP} = 1.1 M_K \sigma_P \sqrt{\frac{\pi a}{Q}} \tag{2.6.2-4}$$

where,

Q	=	flaw shape factor modified for plastic zone size[16],
ϕ	=	is the elliptical integral of the 2nd kind (ϕ = 1.11376 for the fixed aspect ratio of 3 of the code reference flaw)[16],
0.212	=	plastic zone size correction factor[16],
σ_P	=	pressure stress,
σ_y	=	yield stress,
1.1	=	correction factor for surface breaking flaws,
M_K	=	correction factor for constant membrane stress [16], M_K as function of relativeflaw depth (a/t) is shown in Figure 2.4,
a	=	crack depth of 1/4t,
K_{IP}	=	pressure stress intensity factor.

The maximum allowable pressure stress is determined by incrementing σ_p from an initial value of 0.0 psi until a pressure stress is found that computes a K_{IP} value within 1.0001 of the $K_{IM(max)}$ value. After the maximum allowable σ_p is found, the maximum allowable internal pressure is determined by

$$P(T_c) = \sigma_P \left[\frac{r_o^2 - r_i^2}{r_o^2 + r_i^2}\right] \tag{2.6.2-5}$$

where,

$P(T_c)$	=	calculated allowable pressure as a function of coolant temperature.

2.6.3 Finite Cooldown Rate Analyses

For each cooldown rate the pressure-temperature curve is calculated at the inside 1/4t location. First, the thermal stress intensity factor is calculated for a coolant temperature at a given time using the following equation from the Welding Research Council[16]:

$$K_{It} = [\sigma_m 1.1 M_K + \sigma_b M_B]\sqrt{\frac{\pi a}{Q}} \qquad (2.6.3\text{-}1)$$

where,

σ_m = constant membrane stress component from the linearized thermal hoop stress distribution,

σ_b = linear bending stress component from the linearized thermal hoop stress distribution,

M_K = correction factor for membrane stress[16] (see Figure 2.4),

M_B = correction factor for bending stress[16], M_B as a function of relative flaw depth (a/t) is shown in Figure 2.5.

The flaw shape factor Q in equation (2.6.2-6) is calculated from[16]

$$Q = \phi^2 - 0.212\left(\frac{\sigma_m + \sigma_b}{\sigma_y}\right)^2 \qquad (2.6.3\text{-}2)$$

Once K_{It} is computed, the maximum allowable membrane (pressure) stress intensity factor is determined using the factor of 2.0 from equation (2.5-2) and the following equation:

$$K_{IM(max)} = \frac{K_I * (T - RT_{NDT})_{1/4t} - K_{It}(T_c)_{1/4t}}{2.0} \qquad (2.6.3\text{-}3)$$

From $K_{IM(max)}$, the maximum allowable pressure is determined using the iterative process described above and equations (2.6.2-2) through (2.6.2-5).

The steady-state pressure-temperature curve of Section 2.6.2 is compared to the cooldown curves for the 1/4t inside surface flaw at each cooldown rate. At any time, the allowable pressure is the lesser of the two values, and the resulting curve is called the composite cooldown limit curve.

Finally, the 10 CFR Part 50[4] requirement for the closure flange region is incorporated into the cooldown composite curve as described in Section 2.5.

2.6.4 Finite Heatup Rate Analyses

Using the calculated beltline metal temperature and the metal reference nil-ductility transition temperature, the reference stress intensity factor (K_{Ia}) is determined in Equation (2.5-1) or (2.5-2) at both the 1/4t and 3/4t locations where "t" represents the vessel wall thickness. At the 1/4t location, a 1/4 thickness flaw is assumed to originate at the vessel inside radius. At the 3/4t location, a 1/4t flaw is assumed to originate on the outside of the vessel.

For each heatup rate a pressure-temperature curve is calculated at the 1/4t and 3/4t locations. First, the thermal stress intensity factor is calculated at the 1/4t and 3/4t locations for a coolant temperature at a given time using Option 1 or 2 from Section 2.6.3.

Once K_{It} is computed, the maximum allowable membrane (pressure) stress intensity factors at the 1/4t and 3/4t locations are determined using the following equations:

$$\text{At 1/4t,}\quad K_{IM(max)1/4t} = \frac{K_I * (T - RT_{NDT})_{1/4t} - K_{It}(T_c)_{1/4t}}{2.0} \tag{2.6.4-1}$$

$$\text{At 3/4t,}\quad K_{IM(max)3/4t} = \frac{K_I * (T - RT_{NDT})_{3/4t} - K_{It}(T_c)_{3/4t}}{2.0} \tag{2.6.4-2}$$

From $K_{IM(max)1/4t}$ and $K_{IM(max)3/4t}$, the maximum allowable pressure at both the 1/4t and 3/4t locations is determined using the iterative process described in Section 2.6.2 and equations (2.6.2-2) through (2.6.2-5).

As was done with the cooldown case, the steady state pressure-temperature curve of Section 2.6.2 is compared with the 1/4t and 3/4t location heatup curves for each heatup rate, with the lowest of the three being used to generate the composite heatup limit curve. The composite curve is then adjusted for the 10 CFR Part 50(4) rule for closure flange requirements, as discussed in Section 2.5.

2.6.5 Hydrostatic and Leak Test Curve Analyses

The minimum inservice hydrostatic leak test curve is determined by calculating the minimum allowable temperature at two pressure values (pressure values of 2000 psig and 2485 psig, approximately 110% of operating pressure, are generally used). The curve is generated by drawing a line between the two pressure-temperature data points. The governing equation for generating the hydrostatic leak test pressure-temperature limit curve is defined in Appendix G, Section X1, of the ASME Code(5) as follows:

$$1.5\, K_{IM} < K_{Ia} \tag{2.6.5-1}$$

where, K_{IM} is the stress intensity factor caused by the membrane (pressure) stress and K_{Ia} is the reference stress intensity factor as defined in equation (2.5-1). Note that the thermal stress intensity factor is neglected (i.e., $K_{It} = 0$) since the hydrostatic leak test is performed at isothermal conditions.

The pressure stress is determined by,

$$\sigma_p = \left[\frac{r_o^2 + r_i^2}{r_o^2 - r_i^2}\right] P \qquad (2.6.5\text{-}2)$$

where,

P = the input pressure (generally 2000 and 2485 psig)

Next, the pressure stress intensity factor is calculated for a 1/4t flaw by,

$$K_{IM} = \left[1.1 M_K \sqrt{\frac{\pi a}{Q}}\right] \sigma_p \qquad (2.6.5\text{-}3)$$

The K_{IM} result is multiplied by the 1.5 factor of equation (2.5-2) and divided by 1000,

$$K_{HYD} = \frac{1.5 K_{IM}}{1000} \qquad (2.6.5\text{-}4)$$

Finally, the minimum allowable temperature is determined by setting K_{HYD} to K_{Ia} in equation (2.5-1) and solving for temperature T:

$$T = \frac{\ell n\left[\frac{(K_{HYD} - 26.78)}{1.223}\right]}{0.0145} + RT_{NDT} - 160.0 \qquad (2.6.5\text{-}5)$$

The 1983 Amendment to 10CFR50(4) has a rule which addresses the test temperature for hydrostatic pressure tests. This rule states that, when there is no fuel in the reactor vessel during hydrostatic pressure tests or leak tests, the minimum allowable test temperature must be 60°F above the adjusted reference temperature of the beltline region material that is controlling. If fuel is present in the reactor vessel during hydrostatic pressure tests or leak tests, the requirements of this section and Section 2.5 must be met.

2.7 1996 ADDENDA TO ASME SECTION XI, APPENDIX G METHODOLOGY

ASME Section XI, Appendix G was updated in 1996 to incorporate the most recent elastic solutions for K_I due to pressure and radial thermal gradients. The new solutions are based on finite element analyses for inside surface flaws performed at Oak Ridge National Laboratories and sponsored by the NRC, and work published for outside surface flaws. These solutions provide results that are very similar to those obtained by using solutions previously developed by Raju and Newman.

This revision provides consistent computational methods for pressure and thermal K_I, for thermal gradients through the vessel wall at any time during the transient. Consistent with the original version of

Appendix G, no contribution for crack face pressure is included in the K_I due to pressure, and cladding effects are neglected.

Using these elastic solutions in the low temperature region will provide some relief to restrictions associated with reactor operation at relatively low temperatures. Although the relief is relatively small in terms of the absolute allowable pressure, the benefits are substantial, because even a small increase in the allowable pressure can be a significant percentage increase in the operating window at relatively low temperatures. Implementing this revision results in a safety benefit (reduced likelihood of lifting COMS relief valves), with no reduction in vessel integrity.

The following revisions were made to ASME Section XI, Appendix G:

G-2214.1 Membrane Tension:

$$K_{Im} = M_m \times (pR_i / t) \qquad (2.7\text{-}1)$$

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

M_m = 1.85 for $\sqrt{t} < 2$,

M_m = $0.926\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$,

M_m = 3.21 for $\sqrt{t} > 3.464$

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

M_m = 1.77 for $\sqrt{t} < 2$,

M_m = $0.893\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$,

M_m = 3.09 for $\sqrt{t} > 3.464$

where,

p = internal pressure,

Ri = vessel inner radius, and

t = vessel wall thickness.

For Bending Stress, the K_I corresponding to bending stress for the postulated defect is:

$K_{Ib} = M_b$ * maximum bending stress, where $M_b = 0.667\ M_m$

For the Radial Thermal Gradient, the maximum K_I produced by radial thermal gradient for the postulated inside surface defect is:

$$K_{It} = 0.953 \times 10^{-3}\ CR\ t^{2.5} \qquad (2.7\text{-}2)$$

where:

CR = the cooldown rate in °F/hr.

For the Radial Thermal Gradient, the maximum K_I produced by radial thermal gradient for the postulated outside surface defect is:

$$K_{It} = 0.753 \times 10^{-3} \text{ HU } t^{2.5} \tag{2.7-3}$$

where:

HU = the heatup rate in °F/hr.

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

1. The maximum thermal K_I relationship and the temperature relationship in Figure G-2214-1 are applicable only for the conditions given in G-2214.3(a)(1) and (2) of Appendix G to ASME Section XI.

2. Alternatively, the K_I for radial thermal gradient can be calculated for any thermal stress distribution and at any specified time during cooldown for a ¼-thickness inside surface defect using the relationship:

$$K_{It} = (1.0359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3) * \sqrt{\pi a} \tag{2.7-4}$$

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

$$K_{It} = (1.043C_0 + 0.630C_1 + 0.481C_2 + 0.401C_3) * \sqrt{\pi a} \tag{2.7-5}$$

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

$$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3 \tag{2.7-6}$$

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

Once K_{Ia} (As calculated via Equation 2.5-1) is known, the pressure can be solved using Equation 2.5-3 with the newly calculated K_{It} and new equation for K_{IM}.

$$C * [M_m \times (pR_i / t)] + K_{It} < K_{Ia}$$

where:

C = 2.0 for Level A and Level B service limits (for heatup and cooldown),

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

This results in a pressure equation as follows:

$$p = \frac{[K_{It} - K_{Ia}]}{C * M_m * (R_i / t)} \tag{2.7-7}$$

Note that K_{It} is equal to zero for steady state and hydrostatic leak test conditions. In addition, K_{Ia} and K_{It} must be calculated individually for inside and outside flaw locations (i.e., the ¼T and ¾T wall locations) and the minimum pressure must be used from these two locations. [Note: K_{Ia} for ¼ T steady state is not the same as K_{Ia} for ¼T thermal conditions since the wall temperature is equal to the water temperature in steady state, but is not the case under thermal conditions.]

2.8 CODE CASES N-640 FOR KIC AND N-588 FOR CIRCUMFERENTIAL WELD FLAWS

2.8.1 ASME Code Case N-640

In February of 1999, the ASME Code approved Code Case N-640 which allows the use of the reference fracture toughness curve K_{Ic}, as found in Appendix A of Section XI, in lieu of Figure G-2110-1 in Appendix G for the development of pressure-temperature limit curves. (This is also described in Section 2.5 herein). Thus, when developing pressure-temperature limit curves, it is acceptable to calculate the reference stress intensity via Equation 2.5-2, in lieu of Equation 2.5-1. In addition, the K_{Ic} can be substituted for K_{Ia} in Equations 2.5-3, 2.6.2-1, 2.6.3-3, 2.6.4-1, 2.6.4-2, 2.6.5-1 and 2.7-7.

2.8.2 ASME Code Case N-588

In 1997, ASME Section XI, Appendix G was revised to add a methodology for the use of circumferential flaws when considering circumferential welds in developing pressure-temperature limit curves. This change was also implemented in a separate Code Case, N-588.

The original ASME Section XI, Appendix G approach mandated the postulation of an axial flaw in circumferential welds for the purposes of calculating pressure-temperature limits. Postulating the Appendix G reference flaw in a circumferential weld is physically unrealistic because the length of the reference flaw is 1.5 times the vessel thickness and is much longer than the width of the vessel girth welds. In addition, historical experience, with repair weld indications found during pre-service inspection and data taken from destructive examination of actual vessel welds, confirms that any flaws are small, laminar in nature and are not oriented transverse to the weld bead orientation. Because of this, any defects potentially introduced during fabrication process (and not detected during subsequent non-destructive examinations) should only be oriented along the direction of the weld fabrication. Thus, for circumferential welds, any postulated defect should be in the circumferential orientation.

The revision to Section XI, Appendix G now eliminates additional conservatism in the assumed flaw orientation for circumferential welds. The following revisions were made to ASME Section XI, Appendix G:

G-2214.1 Membrane Tension…

The K_I corresponding to membrane tension for the postulated circumferential defect of G-2120 is

$$K_{IM} = M_m \text{ x } (PR_i/t)$$

Where, M_m for an inside surface flaw is given by:

$$M_m = 0.89 \text{ for } \sqrt{t} < 2,$$

$$M_m = 0.443 \sqrt{t} \text{ for } 2 \leq \sqrt{t} \leq 3.464,$$

$$M_m = 1.53 \text{ for } \sqrt{t} > 3.464$$

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

$$M_m = 0.89 \text{ for } \sqrt{t} < 2,$$

$$M_m = 0.443 \sqrt{t} \text{ for } 2 \leq \sqrt{t} \leq 3.464,$$

$$M_m = 1.53 \text{ for } \sqrt{t} > 3.464$$

Note, that the only change relative to the OPERLIM computer code was the addition of the constants for M_m in a circumferential weld limited condition. No other changes were made to the OPERLIM computer code with regard to P-T calculation methodology.

2.9 CLOSURE HEAD/VESSEL FLANGE REQUIREMENTS

10 CFR Part 50, Appendix G contains the requirements for the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure flange regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the pre-service hydrostatic test pressure (3106 psig), which is 621 psig for a typical Westinghouse reactor vessel design.

This requirement was originally based on concerns about the fracture margin in the closure flange region. During the boltup process, stresses in this region typically reach over 70 percent of the steady-state stress, without being at steady-state temperature. The margin of 120°F and the pressure limitation of 20 percent of hydrotest pressure were developed using the K_{Ia} fracture toughness, in the mid 1970s.

Improved knowledge of fracture toughness and other issues which affect the integrity of the reactor vessel have led to the recent change to allow the use of K_{Ic} in the development of pressure-temperature curves, as contained in Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1."

The discussion given in WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants," concluded that the integrity of the closure head/vessel flange region is not a concern for any of the operating plants using the K_{Ic} toughness. Furthermore, there are no known mechanisms of degradation for this region, other than fatigue. The calculated design fatigue usage for this region is less than 0.1, so it may be concluded that flaws are unlikely to initiate in this region. It is therefore clear that no additional boltup requirements are necessary, and therefore the requirement of 10 CFR Part 50, Appendix G, can be eliminated from the Pressure-Temperature Curves, once the requirements of 10CFR50 Appendix G are changed. However, until 10CFR50 Appendix G is revised to eliminate the flange requirement, it must be included in the P-T limits, unless an exemption request is submitted and approved by the NRC.

2.10 MINIMUM BOLTUP TEMPERATURE

The minimum boltup temperature is equal to the material RT_{NDT} of the stressed region. The RT_{NDT} is calculated in accordance with the methods described in Branch Technical Position MTEB 5-2. The Westinghouse position is that the minimum boltup temperature be no lower than 60°F. Thus, the minimum boltup temperature should be 60°F or the material RT_{NDT} whichever is higher.

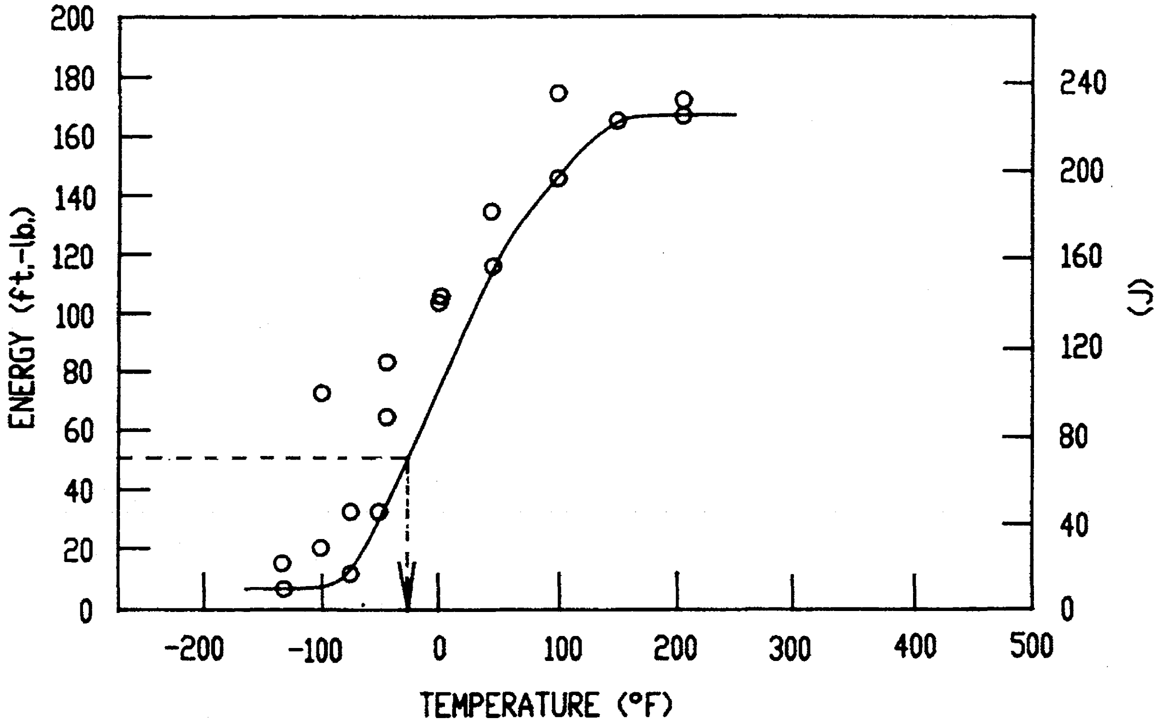


Figure 2.1 Example of a Charpy Impact Energy Curve Used to Determine IRT_{NDT} (Note: 35 mils lateral expansion is required at indicated temperature)

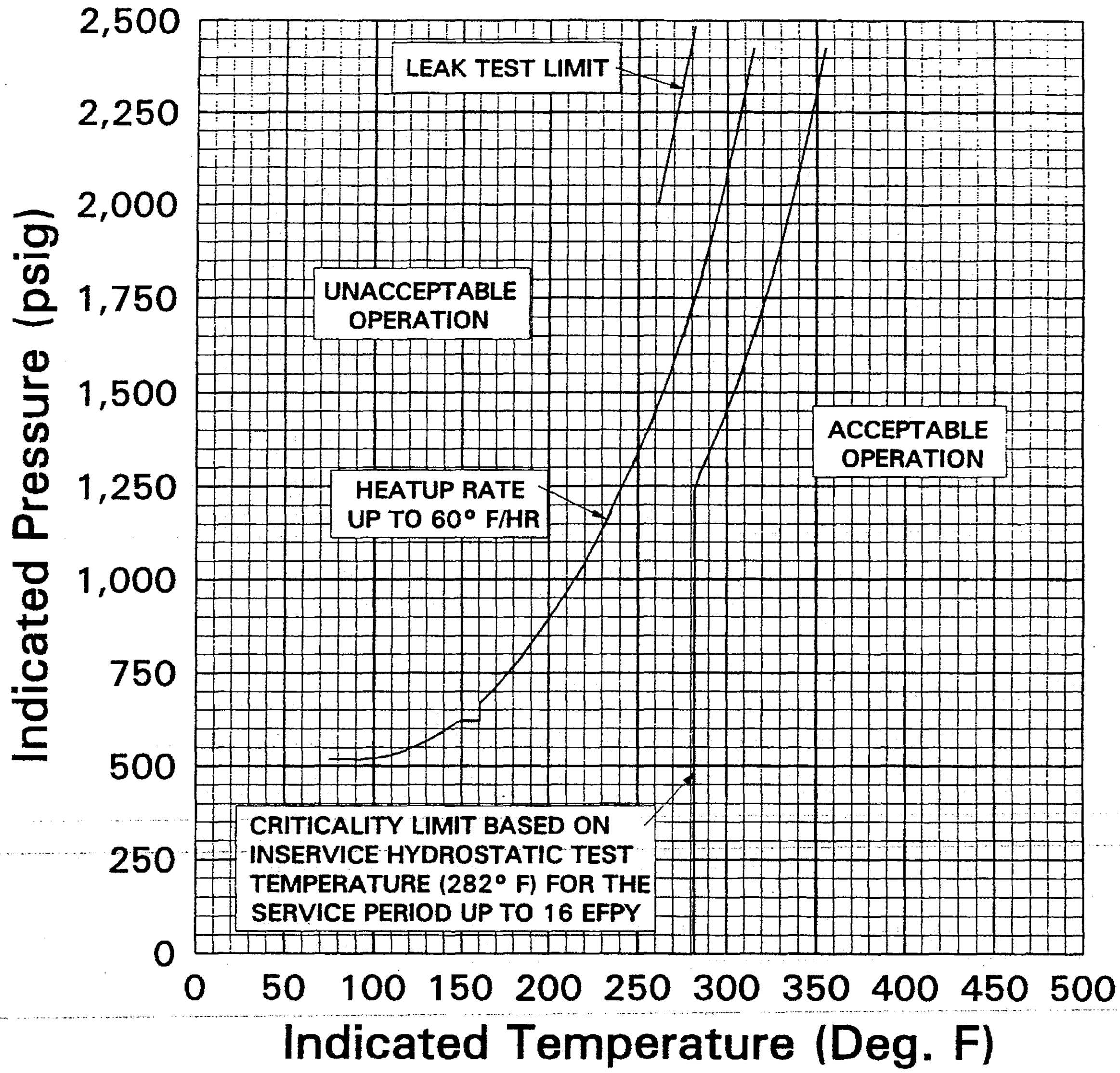


Figure 2.2 Heatup Pressure-Temperature Limit Curve For Heatup Rates up to 60°F/Hr

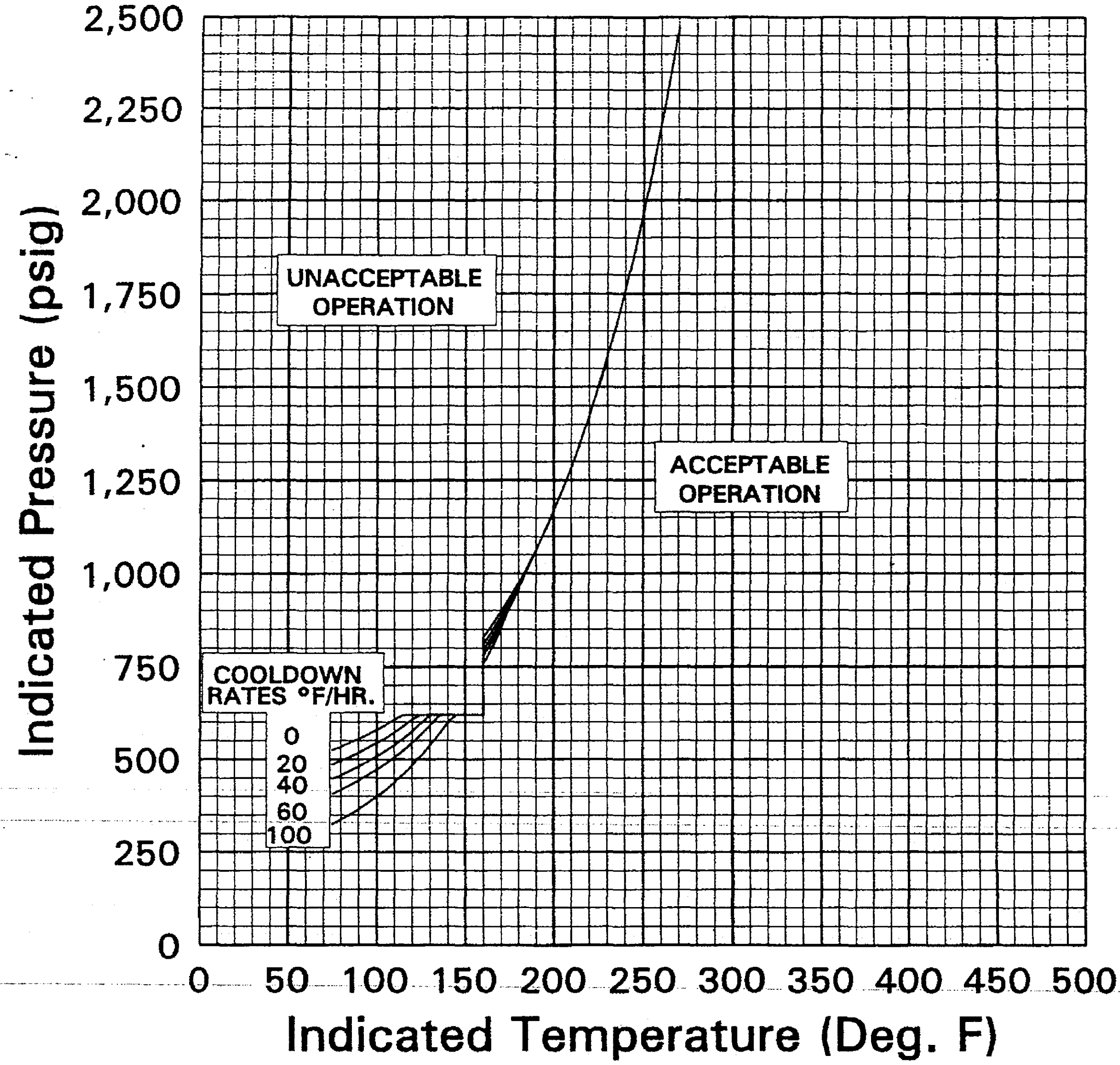


Figure 2.3 Cooldown Pressure-Temperature Limit Curves or Cooldown Rates up to 100°F/Hr

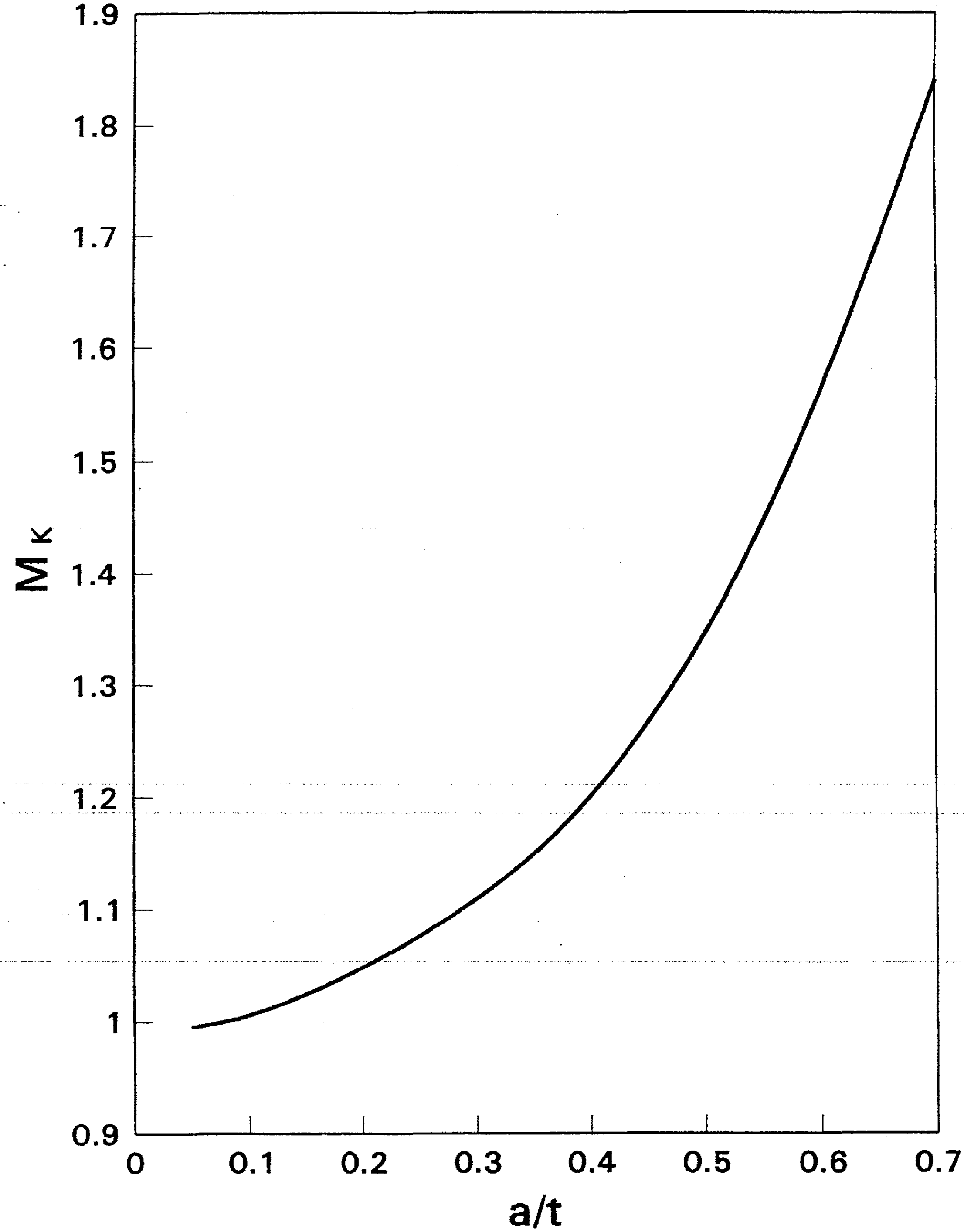


Figure 2.4 Membrane Stress Correction Factor (M_K) vs. a/t Ratio for Flaws Having Length to Depth Ratio of 6 (Welding Research Bulletin 175 Method)

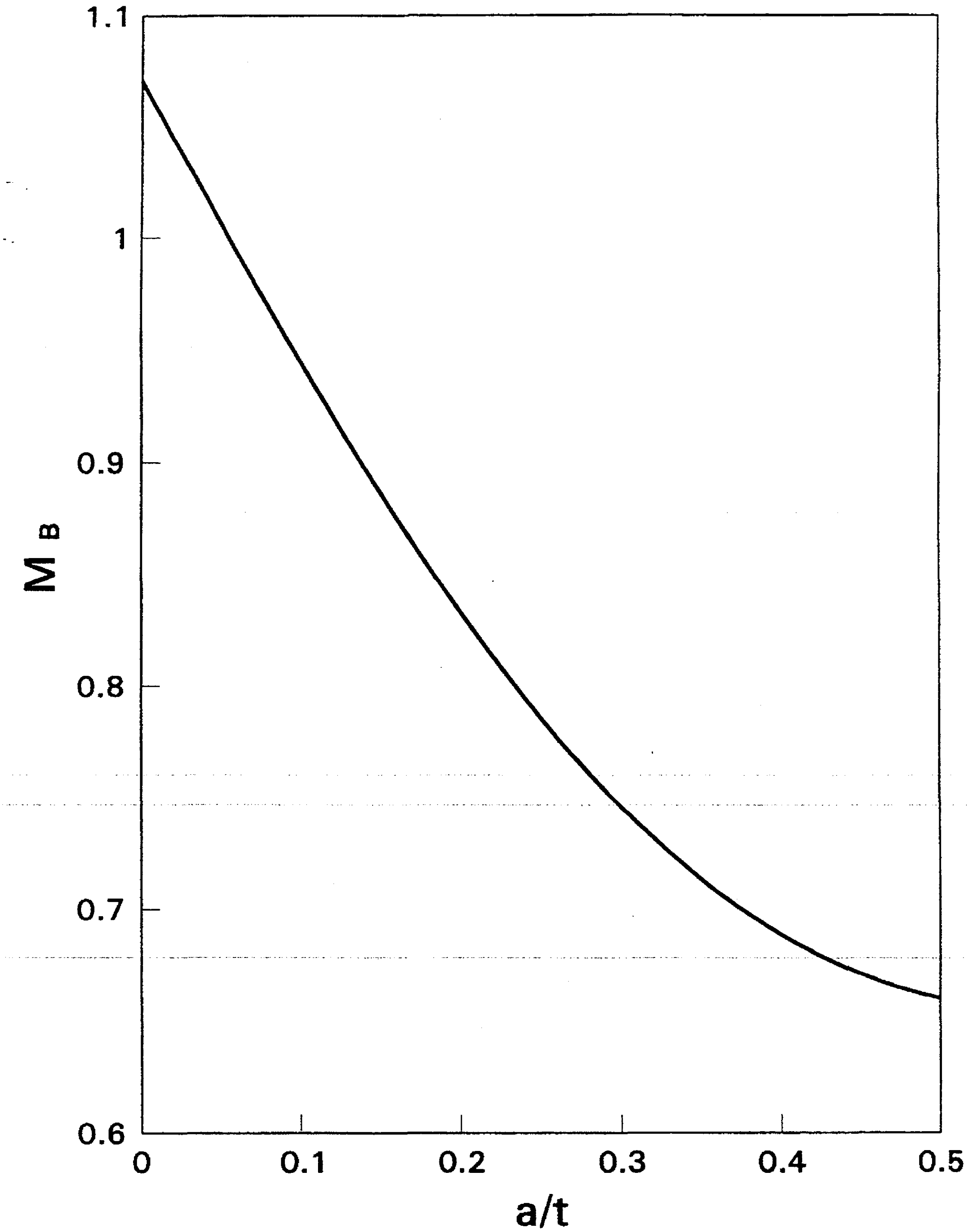


Figure 2.5 Bending Stress Correction Factor (M_B) vs. a/t Ratio for Flaws Having Length to Depth Ratio of 6 (Welding Research Bulletin 175 Method)

3.0 COLD OVERPRESSURE MITIGATING SYSTEM (COMS)

3.1 INTRODUCTION

The purpose of the COMS is to supplement the normal plant operational administrative controls and the water relief valves in the Residual Heat Removal System (RHRS) when they are unavailable to protect the reactor vessel from being exposed to conditions of fast propagating brittle fracture. This has been achieved by conservatively choosing COMS setpoints which prevent exceeding the pressure/temperature limits established by 10 CFR Part 50 Appendix G[4] requirements. The COMS is designed to provide the capability, during relatively low temperature operation (typically less than 350°F), to automatically prevent the RCS pressure from exceeding the applicable limits. Once the system is enabled, no operator action is involved for the COMS to perform its intended pressure mitigation function. Thus, no operator action is modelled in the analyses supporting the setpoint selection, although operator action may be initiated to ultimately terminate the cause of the overpressure event.

The PORVs located near the top of the pressurizer, together with additional actuation logic from the wide-range pressure channels, are utilized to mitigate potential RCS overpressure transients defined below if the RHRS water relief valves are inadvertently isolated from the RCS. The COMS provides the supplemental relief capacity for specific transients which would not be mitigated by the RHRS relief valves. In addition, a limit on the PORV piping is accommodated due to the potential for water hammer effects to be developed in the piping associated with these valves as a result of the cyclic opening and closing characteristics during mitigation of an overpressure transient. Thus, a pressure limit more restrictive than the 10CFR50, Appendix G[4] allowable is imposed above a certain temperature so that the loads on the piping from a COMS event would not affect the piping integrity.

Two specific transients have been defined, with the RCS in a water-solid condition, as the design basis for COMS. Each of these scenarios assumes as an initial condition that the RHRS is isolated from the RCS, and thus the relief capability of the RHRS relief valves is not available. The first transient consists of a heat injection scenario in which a reactor coolant pump in a single loop is started with the RCS temperature as much as 50°F lower than the steam generator secondary side temperature and the RHRS has been inadvertently isolated. This results in a sudden heat input to a water-solid RCS from the steam generators, creating an increasing pressure transient. The second transient has been defined as a mass injection scenario into a water-solid RCS caused by the simultaneous isolation of the RHRS isolation of letdown and failure of the normal charging flow controls to the full flow condition. Various combinations of charging and safety injection flows may also be evaluated on a plant-specific basis; however, the mass injection transient used as a design basis should encompass the limiting pump(s) operability configuration permitted per the plant-specific Technical Specifications during the Modes when COMS is required to be in operation. The resulting mass injection/letdown mismatch causes an increasing pressure transient.

3.2 COMS SETPOINT DETERMINATION

Westinghouse has developed the following methodology which is employed to determine PORV setpoints for mitigation of the COMS design basis cold overpressurization transients. This methodology maximizes the available operating margin for setpoint selection while maintaining an appropriate level of protection in support of reactor vessel integrity.

3.2.1 Parameters Considered

The selection of proper COMS setpoints for actuating the PORVs requires the consideration of numerous system parameters including:

a. Volume of reactor coolant involved in transient

b. RCS pressure signal transmission delay

c. Volumetric capacity of the relief valves versus opening position

d. Stroke time of the relief valves (open & close)

e. Initial temperature and pressure of the RCS

f. Mass input rate into RCS

g. Temperature of injected fluid

h. Heat transfer characteristics of the steam generators

i. Initial temperature asymmetry between RCS and steam generator secondary water

j. Mass of steam generator secondary water

k. RCP startup dynamics

l. 10CFR50, Appendix G pressure/temperature characteristics of the reactor vessel

m. Pressurizer PORV piping/structural analysis limitations

n. Dynamic and static pressure difference between reactor vessel midplane and location of wide range pressure transmitter

These parameters are input to a specialized version of the LOFTRAN computer code which calculates the maximum and minimum system pressures.

3.2.2 Pressure Limits Selection

The function of the COMS is to protect the reactor vessel from fast propagating brittle fracture. This has been implemented by choosing COMS setpoints which prevent exceeding the limits prescribed by the applicable pressure/temperature characteristic for the specific reactor vessel material in accordance with rules given in Appendix G to 10CFR50[(4)]. The COMS design basis takes credit for the fact that overpressure events most likely occur during isothermal conditions in the RCS. Therefore, it is appropriate to utilize the steady-state Appendix G limit. In addition, the COMS also provides for an operational consideration to maintain the integrity of the PORV piping. A typical characteristic 10CFR50

Appendix G curve is shown by Figure 3.1 where the allowable system pressure increases with increasing temperature. This type of curve sets the nominal upper limit on the pressure which should not be exceeded during RCS increasing pressure transients based on reactor vessel material properties. Superimposed on this curve is the PORV piping limit which is conservatively used, for setpoint development, as the maximum allowable pressure above the temperature at which it intersects with the 10CFR50 Appendix G curve.

When a relief valve is actuated to mitigate an increasing pressure transient, the release of a volume of coolant through the valve will cause the pressure increase to be slowed and reversed as described by Figure 3.2. The system pressure then decreases, as the relief valve releases coolant, until a reset pressure is reached where the valve is signalled to close. Note that the pressure continues to decrease below the reset pressure as the valve recloses. The nominal lower limit on the pressure during the transient is typically established based solely on an operational consideration for the reactor coolant pump #1 seal to maintain a nominal differential pressure across the seal faces for proper film-riding performance.

The nominal upper limit (based on the minimum of the steady-state 10CFR50 Appendix G requirement and the PORV piping limitations) and the nominal RCP #1 seal performance criteria create a pressure range from which the setpoints for both PORVs may be selected as shown on Figures 3.3 and 3.4.

Where there is insufficient range between the upper and lower pressure limits to select PORV setpoints to provide protection against violation of both limits, setpoint selection to provide protection against the upper pressure limit violation shall take precedence.

3.2.3 Mass Input Consideration

For a particular mass input transient to the RCS, the relief valve will be signalled to open at a specific pressure setpoint. However, as shown on Figure 3.2, there will be a pressure overshoot during the delay time before the valve starts to move and during the time the valve is moving to the full open position. This overshoot is dependent on the dynamics of the system and the input parameters, and results in a maximum system pressure somewhat higher than the set pressure. Similarly there will be a pressure undershoot, while the valve is relieving, both due to the reset pressure being below the setpoint and to the delay in stroking the valve closed. The maximum and minimum pressures reached (P_{MAX} and P_{MIN}) in the transient are a function of the selected setpoint (P_s) as shown on Figure 3.3. The shaded area represents an optimum range from which to select the setpoint based on the particular mass input case. Several mass input cases may be run at various input flow rates to bound the allowable setpoint range.

3.2.4 Heat Input Consideration

The heat input case is done similarly to the mass input case except that the locus of transient pressure values versus selected setpoints may be determined for several values of the initial RCS temperature. This heat input evaluation provides a range of acceptable setpoints dependent on the reactor coolant temperature, whereas the mass input case is limited to the most restrictive low temperature condition only (i.e., the mass injection transient is not sensitive to temperature). The shaded area on Figure 3.4 describes the acceptable band for a heat input transient from which to select the setpoint for a particular initial reactor coolant temperature.

3.2.5 Final Setpoint Selection

By superimposing the results of multiple mass input and heat input cases evaluated, (from a series of figures such as 3.3 and 3.4) a range of allowable PORV setpoints to satisfy both conditions can be determined. Each of the two PORVs may have a different pressure setpoint versus temperature specification such that only one valve will open at a time and mitigate the transient (i.e., staggered setpoints). The second valve operates only if the first fails to open on command. This design supports a single failure assumption as well as minimizing the potential for both PORVs to open simultaneously, a condition which may create excessive pressure undershoot and challenge the RCP #1 seal performance criteria. However, each of the sets of staggered setpoints must result in the system pressure staying below the P_{MAX} pressure limit shown on Figures 3.3 and 3.4 when either valve is utilized to mitigate the transient.

The function generator used to program the pressure versus setpoint curves for each valve has a limited number of programmable break points (typically 9). These are strategically defined in the final selection process, with consideration given to the slope of any line segment, which is limited to approximately 24 psi/°F.

The selection of the setpoints for the PORVs considers the use of nominal upper and lower pressure limits. The upper limits are specified by the minimum of the steady-state cooldown curve as calculated in accordance with Appendix G to 10CFR50[4] or the peak RCS pressure based upon piping/structural analysis loads. The lower pressure extreme is specified by the reactor coolant pump #1 seal minimum differential pressure performance criteria. The upper pressure limits are already based on conservative assumptions (such as a safety factor of 2 on pressure stress, use of a lower bound K_{IR} curve and an assumed 1/4T flaw depth with a length equal to 1 1/2 times the vessel wall thickness) as discussed in section 2 of this report. However, uncertainties associated with instrumentation utilized by COMS will be determined using a process described by ISA Standard S67.04-1994. These uncertainties will be accounted for in the selection of COMS PORV setpoints.

While the RHR relief valves also provide overpressure protection for certain transients, these transients are not the same as the design basis transients for COMS. The RHR relief valve design basis precedes the development of the COMS design basis, and therefore the RHR relief valves may not provide protection against the COMS design basis events. The design basis described herein should be considered as applicable only when the pressurizer PORVs are used for COMS.

3.3 APPLICATION OF ASME CODE CASE N-514

ASME Code Case N-514[17] allows low temperature overpressure protection systems (LTOPS, as the code case refers to COMS) to limit the maximum pressure in the reactor vessel to 110% of the pressure determined to satisfy Appendix G, paragraph G-2215, of Section XI of the ASME Code[5]. (Note, that the setpoint selection methodology as discussed in Section 3.2.5 specifically utilizes the steady-state curve.) The application of ASME Code Case N-514 increases the operating margin in the region of the pressure-temperature limit curves where the COMS system is enabled. Code Case N-514 requires LTOPS to be effective at coolant temperatures less than 200°F or at coolant temperatures corresponding to a reactor vessel metal temperature less than RT_{NDT} + 50°F, whichever is greater. RT_{NDT} is the highest adjusted

reference temperature for weld or base metal in the beltline region at a distance one-fourth of the vessel section thickness from the vessel inside surface, as determined by Regulatory Guide 1.99, Revision 2.

3.4 ENABLE TEMPERATURE FOR COMS

The enable temperature is the temperature below which the COMS system is required to be operable. The definition of the enabling temperature currently approved and supported by the NRC is described in Branch Technical Position RSB 5-2[18]. This position defines the enable temperature for LTOP systems as the water temperature corresponding to a metal temperature of at least RT_{NDT} + 90°F at the beltline location (1/4t or 3/4t) that is controlling in the Appendix G limit calculations. This definition is very conservative, and is mostly based on material properties and fracture mechanics, with the understanding that material temperatures of RT_{NDT} + 90°F at the critical location will be well up the transition curve from brittle to ductile properties, and therefore brittle fracture of the vessel is not expected.

The ASME Code Case N-514 supports an enable temperature of RT_{NDT} + 50°F or 200°F, whichever is greater as described in Section 3.3.

A significant improvement in the enable temperature can be obtained by application of code case N641. This code case incorporates the benefits of code cases N588, and N640. The resulting enable temperatures for the Westinghouse designs obtained using code case N641 are listed below.

Vessel Type	Axial Flaw	Circumferential Flaw
2 – loop	RT_{NDT} + 23F	Any temperature
3 – loop	RT_{NDT} + 30F	RT_{NDT} – 174F
4 – loop	RT_{NDT} + 34F	RT_{NDT} – 110F

The RCS cold leg temperature limitation for starting an RCP is the same value as the COMS enable temperature to ensure that the basis of the heat injection transient is not violated. The Standard Technical Specifications (STS) prohibit starting an RCP when any RCS cold leg temperatures is less than or equal to the COMS enable temperature unless the secondary side water temperature of each steam generator is less than or equal to 50°F above each of the RCS cold leg temperatures.

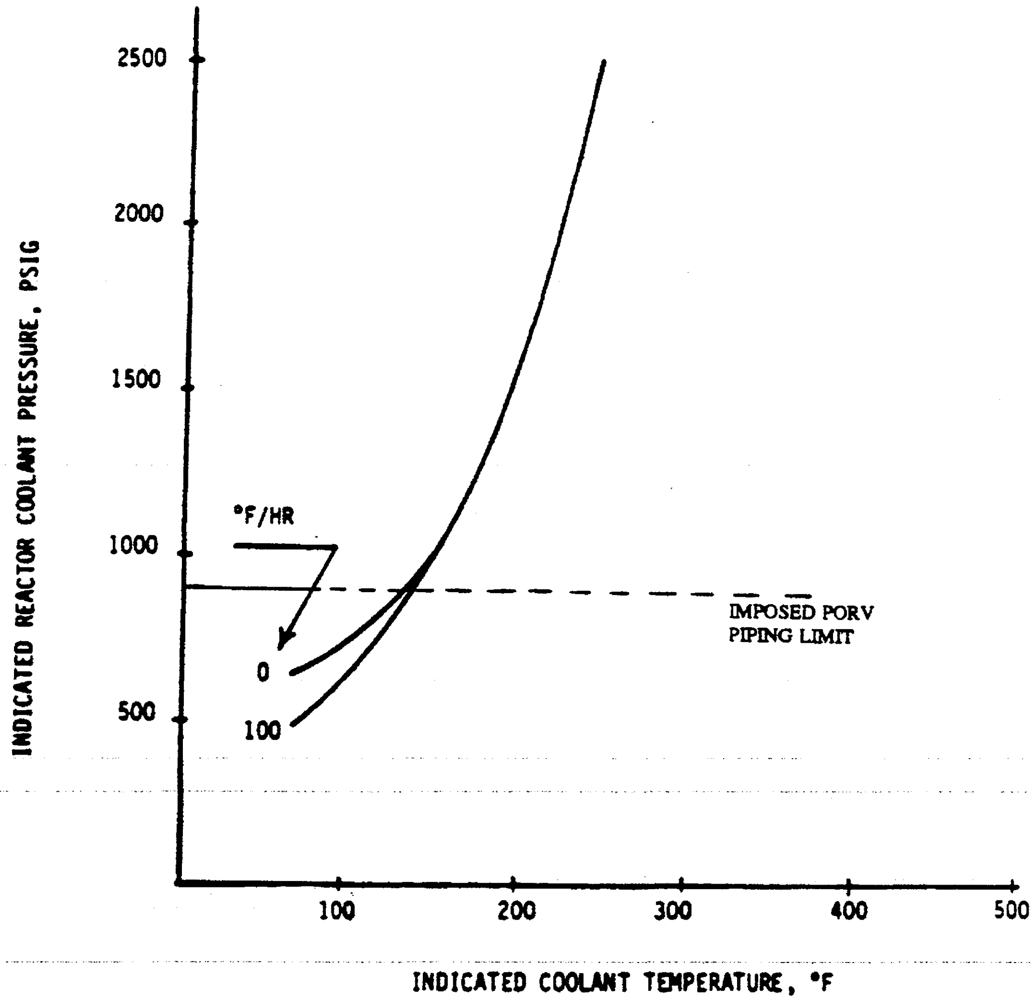


Figure 3.1 Typical Appendix G P/T Characteristics

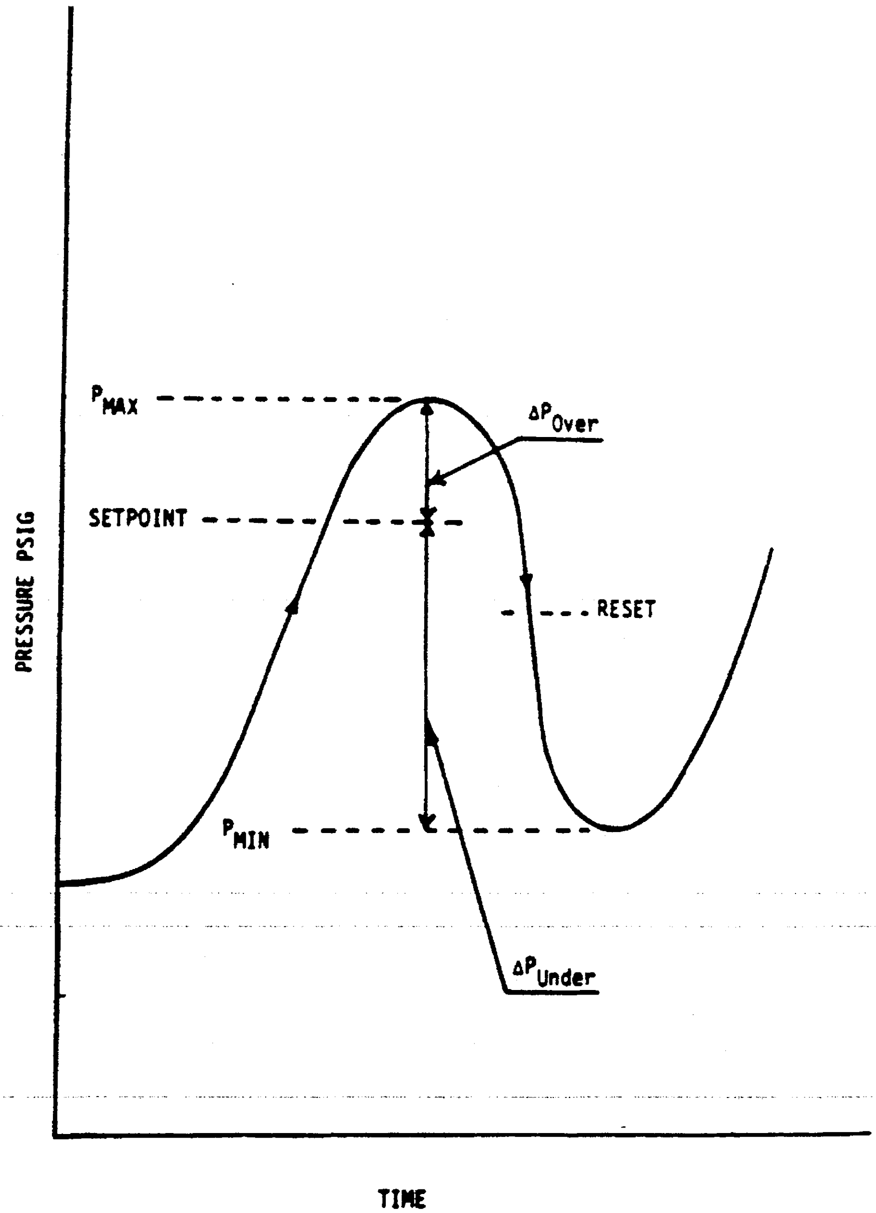
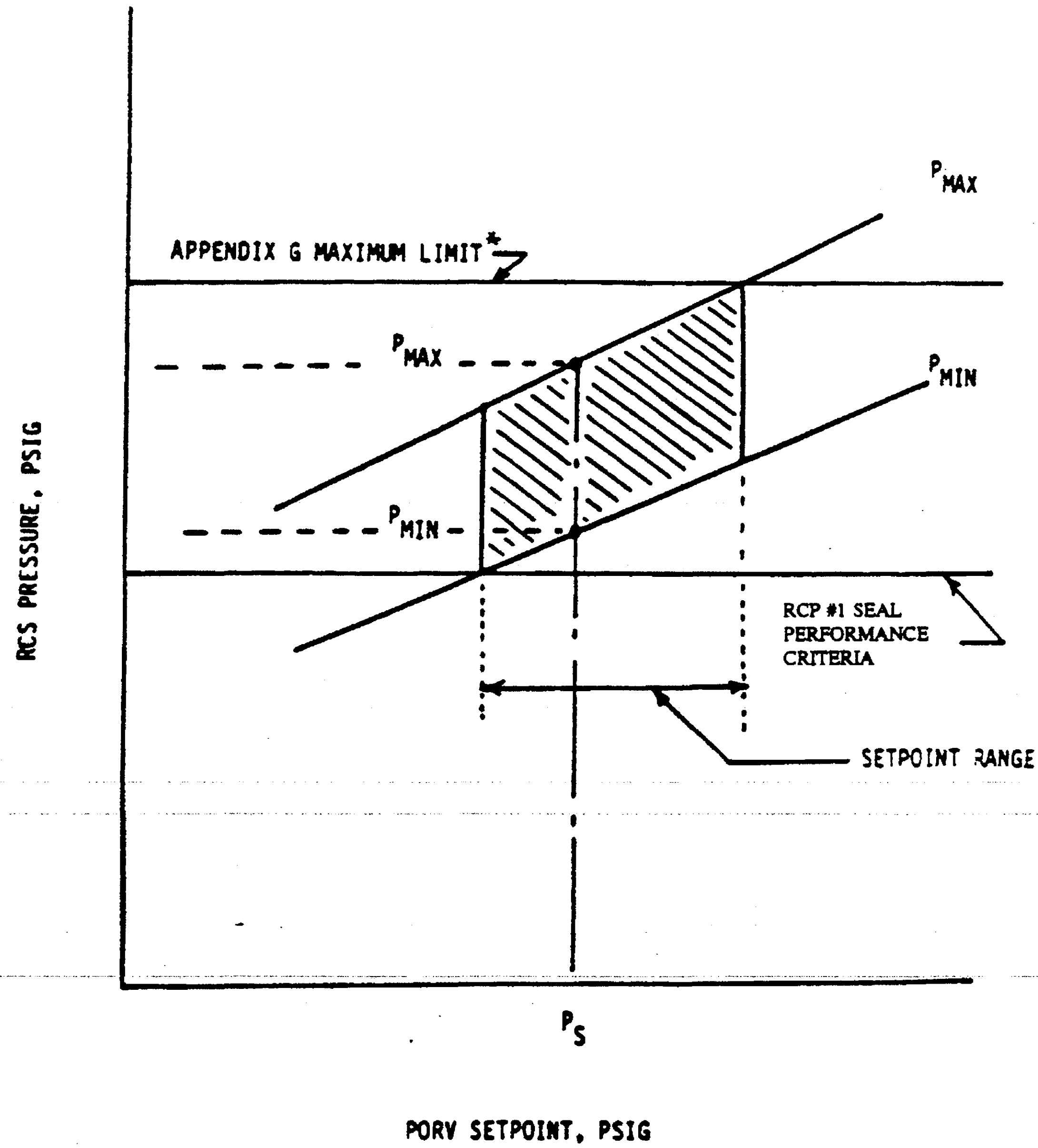
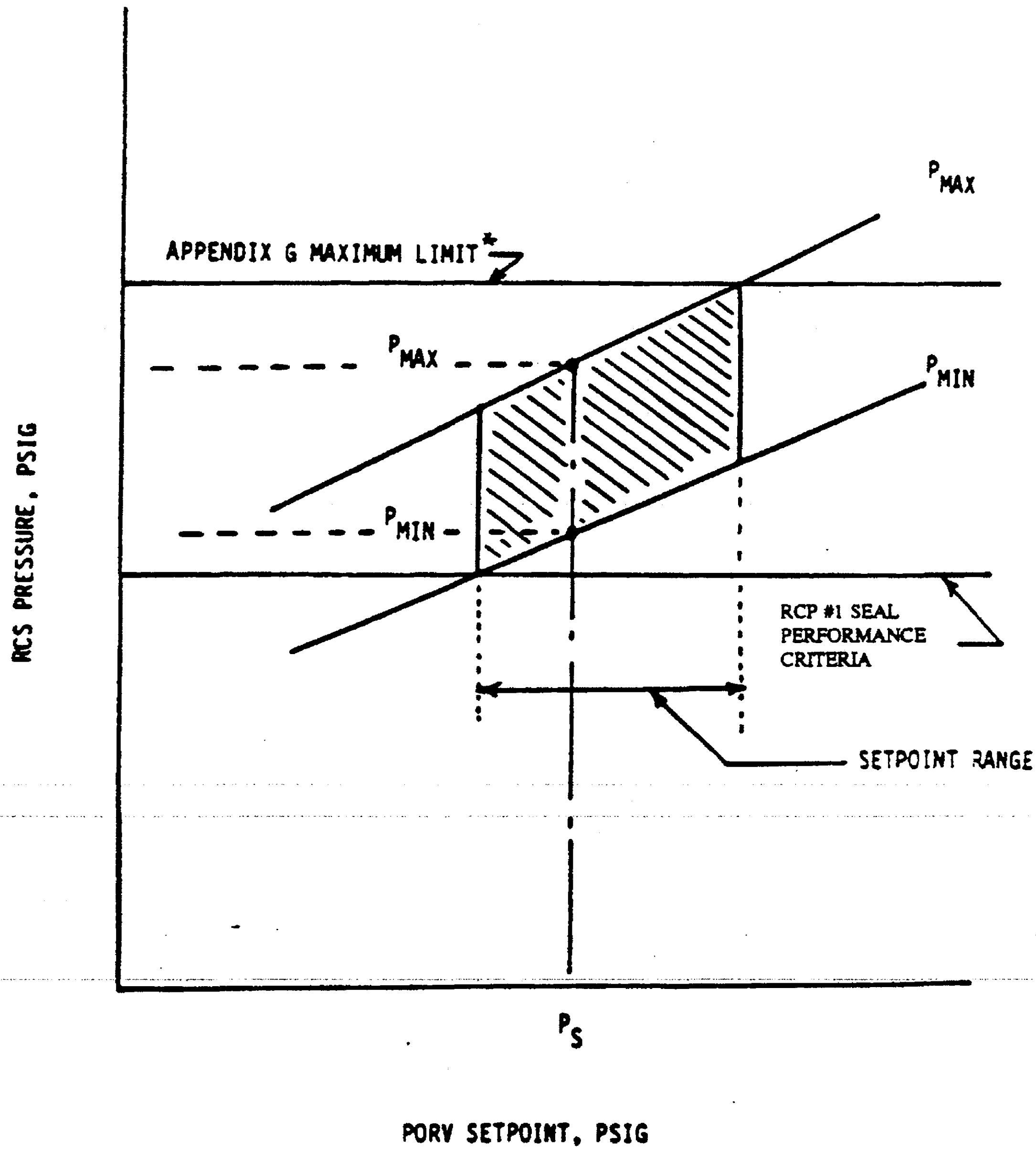


Figure 3.2 Typical Pressure Transient (1 Relief Valve Cycle)



* The maximum pressure limit is the minimum of the Appendix G limit or the PORV discharge piping structural analysis limit.

Figure 3.3 Setpoint Determination (Mass Input)



* The maximum pressure limit is the minimum of the Appendix G limit or the PORV discharge piping structural analysis limit.

Figure 3.4 Setpoint Determination (Heat Input)

4.0 REFERENCES

1. NUREG 1431, "Standard Technical Specifications for Westinghouse Plants," Revision 2.

2. U.S. Nuclear Regulatory Commission, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," Generic Letter 88-16, October, 1988.

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

4. Code of Federal Regulations, Title 10, Part 50, "Fracture Toughness Requirements for Light-Water Nuclear Power Reactors," Appendix G, Fracture Toughness Requirements.

5. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Appendix G, Fracture Toughness Criteria For Protection Against Failure.

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

7. RSICC Computer Code Collection CCC-650, "DOORS 3.1, One-, Two-, and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System," August 1996.

8. ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Power Plant Components," Division 1, Subsection NB: Class 1 Components.

9. Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements," NUREG-0800 Standard Review Plan 5.3.2, Pressure-Temperature Limits, July 1981, Rev. 1.

10. ASTM E-208, Standard Test Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels, ASTM Standards, Section 3, American Society for Testing and Materials.

11. RSICC Data Library Collection DLC-185, "BUGLE-96, Coupled 47 Neutron, 20 Gamma-Ray Group Cross-Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications," March 1996.

12. I. Remec and F. B. K. Kam, "Pool Critical Assembly Pressure Vessel Benchmark," NUREG/CR-6454 (ORNL/TM-13205), July 1997.

13. Code of Federal Regulations, Title 10, Part 50, "Fracture Toughness Requirements for Light-Water Nuclear Power Reactors," Appendix H, Reactor Vessel Material Surveillance Program Requirements.

14. Timoshenko, S. P. and Goodier, J. N., Theory of Elasticity, Third Edition, McGraw-Hill Book Co., New York, 1970.

15. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Appendix A, Analysis of Flaws, Article A3000, Method For K_I Determination.

16. WRC Bulletin No. 175, "PVRC Recommendations on Toughness Requirements for Ferritic Materials," Welding Research Council, New York, August 1972.

17. ASME Boiler and Pressure Vessel Code Case N-514, Section X1, Division 1, "Low Temperature Overpressure Protection," Approval date: February 12, 1992.

18. Branch Technical Position RSB 5-2, "Overpressurization Protection of Pressurized Water Reactors While Operating at Low Temperatures," NUREG-0800 Standard Review Plan 5.2.2, Overpressure Protection, November 1988, Rev. 2.

19. ASME Boiler and Pressure Vessel Code Case N640, Section XI, Division 1, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves," February 26, 1996.

20. ASME Boiler and Pressure Vessel Code Case N588, Section XI, Division 1, "Alternative to Reference Flow Orientation of Appendix G for Circumferential Welds in Reactor Vessels," December 12, 1997.

21. ASME Boiler and Pressure Vessel Code Case N641, Section XI, Division 1, "Alternative Pressure-Temperature Relationship and Low Temperature Overpressure Protection System Requirements," January 17, 2000.

22. I. Remec and F. B. K. Kam, "H. B. Robinson Pressure Vessel Benchmark," NUREG/CR-6453 (ORNL/TM-13204), February 1998.

APPENDIX A

RELEVANT ASME NUCLEAR CODE CASES

Table A-1	Status of ASME Nuclear Code Cases Associated with the P-T Limit Curve/COMS Methodology		
Code Case	**Title**	**Approved by ASME**	**Section XI of the ASME Code**
514	Low Temperature Overpressure Protection	2/12/92	1995 Edition through the 1996 Addenda
588	Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessel	12/12/97	1998 Edition through the 2000 Addenda
640	Alternative Reference Fracture Toughness for Development of P-T Limit Curves	2/26/99	1998 Edition through the 2000 Addenda
641	Alternative Pressure Temperature Relationship and Low Temperature Overpressure Protection System Requirement	1/17/00	1998 Edition through the 2000 Addenda

APPENDIX B

CORRESPONDENCE WITH THE NRC

Domestic Members
AmerenUE
Callaway
American Electric Power Co.
D.C. Cook 1 & 2
Arizona Public Service Co.
Palo Verde 1, 2 & 3
Constellation Energy Group
Calvert Cliffs 1 & 2
Dominion Nuclear Connecticut
Millstone 2 & 3
Dominion Virginia Power
North Anna 1 & 2
Surry 1 & 2
Duke Energy
Catawba 1 & 2
McGuire 1 & 2
Entergy Nuclear Northeast
Indian Point 2 & 3
Entergy Nuclear South
ANO 2
Waterford 3
Exelon Generation Company LLC
Braidwood 1 & 2
Byron 1 & 2
FirstEnergy Nuclear Operating Co.
Beaver Valley 1 & 2
FPL Group
St. Lucie 1 & 2
Seabrook
Turkey Point 3 & 4
Nuclear Management Co.
Kewaunee
Palisades
Point Beach 1 & 2
Prairie Island
Omaha Public Power District
Fort Calhoun
Pacific Gas & Electric Co.
Diablo Canyon 1 & 2
Progress Energy
H. B. Robinson 2
Shearon Harris
PSEG – Nuclear
Salem 1 & 2
Rochester Gas & Electric Co.
R. E. Ginna
South Carolina Electric & Gas Co.
V. C. Summer
Southern California Edison
SONGS 2 & 3
STP Nuclear Operating Co.
South Texas Project 1 & 2
Southern Nuclear Operating Co.
J. M. Farley 1 & 2
A. W. Vogtle 1 & 2
Tennessee Valley Authority
Sequoyah 1 & 2
Watts Bar 1
TXU Electric
Commanche Peak 1 & 2
Wolf Creek Nuclear Operating Corp.
Wolf Creek

International Members
Electrabel
Doel 1, 2, 4
Tihange 1 & 3
Electricité de France
Kansai Electric Power Co.
Mihama 1
Takahama 1
Ohi 1 & 2
Korea Hydro & Nuclear Power Co.
Kori 1 – 4
Ulchin 3 & 4
Yonggwang 1 - 5
British Energy plc
Sizewell B
NEK
Krško
Spanish Utilities
Asco 1 & 2
Vandellos 2
Almaraz 1 & 2
Ringhals AB
Ringhals 2 – 4
Taiwan Power Co.
Maanshan 1 & 2

WOG-04-086
February 18, 2004

WCAP-14040, Rev. 3
Project Number 694

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

Attention: Chief, Information Management Branch,
Division of Program Management

Subject: Westinghouse Owners Group
Transmittal of Comments on the Draft Safety Evaluation for WCAP-14040, Rev. 3, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" (MUHP-3073, TAC No. MB5754)

On February 2, 2004, the NRC provided a draft Safety Evaluation (SE) of WCAP-14040, Revision 3, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," to the Westinghouse Owners Group (WOG) for review and comment (Ref. 1). Enclosure 1 contains a mark-up of the suggested clarifications to the SE for consideration by the NRC. The WOG requests that the NRC issue a final Safety Evaluation by April 1, 2004.

If you require further information, feel free to contact Mr. Ken Vavrek, Owners Group Project Office at 412-374-4302.

Sincerely,

Frederick. P. "Ted" Schiffley, II, Chairman
Westinghouse Owners Group

Enclosure

WOG-04-086
February 18, 2004

cc: WOG Steering Committee
WOG Management Committee
WOG Licensing Subcommittee
WOG Materials Subcommittee
D. Holland, USNRC OWFN 07 E1 (1L, 1A) (via Federal Express)
S. Dinsmore, USNRC (1L, 1A) OWFN 10H4
Project Management Office
J. Gresham
J. D. Andrachek
W.H. Bamford
T.J. Laubham
J. Perock

Reference:

1. NRC Letter, S. Dembeck (NRC) to G. Bischoff (Westinghouse), "Draft Safety Evaluation of Topical Report WCAP-14040, Revision 3, 'Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,' (TAC No. MB5754)," February 2, 2004.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DRAFT SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

WCAP-14040, REVISION 3, "METHODOLOGY USED TO DEVELOP COLD

OVERPRESSURE MITIGATING SYSTEM SETPOINTS AND REACTOR COOLANT

SYSTEM HEATUP AND COOLDOWN LIMIT CURVES"

WESTINGHOUSE OWNERS GROUP

PROJECT NO. 694

1.0 INTRODUCTION

By letter dated May 23, 2002, the Westinghouse Owners Group (WOG) submitted Topical Report (TR) WCAP-14040, Revision 3, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," for NRC staff review and approval. This TR was developed to define a methodology for reactor pressure vessel (RPV) pressure-temperature (P-T) limit curve development and, consistent with the guidance provided in NRC Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," for the development of plant-specific Pressure-Temperature Limit Reports (PTLRs). A prior revision, WCAP-14040, Revision 2, had been approved as a PTLR methodology by the NRC staff's safety evaluation dated October 16, 1995. WCAP-14040, Revision 3, was submitted for NRC staff approval to reflect recent changes in the WOG methodology. Given the scope of the changes incorporated in WCAP-14040, Revision 3, and a significant amount of rewriting which was done to improve clarity of some sections, the NRC staff reviewed the TR in its entirety. Based on questions posed by the NRC staff necessitating clarification of statements or editorial changes, the WOG revised WCAP-14040, Revision 3, and submitted ~~the revised~~ revisions to the TR for NRC staff review and approval by letter dated October 20, 2003.

2.0 REGULATORY EVALUATION

Four specific topics are addressed in the context of the development of a PTLR methodology: (1) the calculation of neutron fluences for the RPV and RPV surveillance capsules; (2) the evaluation of RPV material properties due to changes caused by neutron radiation; (3) the development of appropriate P-T limit curves based on these RPV material properties and the establishment of cold overpressure mitigating system (COMS) setpoints to protect the RPV from brittle failure; and (4) the development of an RPV material surveillance program to monitor changes in RPV material properties due to radiation. Regulatory requirements related to the four topics noted above are addressed in Appendices G and H to Title 10 of the Code of Federal Regulations Part 50 (10 CFR Part 50). Appendix G to 10 CFR Part 50 provides requirements related to RPV P-T limit development and directly or indirectly addresses topics (1) through (3) above. Appendix H to 10 CFR Part 50 defines regulatory requirements related to RPV material surveillance programs and addresses topic (4) above.

For the staff's review of WCAP-14040, Revision 3, several additional guidance documents were used. NRC Standard Review Plan (SRP) Sections 5.2.2, "Overpressure Protection," 5.3.1, "Reactor Vessel Materials," and 5.3.2, "Pressure-Temperature Limits," provide specific review guidance related to RPV material property determination, P-T limit development, and COMS performance. Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," describes analysis procedures acceptable to the NRC staff for the purpose of assessing RPV material property changes due to radiation. RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," addresses NRC staff expectations for an acceptable fluence calculation methodology. American Society for Testing and Materials (ASTM) Standard Practice E 185, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," provides guidance on the establishment of RPV material surveillance programs and editions of ASTM E 185 are incorporated by reference into Appendix H to 10 CFR Part 50. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI, Appendix G provides specific requirements regarding the development of P-T limit curves.

Finally, specific guidance regarding topics and the level of detail to which they must be addressed as part of an acceptable PTLR methodology is given in GL 96-03.

3.0 TECHNICAL EVALUATION

The technical requirements to be addressed in an acceptable PTLR methodology are provided under the column heading "Minimum Requirements to be Included in Methodology" in the table entitled "Requirements for Methodology and PTLR" in Attachment 1 to GL 96-03. Summarized versions of the seven requirements are given below, along with the staff's technical evaluation of information in WCAP-14040, Revision 3, related to each requirement.

Requirement 1: Regarding the reactor vessel material surveillance program, the methodology should briefly describe the surveillance program. The methodology should clearly reference the requirements of Appendix H to 10 CFR Part 50.

The provisions of the methodology described in WCAP-14040, Revision 3, do not specify how the plant-specific RPV surveillance programs should be maintained in order to be in compliance with Appendix H to 10 CFR Part 50. Licensees who wish to use WCAP-14040, Revision 3, as their PTLR methodology must submit additional information to address the methodology requirements in GL 96-03 related to RPV material surveillance program ~~issues.~~
discussed the

Requirement 2: Regarding the calculation of RPV materials' adjusted reference temperatures (ART) values, the methodology should describe the method for calculating material ART values using RG 1.99, Revision 2.

provision 2 in the table of Attachment 1 to

Information regarding how material ARTs are to be determined within the WCAP-14040, Revision 3, PTLR methodology is provided in Section 2.3 and 2.4 of the TR. In Section 2.3, the determination of initial, unirradiated material properties from Charpy V-notch impact tests and/or nil-ductility drop weight tests is clearly defined. The methodology specified in Section 2.3 accurately incorporates the guidance found in ASME Code Section III, paragraph NB-2331 and additional information in SRP Section 5.3.1.

-3- No comments on pages 3, 4 and 5

In Section 2.4 of the TR, the determination of changes in material properties due to irradiation is addressed, along with the determination of margins necessary to account for uncertainties in initial properties and irradiation damage assessment. The methodology specified in Section 2.4 accurately incorporates the guidance found in RG 1.99, Revision 2.

The NRC staff, therefore, determined that the methodology described for determining material ART values in WCAP-14040, Revision 3, was consistent with the guidance provided in the ASME Code, SRP Section 5.3.1, and RG 1.99, Revision 2, and was, therefore, acceptable.

Requirement 3: Regarding the development of RPV P-T limit curves, the methodology should describe the application of fracture mechanics-based calculations in constructing P-T limit curves based on the provisions of Appendix G to Section XI of the ASME Code and SRP Section 5.3.2.

Basic and optional elements of the methodology for RPV P-T limit curve development in WCAP-14040, Revision 3, are given in Sections 2.5, 2.6, 2.7, 2.8, and Appendix A of the TR.

In Section 2.5 of the TR, the fracture toughness-based guidelines from Appendix G to Section XI of the ASME Code are specified (based on the 1995 Edition through 1996 Addenda of the ASME Code). Notably, specific reference is made to the use of: (1) the ASME Code lower bound dynamic crack initiation/crack arrest (K_{IA}) fracture toughness curve; (2) the use of a postulated flaw that has a depth of one-quarter of the wall thickness and a 6:1 aspect ratio; and (3) the use of a structural factor of 2 on primary membrane stress intensities (K_{IM}) when evaluating normal heatup and cooldown and a structural factor of 1.5 on K_{IM} when evaluating hydrostatic/leak test conditions.

Optional guidelines for P-T limit curve development are also addressed in WCAP-14040, Revision 3. The option of using the ASME Code static crack initiation fracture toughness curve (K_{IC}), as given in ASME Code Case N-640, is addressed in Sections 2.5 and 2.8. The option of using ASME Code Case N-588, which enables the postulation of a circumferentially-oriented flaw (with appropriate stress magnification factors) when evaluating a circumferential weld, is addressed in Section 2.8. WCAP-14040, Revision 3, notes, however, that licensee use of the provisions of either ASME Code Case N-640 or N-588 requires, in accordance with 10 CFR 50.60(b), an exemption if the provisions of the Code Case are not contained in the edition of the ASME Code included in a facility's licensing basis. Appendix A to WCAP-14040, Revision 3, provides additional details regarding the application of optional ASME Code Cases and includes copies of ASME Code Case N-588, N-640, and N-641 (which effectively combines the provisions of N-588 and N-640 into a single Code Case).

A detailed discussion of the calculational methodology for P-T limit curve generation is given in Section 2.6 of the TR. Specific equations are given for the determination of primary membrane stresses due to internal pressure and membrane and bending stresses due to thermal gradients. Equations related to the generation of P-T limit curves for steady-state conditions, finite heatup rates, finite cooldown rates, and hydrostatic/leak test conditions are given. The equations given in WCAP-14040, Revision 3, are equivalent to those provided in Section XI of the ASME Code and consistent with the guidance given in SRP Section 5.3.2.

Therefore, the NRC staff has concluded that the basic methodology specified in WCAP-14040, Revision 3, for establishing P-T limit curves meets the regulatory requirements of Appendix G to 10 CFR Part 50 and the guidance provided in SRP Section 5.3.2. However, the NRC staff has concluded that the discussion provided in WCAP-14040, Revision 3, regarding the use of optional guidelines for the development of P-T limit curves, including the use of ASME Code Cases N-588, N-640, and N-641 is not acceptable. The NRC staff has concluded, based on guidance provided by the NRC's Office of the General Counsel, that licensees do not need to obtain exemptions to use the provisions of ASME Code Case N-588, N-640, or N-641. The basis for this decision is as follows. Appendix G to 10 CFR Part 50 references the use of ASME Code Section XI, Appendix G and defines the acceptable Editions and Addenda of the Code by reference to those endorsed in 10 CFR 50.55a. The 2003 Edition of 10 CFR Part 50, 10 CFR 50.55a, endorses editions and addenda of ASME Section XI up through the 1998 Edition and 2000 Addenda. The provisions of N-588, N-640, and N-641 have been directly incorporated into the Code in the 2000 Addenda version of ASME Section XI, Appendix G. Therefore, licensees may freely make use of the provisions in Code Cases N-588, N-640, and N-641 by using the methodology in the 2000 Addenda version of ASME Section XI without the need for an exemption. When published, the approved revision of TR WCAP-14040 should be modified to reflect this NRC staff conclusion.

Requirement 4: Regarding the development of RPV P-T limit curves, the methodology should describe how the minimum temperature requirements in Appendix G to 10 CFR Part 50 are applied when constructing P-T limit curves.

Minimum temperature requirements regarding the material in the highly stressed region of the RPV flange are given in Appendix G to 10 CFR Part 50. Information provided in Sections 2.9 and 2.10 of the TR addresses the incorporation of minimum temperature requirements into the development of P-T limit curves. In Section 2.9, the 10 CFR Part 50, Appendix G requirements are cited. WCAP-14040, Revision 3, goes on to note that there is an effort underway to revise or eliminate these requirements based on information contained in WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants." However, WCAP-14040, Revision 3, states that until Appendix G to 10 CFR Part 50 is revised to modify/eliminate the existing RPV flange minimum temperature requirements or an exemption request to modify/eliminate these requirements is approved by the NRC for a specific facility, the stated minimum temperature must be incorporated into a facility's P-T limit curves.

WCAP-14040, Revision 3, provides supplemental information in Section 2.10 regarding the establishment of RPV boltup temperature, specifically that the minimum boltup temperature should be 60 °F or equal to the highest material reference temperature in the highly stressed RPV flange region, whichever is higher (i.e., more conservative). Although no specific requirements related to boltup temperature are provided in Appendix G to 10 CFR Part 50, the information in WCAP-14040, Revision 3, is consistent with other, related requirements in Appendix G to 10 CFR Part 50 and in Appendix G to Section XI of the ASME Code.

The NRC staff concludes that the methodology specified in WCAP-14040, Revision 3, addresses RPV minimum temperature requirements in a way which is consistent with Appendix G to 10 CFR Part 50 and Appendix G to Section XI of the ASME Code and is, therefore, acceptable.

Requirement 5: Regarding the calculation of RPV materials' ARTs, the methodology should describe how the data from multiple surveillance capsules may be used in ART calculations.

Requirement 2 of Section 2.4 of WCAP-14040, Revision 3, addresses the determination of changes in material properties due to irradiation. This information includes a description of how surveillance capsule test results may be used to calculate RPV material properties in a manner which is consistent with Section C.2.1 of RG 1.99, Revision 2, and other NRC staff guidance.

The NRC staff has reviewed the information in Section 2.4 of the TR and determined that it is consistent with NRC staff guidance, including RG 1.99, Revision 2, and is, therefore, acceptable.

Requirement 6: Regarding the calculation of the neutron fluence, the methodololgy should describe how the neutron fluence is calculated.

Neutron Fluence Methodology

WCAP-14040, Revision 3, includes a revised Section 2.2. The revised section includes plant-specific transport calculations and the validity of the calculations. For the neutron transport calculations, the applicant is using the two-dimensional discrete ordinates code, DORT (Reference 1) with the BUGLE-96 cross section library (Reference 2). Approximations include a P_5 Legendre expansion for anisotropic scattering and a S_{16} order of angular quadrature. Space and energy dependent core power (neutron source) distributions and associated core parameters are treated on a fuel cycle specific basis. Two dimensional flux solutions $\Phi(r, \theta, z)$ are constructed using (r,θ) and (r,z) distributions. Extreme cases, with respect to power distribution arising from part-length fuel assemblies, use the three-dimensional TORT Code (Reference 1) with the BUGLE-96 cross section library. Source distribution is obtained from a burn-up weighted average of the power distributions of individual fuel cycles. The method accounts for source energy spectral effects and neutrons/fission due to burnup by tracking the concentration of U-235, U-238, Pu-239, Pu-240, and Pu-241. Mesh spacing accounts for flux gradients and material interfaces.

The proposed methodology, as outlined above, adheres to the guidance of RG 1.190, and therefore, is acceptable.

Validation of Transport Calculations

The Westinghouse validation is structured in four parts:

- comparison to pool critical assembly (PCA) simulator results (Reference 3),
- comparison to calculations in the H. B. Robinson benchmark (Reference 4),
- comparison to a measurement database from pressurized water reactor (PWR) surveillance capsules, and

- an analytical sensitivity study addressing the uncertainty components of the transport calculations.

Comparisons of calculated results to the corresponding PCA measured quantities establish the adequacy of the basic transport calculation and the associated cross sections. Comparison to the H.B. Robinson benchmark addresses uncertainties related to the method and generally to the neutron exposure. Comparisons to the PWR database provides an indication of the presence of a bias and of the uncertainty of the calculated value with respect to the corresponding measured values. Finally, the analytical sensitivity study validates the overall uncertainties whether from the methodology or the lack of precise knowledge of the input parameters.

Comparison of the measured data to the calculations was performed on the basis of measured/calculated (M/C) ratios, and with best estimate values calculated using least squares adjusted measured values. The least squares adjustment is based on weighing individual measurements based on spectral coverage. Comparisons are done before and after spectral adjustments. This method is addressed in RG 1.190, as well as in the ASTM Standard E944-96.

The NRC staff requested that the WOG address the completeness of its database. By letter dated October 20, 2003, the WOG responded by indicating that all of the surveillance capsules analyzed with the proposed methodology (DORT and BUGLE-96) are included in the database. The NRC staff found the response acceptable.

The NRC staff concludes that the proposed benchmarking methodology adheres to the guidance in RG 1.190 and to ASTM standards, and therefore, is acceptable.

Requirement 7: Regarding the low temperature overpressure protection/cold overpressure mitigating system, the lift setting limits for the power operated relief valves should be developed using NRC-approved methodologies.

The method in this section is identical to the existing method in the approved Revision 2 of WCAP-14040. The thermal hydraulics analysis for the mass and heat input transients is using the same specialized version of LOFTRAN, which was approved in Revision 2.

The cold overpressure mitigating system is the same as in the approved version, and therefore, the NRC staff finds it acceptable.

4.0 CONCLUSION

The NRC staff has reviewed the information provided in WCAP-14040, Revision 3, related to the requirements of GL 96-03, as cited in Section 3.0 of this SE, and finds WCAP-14040, Revision 3, to be acceptable for referencing as a PTLR methodology, subject to the following conditions:

-7- provision 2 in the table of Attachment 1 to

a. Licensees who wish to use WCAP-14040, Revision 3, as their PTLR methodology must provide additional information to address the methodology requirements in GL 96-03 related to the RPV material surveillance program ~~issues.~~ discussed

b. Contrary to the information in WCAP-14040, Revision 3, licensee use of the provisions of ASME Code Cases N-588, N-640, or N-641 in conjunction with the basic methodology in WCAP-14040, Revision 3, does not require an exemption since the provisions of these Code Cases are contained in the edition and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a. When published, the approved revision (Revision 4) of TR WCAP-14040 should be modified to reflect this NRC staff conclusion.

c. As stated in WCAP-14040, Revision 3, until Appendix G to 10 CFR Part 50 is revised to modify/eliminate the existing RPV flange minimum temperature requirements or an exemption request to modify/eliminate these requirements is approved by the NRC for a specific facility, the stated minimum temperature must be incorporated into a facility's P-T limit curves.

5.0 REFERENCES

1. "DOORS 3.1, One-, Two- and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System," Radiation Shielding Information Center (RSIC) Computer Code Collection CCC-650, Oak Ridge National Laboratory, August 1996.

2. "Bugle-96, Coupled 47 Neutron, 20 Gamma-Ray Group Cross-Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications," RSIC Data Library Collection DLC-185, Oak Ridge National Laboratory, March 1996.

3. NUREG/CR-6454 (ORNL/TM-1320~~4~~5), "Pool Critical Assembly Pressure Vessel Benchmark," by I. Remek and F.B.K. Kam, Oak Ridge National Laboratory, July 1997.

4. NUREG/CR-6453 (ORNL/TM-13204), "H.B. Robinson Pressure Vessel Benchmark," by I. Remek and F.B.K. Kam, Oak Ridge National Laboratory, February 1998.

Principal Contributors: M. Mitchell
L. Lois

Date: February 2, 2004

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001
February 2, 2004

RECEIVED
FEB 04 2004
WOG PROJECT OFFICE

Mr. Gordon Bischoff, Manager
Owners Group Program Management Office
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

SUBJECT: DRAFT SAFETY EVALUATION OF TOPICAL REPORT WCAP-14040, REVISION 3, "METHODOLOGY USED TO DEVELOP COLD OVERPRESSURE MITIGATING SYSTEM SETPOINTS AND RCS HEATUP AND COOLDOWN LIMIT CURVES" (TAC NO. MB5754)

Dear Mr. Bischoff:

On May 23, 2002, the Westinghouse Owners Group (WOG) submitted Topical Report (TR) WCAP-14040, Revision 3, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" to the staff for review. Based on questions posed by the NRC staff necessitating clarification of statements or editorial changes, the WOG revised WCAP-14040, Revision 3, and submitted the revised TR for staff review by letter dated October 20, 2003. Enclosed for the WOG's review and comment is a copy of the staff's draft safety evaluation (SE) for TR WCAP-14040, Revision 3.

Twenty working days are provided to you to comment on any factual errors or clarity concerns contained in the SE. The final SE will be issued after making any necessary changes, and will be made publicly available. The staff's disposition of your comments on the draft SE will be discussed in the final SE.

To facilitate the staff's review of your comments, please provide a marked-up copy of the draft SE showing proposed changes. Number the lines in the marked-up SE sequentially and provide a summary table of the proposed changes.

If you have any questions, please contact Drew Holland at (301) 415-1436.

Sincerely,

Stephen Dembek

Stephen Dembek, Chief, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 694

Enclosure: Draft Safety Evaluation

cc w/encl: See next page

Westinghouse Owners Group Project No. 694

cc:
Mr. John S. Galembush, Acting Manager
Regulatory Compliance and Plant Licensing
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DRAFT SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

WCAP-14040, REVISION 3, "METHODOLOGY USED TO DEVELOP COLD OVERPRESSURE MITIGATING SYSTEM SETPOINTS AND REACTOR COOLANT SYSTEM HEATUP AND COOLDOWN LIMIT CURVES"

WESTINGHOUSE OWNERS GROUP

PROJECT NO. 694

1.0 INTRODUCTION

By letter dated May 23, 2002, the Westinghouse Owners Group (WOG) submitted Topical Report (TR) WCAP-14040, Revision 3, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," for NRC staff review and approval. This TR was developed to define a methodology for reactor pressure vessel (RPV) pressure-temperature (P-T) limit curve development and, consistent with the guidance provided in NRC Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," for the development of plant-specific Pressure-Temperature Limit Reports (PTLRs). A prior revision, WCAP-14040, Revision 2, had been approved as a PTLR methodology by the NRC staff's safety evaluation dated October 16, 1995. WCAP-14040, Revision 3, was submitted for NRC staff approval to reflect recent changes in the WOG methodology. Given the scope of the changes incorporated in WCAP-14040, Revision 3, and a significant amount of rewriting which was done to improve clarity of some sections, the NRC staff reviewed the TR in its entirety. Based on questions posed by the NRC staff necessitating clarification of statements or editorial changes, the WOG revised WCAP-14040, Revision 3, and submitted the revised TR for NRC staff review and approval by letter dated October 20, 2003.

2.0 REGULATORY EVALUATION

Four specific topics are addressed in the context of the development of a PTLR methodology: (1) the calculation of neutron fluences for the RPV and RPV surveillance capsules; (2) the evaluation of RPV material properties due to changes caused by neutron radiation; (3) the development of appropriate P-T limit curves based on these RPV material properties and the establishment of cold overpressure mitigating system (COMS) setpoints to protect the RPV from brittle failure; and (4) the development of an RPV material surveillance program to monitor changes in RPV material properties due to radiation. Regulatory requirements related to the four topics noted above are addressed in Appendices G and H to Title 10 of the Code of Federal Regulations Part 50 (10 CFR Part 50). Appendix G to 10 CFR Part 50 provides requirements related to RPV P-T limit development and directly or indirectly addresses topics (1) through (3) above. Appendix H to 10 CFR Part 50 defines regulatory requirements related to RPV material surveillance programs and addresses topic (4) above.

For the staff's review of WCAP-14040, Revision 3, several additional guidance documents were used. NRC Standard Review Plan (SRP) Sections 5.2.2, "Overpressure Protection," 5.3.1, "Reactor Vessel Materials," and 5.3.2, "Pressure-Temperature Limits," provide specific review guidance related to RPV material property determination, P-T limit development, and COMS performance. Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," describes analysis procedures acceptable to the NRC staff for the purpose of assessing RPV material property changes due to radiation. RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," addresses NRC staff expectations for an acceptable fluence calculation methodology. American Society for Testing and Materials (ASTM) Standard Practice E 185, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," provides guidance on the establishment of RPV material surveillance programs and editions of ASTM E 185 are incorporated by reference into Appendix H to 10 CFR Part 50. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI, Appendix G provides specific requirements regarding the development of P-T limit curves.

Finally, specific guidance regarding topics and the level of detail to which they must be addressed as part of an acceptable PTLR methodology is given in GL 96-03.

3.0 TECHNICAL EVALUATION

The technical requirements to be addressed in an acceptable PTLR methodology are provided under the column heading "Minimum Requirements to be Included in Methodology" in the table entitled "Requirements for Methodology and PTLR" in Attachment 1 to GL 96-03. Summarized versions of the seven requirements are given below, along with the staff's technical evaluation of information in WCAP-14040, Revision 3, related to each requirement.

Requirement 1: Regarding the reactor vessel material surveillance program, the methodology should briefly describe the surveillance program. The methodology should clearly reference the requirements of Appendix H to 10 CFR Part 50.

The provisions of the methodology described in WCAP-14040, Revision 3, do not specify how the plant-specific RPV surveillance programs should be maintained in order to be in compliance with Appendix H to 10 CFR Part 50. Licensees who wish to use WCAP-14040, Revision 3, as their PTLR methodology must submit additional information to address the methodology requirements in GL 96-03 related to RPV material surveillance program issues.

Requirement 2: Regarding the calculation of RPV materials' adjusted reference temperatures (ART) values, the methodology should describe the method for calculating material ART values using RG 1.99, Revision 2.

Information regarding how material ARTs are to be determined within the WCAP-14040, Revision 3, PTLR methodology is provided in Section 2.3 and 2.4 of the TR. In Section 2.3, the determination of initial, unirradiated material properties from Charpy V-notch impact tests and/or nil-ductility drop weight tests is clearly defined. The methodology specified in Section 2.3 accurately incorporates the guidance found in ASME Code Section III, paragraph NB-2331 and additional information in SRP Section 5.3.1.

In Section 2.4 of the TR, the determination of changes in material properties due to irradiation is addressed, along with the determination of margins necessary to account for uncertainties in initial properties and irradiation damage assessment. The methodology specified in Section 2.4 accurately incorporates the guidance found in RG 1.99, Revision 2.

The NRC staff, therefore, determined that the methodology described for determining material ART values in WCAP-14040, Revision 3, was consistent with the guidance provided in the ASME Code, SRP Section 5.3.1, and RG 1.99, Revision 2, and was, therefore, acceptable.

Requirement 3: Regarding the development of RPV P-T limit curves, the methodology should describe the application of fracture mechanics-based calculations in constructing P-T limit curves based on the provisions of Appendix G to Section XI of the ASME Code and SRP Section 5.3.2.

Basic and optional elements of the methodology for RPV P-T limit curve development in WCAP-14040, Revision 3, are given in Sections 2.5, 2.6, 2.7, 2.8, and Appendix A of the TR.

In Section 2.5 of the TR, the fracture toughness-based guidelines from Appendix G to Section XI of the ASME Code are specified (based on the 1995 Edition through 1996 Addenda of the ASME Code). Notably, specific reference is made to the use of: (1) the ASME Code lower bound dynamic crack initiation/crack arrest (K_{IA}) fracture toughness curve; (2) the use of a postulated flaw that has a depth of one-quarter of the wall thickness and a 6:1 aspect ratio; and (3) the use of a structural factor of 2 on primary membrane stress intensities (K_{IM}) when evaluating normal heatup and cooldown and a structural factor of 1.5 on K_{IM} when evaluating hydrostatic/leak test conditions.

Optional guidelines for P-T limit curve development are also addressed in WCAP-14040, Revision 3. The option of using the ASME Code static crack initiation fracture toughness curve (K_{IC}), as given in ASME Code Case N-640, is addressed in Sections 2.5 and 2.8. The option of using ASME Code Case N-588, which enables the postulation of a circumferentially-oriented flaw (with appropriate stress magnification factors) when evaluating a circumferential weld, is addressed in Section 2.8. WCAP-14040, Revision 3, notes, however, that licensee use of the provisions of either ASME Code Case N-640 or N-588 requires, in accordance with 10 CFR 50.60(b), an exemption if the provisions of the Code Case are not contained in the edition of the ASME Code included in a facility's licensing basis. Appendix A to WCAP-14040, Revision 3, provides additional details regarding the application of optional ASME Code Cases and includes copies of ASME Code Case N-588, N-640, and N-641 (which effectively combines the provisions of N-588 and N-640 into a single Code Case).

A detailed discussion of the calculational methodology for P-T limit curve generation is given in Section 2.6 of the TR. Specific equations are given for the determination of primary membrane stresses due to internal pressure and membrane and bending stresses due to thermal gradients. Equations related to the generation of P-T limit curves for steady-state conditions, finite heatup rates, finite cooldown rates, and hydrostatic/leak test conditions are given. The equations given in WCAP-14040, Revision 3, are equivalent to those provided in Section XI of the ASME Code and consistent with the guidance given in SRP Section 5.3.2.

Therefore, the NRC staff has concluded that the basic methodology specified in WCAP-14040, Revision 3, for establishing P-T limit curves meets the regulatory requirements of Appendix G to 10 CFR Part 50 and the guidance provided in SRP Section 5.3.2. However, the NRC staff has concluded that the discussion provided in WCAP-14040, Revision 3, regarding the use of optional guidelines for the development of P-T limit curves, including the use of ASME Code Cases N-588, N-640, and N-641 is not acceptable. The NRC staff has concluded, based on guidance provided by the NRC's Office of the General Counsel, that licensees do not need to obtain exemptions to use the provisions of ASME Code Case N-588, N-640, or N-641. The basis for this decision is as follows. Appendix G to 10 CFR Part 50 references the use of ASME Code Section XI, Appendix G and defines the acceptable Editions and Addenda of the Code by reference to those endorsed in 10 CFR 50.55a. The 2003 Edition of 10 CFR Part 50, 10 CFR 50.55a, endorses editions and addenda of ASME Section XI up through the 1998 Edition and 2000 Addenda. The provisions of N-588, N-640, and N-641 have been directly incorporated into the Code in the 2000 Addenda version of ASME Section XI, Appendix G. Therefore, licensees may freely make use of the provisions in Code Cases N-588, N-640, and N-641 by using the methodology in the 2000 Addenda version of ASME Section XI without the need for an exemption. When published, the approved revision of TR WCAP-14040 should be modified to reflect this NRC staff conclusion.

Requirement 4: Regarding the development of RPV P-T limit curves, the methodology should describe how the minimum temperature requirements in Appendix G to 10 CFR Part 50 are applied when constructing P-T limit curves.

Minimum temperature requirements regarding the material in the highly stressed region of the RPV flange are given in Appendix G to 10 CFR Part 50. Information provided in Sections 2.9 and 2.10 of the TR addresses the incorporation of minimum temperature requirements into the development of P-T limit curves. In Section 2.9, the 10 CFR Part 50, Appendix G requirements are cited. WCAP-14040, Revision 3, goes on to note that there is an effort underway to revise or eliminate these requirements based on information contained in WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants." However, WCAP-14040, Revision 3, states that until Appendix G to 10 CFR Part 50 is revised to modify/eliminate the existing RPV flange minimum temperature requirements or an exemption request to modify/eliminate these requirements is approved by the NRC for a specific facility, the stated minimum temperature must be incorporated into a facility's P-T limit curves.

WCAP-14040, Revision 3, provides supplemental information in Section 2.10 regarding the establishment of RPV boltup temperature, specifically that the minimum boltup temperature should be 60 °F or equal to the highest material reference temperature in the highly stressed RPV flange region, whichever is higher (i.e., more conservative). Although no specific requirements related to boltup temperature are provided in Appendix G to 10 CFR Part 50, the information in WCAP-14040, Revision 3, is consistent with other, related requirements in Appendix G to 10 CFR Part 50 and in Appendix G to Section XI of the ASME Code.

The NRC staff concludes that the methodology specified in WCAP-14040, Revision 3, addresses RPV minimum temperature requirements in a way which is consistent with Appendix G to 10 CFR Part 50 and Appendix G to Section XI of the ASME Code and is, therefore, acceptable.

Requirement 5: Regarding the calculation of RPV materials' ARTs, the methodology should describe how the data from multiple surveillance capsules may be used in ART calculations.

Requirement 2 of Section 2.4 of WCAP-14040, Revision 3, addresses the determination of changes in material properties due to irradiation. This information includes a description of how surveillance capsule test results may be used to calculate RPV material properties in a manner which is consistent with Section C.2.1 of RG 1.99, Revision 2, and other NRC staff guidance.

The NRC staff has reviewed the information in Section 2.4 of the TR and determined that it is consistent with NRC staff guidance, including RG 1.99, Revision 2, and is, therefore, acceptable.

Requirement 6: Regarding the calculation of the neutron fluence, the methodololgy should describe how the neutron fluence is calculated.

Neutron Fluence Methodology

WCAP-14040, Revision 3, includes a revised Section 2.2. The revised section includes plant-specific transport calculations and the validity of the calculations. For the neutron transport calculations, the applicant is using the two-dimensional discrete ordinates code, DORT (Reference 1) with the BUGLE-96 cross section library (Reference 2). Approximations include a P_5 Legendre expansion for anisotropic scattering and a S_{16} order of angular quadrature. Space and energy dependent core power (neutron source) distributions and associated core parameters are treated on a fuel cycle specific basis. Two dimensional flux solutions $\Phi(r, \theta, z)$ are constructed using (r,θ) and (r,z) distributions. Extreme cases, with respect to power distribution arising from part-length fuel assemblies, use the three-dimensional TORT Code (Reference 1) with the BUGLE-96 cross section library. Source distribution is obtained from a burn-up weighted average of the power distributions of individual fuel cycles. The method accounts for source energy spectral effects and neutrons/fission due to burnup by tracking the concentration of U-235, U-238, Pu-239, Pu-240, and Pu-241. Mesh spacing accounts for flux gradients and material interfaces.

The proposed methodology, as outlined above, adheres to the guidance of RG 1.190, and therefore, is acceptable.

Validation of Transport Calculations

The Westinghouse validation is structured in four parts:

- comparison to pool critical assembly (PCA) simulator results (Reference 3),
- comparison to calculations in the H. B. Robinson benchmark (Reference 4),
- comparison to a measurement database from pressurized water reactor (PWR) surveillance capsules, and

- an analytical sensitivity study addressing the uncertainty components of the transport calculations.

Comparisons of calculated results to the corresponding PCA measured quantities establish the adequacy of the basic transport calculation and the associated cross sections. Comparison to the H.B. Robinson benchmark addresses uncertainties related to the method and generally to the neutron exposure. Comparisons to the PWR database provides an indication of the presence of a bias and of the uncertainty of the calculated value with respect to the corresponding measured values. Finally, the analytical sensitivity study validates the overall uncertainties whether from the methodology or the lack of precise knowledge of the input parameters.

Comparison of the measured data to the calculations was performed on the basis of measured/calculated (M/C) ratios, and with best estimate values calculated using least squares adjusted measured values. The least squares adjustment is based on weighing individual measurements based on spectral coverage. Comparisons are done before and after spectral adjustments. This method is addressed in RG 1.190, as well as in the ASTM Standard E944-96.

The NRC staff requested that the WOG address the completeness of its database. By letter dated October 20, 2003, the WOG responded by indicating that all of the surveillance capsules analyzed with the proposed methodology (DORT and BUGLE-96) are included in the database. The NRC staff found the response acceptable.

The NRC staff concludes that the proposed benchmarking methodology adheres to the guidance in RG 1.190 and to ASTM standards, and therefore, is acceptable.

Requirement 7: Regarding the low temperature overpressure protection/cold overpressure mitigating system, the lift setting limits for the power operated relief valves should be developed using NRC-approved methodologies.

The method in this section is identical to the existing method in the approved Revision 2 of WCAP-14040. The thermal hydraulics analysis for the mass and heat input transients is using the same specialized version of LOFTRAN, which was approved in Revision 2.

The cold overpressure mitigating system is the same as in the approved version, and therefore, the NRC staff finds it acceptable.

4.0 CONCLUSION

The NRC staff has reviewed the information provided in WCAP-14040, Revision 3, related to the requirements of GL 96-03, as cited in Section 3.0 of this SE, and finds WCAP-14040, Revision 3, to be acceptable for referencing as a PTLR methodology, subject to the following conditions:

a. Licensees who wish to use WCAP-14040, Revision 3, as their PTLR methodology must provide additional information to address the methodology requirements in GL 96-03 related to RPV material surveillance program issues.

b. Contrary to the information in WCAP-14040, Revision 3, licensee use of the provisions of ASME Code Cases N-588, N-640, or N-641 in conjunction with the basic methodology in WCAP-14040, Revision 3, does not require an exemption since the provisions of these Code Cases are contained in the edition and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a. When published, the approved revision of TR WCAP-14040 should be modified to reflect this NRC staff conclusion.

c. As stated in WCAP-14040, Revision 3, until Appendix G to 10 CFR Part 50 is revised to modify/eliminate the existing RPV flange minimum temperature requirements or an exemption request to modify/eliminate these requirements is approved by the NRC for a specific facility, the stated minimum temperature must be incorporated into a facility's P-T limit curves.

5.0 REFERENCES

1. "DOORS 3.1, One-, Two- and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System," Radiation Shielding Information Center (RSIC) Computer Code Collection CCC-650, Oak Ridge National Laboratory, August 1996.

2. "Bugle-96, Coupled 47 Neutron, 20 Gamma-Ray Group Cross-Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications," RSIC Data Library Collection DLC-185, Oak Ridge National Laboratory, March 1996.

3. NUREG/CR-6454 (ORNL/TM-13204), "Pool Critical Assembly Pressure Vessel Benchmark," by I. Remek and F.B.K. Kam, Oak Ridge National Laboratory, July 1997.

4. NUREG/CR-6453 (ORNL/TM-13204), "H.B. Robinson Pressure Vessel Benchmark," by I. Remek and F.B.K. Kam, Oak Ridge National Laboratory, February 1998.

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Date: February 2, 2004

October 20, 2003
WOG-03-550

WCAP-14040 Rev. 3
Project Number 694

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Washington, DC 20555-0001

Attention: Chief, Information Management Branch,
Division of Program Management

Subject: **Westinghouse Owners Group Response to Request for Additional Information on WCAP-14040 Rev. 3, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," (TAC No. MB5754)**

References:
1. WOG Letter, R. Bryan to Document Control Desk, "Transmittal of WCAP-14040, Rev. 3, Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," OG-02-018, May 23, 2002.

2. NRC Letter, D. Holland to G. Bischoff, "Request for Additional Information - WCAP-14040, Revision 3, Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," TAC NO. 5754, June 18, 2003.

In May 2002, the Westinghouse Owners Group submitted WCAP-14040, Rev. 3, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," for approval (Ref. 1). In June 2003, the NRC issued Requests for Additional Information (RAIs) concerning WCAP-14040, Rev. 3 (Ref. 2).

Attachment 1 to this letter contains the responses to the RAIs. Attachment 2 contains revisions to the affected pages of WCAP-14040, Rev. 3 that incorporate the responses to the RAIs. Attachment 3 contains revisions to Section 2.2 "Neutron Fluence Methodology" of WCAP-14040, Rev. 3. Although not made in response to any RAI, the changes to Section 2.2 of WCAP-14040, Rev. 3 were made to:

- Discuss how the current neutron fluence methodology follows the guidance contained in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," dated March 2001.

- Revise the text and benchmarking results to reflect the use of the BUGLE-96 ENDF/B-VI based cross-section library. The BUGLE-96 library provides an improved calculation relative to the previously used BUGLE-93 data set for some comparisons, particularly in the vessel wall and at ex-vessel dosimetry locations.

- Revise the discussion of the current version of the DORT code currently used.

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- Revise the text to reflect that consistent with the guidance of Regulatory Guide 1.190, the final results for the pressure vessel fluence projections are based on the plant specific transport calculations, and that the dosimetry data is only used to validate the calculated results.

The approved version of WCAP-14040 that will be issued following receipt of the NRC Safety Evaluation will incorporate the changes contained in Attachments 2 and 3.

If you require further information, feel free to contact Mr. Ken Vavrek, Westinghouse Owners Group Project Office at 412-374-4302.

Sincerely,

Frederick P. "Ted" Schiffley, II
Chairman, Westinghouse Owners Group

Attachments

cc: WOG Management Committee
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Attachment 1

Responses to NRC Request For Additional Information on WCAP-14040, Rev. 3, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves"

1. Section 2.3, page 2-5, Branch Technical Position MTEB 5-2 does not give fracture toughness "requirements." Revise WCAP-14040, Revision 3, to refer to the information in MTEB 5-2 as "guidelines" rather than "requirements."

   Response to RAI 1:

   The first sentence in the last paragraph of Section 2.3 on page 2-5 will be revised to "fracture toughness guidelines" rather than "fracture toughness requirements."

2. Section 2.4, page 2-6, when referring to the "Ai" term in Equation 2.4-3, revise your definition which refers to it as the "measured value of ΔRTNDT" - instead call it the "measured shift in the Charpy V-notch 30 ft-lb energy level between the unirradiated condition and the irradiated condition, fi."

   Response to RAI 2:

   The fifth paragraph in Section 2.4 on page 2-6 will be revised to "the measured shift in the Charpy V-notch 30 ft-lb energy level between the unirradiated condition and the irradiated condition, f_i."

3. Section 2.4, page 2-7, revise the sentence which reads, "If the measured value exceeds the predicted value (ΔRTNDT + 2σΔ), a supplement to the PTLR must be provided to demonstrate how the results affect the approved methodology," to state "If the measured value exceeds the predicted value (ΔRTNDT + 2σΔ), a supplement to the PTLR methodology must be provided for NRC staff review and approval to demonstrate how the results affect the approved methodology."

   Response to RAI 3:

   The last sentence in the second paragraph on page 2-7 will be revised to state that "a supplement to the PTLR must be submitted for NRC review and approval..."

4. Section 2.5, page 2-7, it is stated that Kia is the reference fracture toughness curve in Appendix G to Section XI of the ASME Code. Clarify this to note that this refers to Editions of the Code through the 1995 Edition/1996 Addenda. The most recent Edition and Addenda of the Code (1998 Edition through 2000 Addenda) incorporated by reference into 10 CFR 50.55a, however, uses Kic as the reference fracture toughness curve.

   Response to RAI 4:

   The reference to Appendix G, to Section XI of the ASME Code will be clarified that it is referring to the 1995 Edition through the 1996 Addenda in the first sentence of the second paragraph of Section 2.5 on page 2-7.

5. Section 2.5, page 2-8, the "note" regarding the use of a 1.223 vs. 1.233 coefficient in the Kia equation is meaningless and confusing unless one also explains that there was a typographical error in the 1989 Edition of Section XI, Appendix G (i.e., where the 1.233 was used). Revise WCAP-14040, Revision 3, to either eliminate this note or revise the note to offer additional explanation regarding the historical basis for the 1.223 vs. 1.233 issue.

   Response to RAI 5:

   The Note in the first paragraph on page 2-8 discussing the historical basis of 1.223 versus 1.233 will be deleted.

6. Section 2.5, page 2-8, when discussing ASME Code Case N-640, it is not correct to say that an exemption is required to implement N-640 because the NRC has not "endorsed" the Code Case. "Endorsement" implies that it has been included in Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability -- ASME Section XI, Division 1." Code Case N-640 would have to be included in the edition of the ASME Code which the licensee has adopted in their facility's licensing basis in order to comply with 10 CFR 50.55a before an exemption is no longer required.

   Response to RAI 6:

   The fifth paragraph on page 2-8 will be revised to delete the text "has not yet been endorsed by the NRC, and therefore use of this Code Case will" and to add the statement "if it is not contained in the edition of the ASME Code included in the unit licensing basis."

7. The statement in Section 2.5, page 2-10, regarding need for an exemption relative to modifying existing 10 CFR Part 50, Appendix G flange requirements should, for consistency be repeated in Section 2.8.

   Response to RAI 7:

   A statement that the flange requirement must be included in the P-T limits unless an exemption request is submitted and approved by the NRC will be added to the fourth paragraph in Section 2.8 on page 2-20.

8. Section 2.6.1, page 2-12, it is stated "[t]hese stress components are used for determining the thermal stress intensity factors, Kit, as described in the following subsection." The following subsection is 2.6.2, "Steady-State Analyses," and it does not address the calculation of Kit. Revise WCAP-14040, Revision 3, to address this apparent inconsistency.

   Response to RAI 8:

   The last sentence in the last paragraph of Subsection 2.6.1 on page 2-12 will be revised to "in subsections 2.6.3 and 2.6.4."

9. Section 2.6.2, page 2-14, and Section 2.6.5, page 2-15, Mm factors of 1.84, 0.918, and 3.18 are given for various reactor pressure vessel wall thickness ranges to be used when steady-state analyses are performed. It is unclear as to where these Mm factors come from (unable to locate them in any edition of ASME Section XI, Appendix G). Further, they are not consistent with what should be the same Mm factors cited on page 2-15. Revise WCAP-14040, Revision 3, to address this apparent inconsistency in the cited Mm factors.

   Response to RAI 9:

   The M_m factors discussed in Subsection 2.6.2 on page 2-14, and in Subsection 2.6.5 on page 2-15 will be deleted.

10. Section 2.7, page 2-19, it should be noted that an exemption is required when a licensee wishes to make use of ASME Code Case N-588. Revise WCAP-14040, Revision 3, accordingly.

   Response to RAI 10:

   A sentence will be added to the first paragraph in Section 2.7 on page 2-19 that states "An exemption request must be submitted and approved by the NRC if Code Case N-588 is not contained in the edition of the ASME Code included in the unit licensing basis."

Attachment 2

Revised WCAP-14040, Revision 3 Pages Incorporating NRC RAIs

The results of this latter comparison expressed in terms of the ratio of adjusted flux to calculated flux (A/C) are summarized as follows for the 82 capsule data base:

PARAMETER	A/C	STD DEV
$\phi(E > 1.0\ MeV)$	1.00	7.3%

As with the comparisons based on the linear average of reaction rate M/C ratios, the comparisons of the least squares adjusted results with the plant specific transport calculations demonstrate that the calculated results are essentially unbiased with an uncertainty well within the 20% criterion established in Regulatory Guide 1.190.

2.3 FRACTURE TOUGHNESS PROPERTIES

The fracture toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the requirements of Appendix G, 10 CFR Part 50[(4)], as augmented by the additional requirements in subsection NB-2331 of Section III of the ASME B&PV Code[(8)]. These fracture toughness requirements are also summarized in Branch Technical Position MTEB 5-2 ("Fracture Toughness Requirements")[(9)] of the NRC Regulatory Standard Review Plan.

These requirements are used to determine the value of the reference nil-ductility transition temperature (RT_{NDT}) for unirradiated material (defined as initial RT_{NDT}, IRT_{NDT}) and to calculate the adjusted reference temperature (ART) as described in Section 2.4. Two types of tests are required to determine a material's value of IRT_{NDT}: Charpy V-notch impact (C_v) tests and drop-weight tests. The procedure is as follows:

1. Determine a temperature T_{NDT} that is at or above the nil-ductility transition temperature by drop weight tests.

2. At a temperature not greater than T_{NDT} + 60°F, each specimen of the C_v test shall exhibit at least 35 mils lateral expansion and not less than 50 ft-lb absorbed energy. When these requirements are met, T_{NDT} is the reference temperature RT_{NDT}.

3. If the requirements of (2) above are not met, conduct additional C_v tests in groups of three specimens to determine the temperature T_{Cv} at which they are met. In this case the reference temperature $RT_{NDT} = T_{Cv}$ - 60°F. Thus, the reference temperature RT_{NDT} is the higher of T_{NDT} and (T_{Cv} - 60°F).

4. If the C_v test has not been performed at T_{NDT} + 60°F, or when the C_v test at T_{NDT} + 60°F does not exhibit a minimum of 50 ft-lb and 35 mils lateral expansion, a temperature representing a minimum of 50 ft-lb and 35 mils lateral expansion may be obtained from a full C_v impact curve developed from the minimum data points of all the C_v tests performed as shown in Figure 2.1.

Plants that do not follow the fracture toughness guidelines in Branch Technical Position MTEB 5-2 to determine IRT_{NDT} can use alternative procedures. However, sufficient technical justification and special circumstances per the criteria of 10CFR50.12(a)(2) must be provided for an exemption from the regulations to be granted by the NRC.

2.4 CALCULATION OF ADJUSTED REFERENCE TEMPERATURE

The adjusted reference temperature (ART) for each material in the beltline region is calculated in accordance with Regulatory Guide 1.99, Revision 2[3]. The most limiting ART values (i.e., highest value at 1/4t and 3/4t locations) are used in determining the pressure-temperature limit curves. ART is calculated by the following equation:

$$ART = IRT_{NDT} + \Delta RT_{NDT} + Margin \tag{2.4-1}$$

IRT_{NDT} is the reference temperature for the unirradiated material as defined in paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code[8] and calculated per Section 2.3. If measured values of IRT_{NDT} are not available for the material in question, generic mean values for that class of material can be used if there are sufficient test results to establish a mean and standard deviation for the class.

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

$$\Delta RT_{NDT} = CF\ f^{\,(0.28 - 0.10 \log f)} \tag{2.4-2}$$

CF (°F) is the chemistry factor and is a function of copper and nickel content. CF is given in Table 1 of Reference 3 for weld metal and in Table 2 in Reference 3 for base metal (Position 1.1 of Regulatory Guide 1.99, Revision 2). In Tables 1 and 2 of Reference 3 "weight-percent copper" and "weight-percent nickel" are the best-estimate values for the material and linear interpolation is permitted. When two or more credible surveillance data sets (as defined in Regulatory Guide 1.99, Revision 2, Paragraph B.4) become available they may be used to calculate the chemistry factor per Position 2.1 of Regulatory Guide 1.99, Revision 2, as follows:

$$CF = \frac{\sum_{i=1}^{n} [A_i\ f_i^{\,(0.28-0.10 \log f_i)}]}{\sum_{i=1}^{n} [f_i^{\,(0.28-0.10 \log f_i)}]^2} \tag{2.4-3}$$

Where "n" is the number of surveillance data points, "A_i" is the measured shift in the Charpy V-notch 30 ft-lb energy level between the unirradiated condition and the irradiated condition, "f_i." Where "f_i" is the fluence for each surveillance data point.

If Position 2.1 of Regulatory Guide 1.99, Revision 2, results in a higher value of ART than Position 1.1 of Regulatory Guide 1.99, Revision 2, the ART calculated per Position 2.1 must be used. However, if Position 2.1 of Regulatory Guide 1.99, Revision 2, results in a lower value of ART than Position 1.1 of Regulatory Guide 1.99, Revision 2, either value of ART may be used.

To calculate ΔRT_{NDT} at any depth (e.g., at 1/4t or 3/4t), the following formula is used to attenuate the fast neutron fluence (E> 1 MeV) at the specified depth.

$$f = f_{surface}\, e^{(-0.24x)} \qquad (2.4\text{-}4)$$

where $f_{surface}$ 10^{19} n/cm^2, E > 1 MeV) is the value, calculated per Section 2.2, of the neutron fluence at the base metal surface of the vessel at the location of the postulated defect, and x (in inches) is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then put into equation (2.4-2) to calculate ΔRT_{NDT} at the specified depth.

When two or more credible surveillance capsules have been removed, the measured increase in reference temperature (ΔRT_{NDT}) must be compared to the predicted increase in RT_{NDT} for each surveillance material. The predicted increase in RT_{NDT} is the mean shift in RT_{NDT} calculated by equation (2.4-2) plus two standard deviations ($2\sigma_\Delta$) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value ($\Delta RT_{NDT} + 2\sigma_\Delta$), a supplement to the PTLR must be submitted for NRC review and approval to demonstrate how the results affect the approved methodology.

Margin is the temperature value that is included in the ART calculations to obtain conservative, upper-bound values of ART for the calculations required by Appendix G to 10 CFR Part 50[4]. Margin is calculated by the following equation:

$$\text{Margin} = 2\,[(\sigma_I^2 + \sigma_\Delta^2)]^{0.5} \qquad (2.4\text{-}5)$$

σ_I, is the standard deviation for IRT_{NDT} and σ_Δ is the standard deviation for ΔRT_{NDT}. If IRT_{NDT} is a measured value, σ_I, is estimated from the precision of the test method ($\sigma_I = 0$ for a measured IRT_{NDT} of a single material). If IRT_{NDT} is not a measured value and generic mean values for that class of material are used, σ_I is the standard deviation obtained from the set of data used to establish the mean. Per Regulatory Guide 1.99, σ_Δ is 28°F for welds and 17°F for base metal. When surveillance data is used to calculate ΔRT_{NDT}, σ_Δ values may be reduced by one-half. In all cases, σ_Δ need not exceed half of the mean value of ΔRT_{NDT}.

2.5 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

The ASME Code requirements[5] for calculating the allowable pressure-temperature limit curves for various heatup and cooldown rates specify 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, the fracture toughness for the metal temperature at that time. Two values of fracture toughness may be used, K_{Ia} or K_{Ic}.

K_{Ia} is obtained from the reference fracture toughness curve, defined in Appendix G, to Section XI of the ASME Code (1995 Edition through the 1996 Addenda). (Note that in Appendix G, to Section III of the ASME Code, the reference fracture toughness is denoted as K_{IR}, whereas in Appendix G of Section XI, the reference fracture toughness is denoted as K_{Ia}. However, the K_{IR} and K_{Ia} curves are identical and are defined with the identical functional form.) The K_{Ia} curve is given by the following equation:

$$K_{Ia} = 26.78 + 1.223 \exp\,[0.0145\,(T - RT_{NDT} + 160)] \qquad (2.5\text{-}1)$$

where,

K_{Ia} = lower bound of dynamic and crack arrest toughness as a function of the metal temperature T and the metal reference nil-ductility transition temperature RT_{NDT}, (ksi $\sqrt{in}$). The value of RT_{NDT} is the adjusted reference temperature (ART) of Section 2.4.

K_{Ic} is also obtained from Section XI of the ASME Code, for example in Appendix A, and is a lower bound of static fracture toughness. Since heatup and cooldown is a slow process, static properties are appropriate. The K_{Ic} curve is given by the following expression:

$$K_{Ic} = 33.20 + 20.734 \exp [0.0200 (T - RT_{NDT})] \qquad (2.5\text{-}2)$$

The use of the K_{Ic} curve (Section XI, Appendix A) as a basis for developing P-T limit curves is currently contained in ASME Code Case N640. Use of the K_{Ic} fracture toughness will yield less limiting P-T curves, which is clearly a benefit.

However, the use of Code Case 640 presently includes a restriction on the setpoints for the Cold Overpressure Mitigation System (COMS). This maximum pressure for the COMS system is 100% of the pressure allowed by the P-T limit curves. This essentially disallows the use of Code Case N514 in these circumstances, meaning that the COMS system must protect to the actual P-T limit curve, rather than 110 percent, as allowed by Code Case N514.

The use of Code Case N640 requires an exemption under 10CFR50.60 paragraph (b), pertaining to proposed alternatives to the requirements of Appendices G and H, if it is not contained in the edition of the ASME Code included in the unit licensing basis.

The governing equation for generating pressure-temperature limit curves is defined in Appendix G of the ASME Code[5] as follows:

$$C\,K_{IM} + K_{It} < \text{Reference Fracture Toughness} \qquad (2.5\text{-}3)$$

where,

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

K_{It} = stress intensity factor caused by the thermal gradients through the vessel wall,

C = 2.0 for Level A and Level B service limits (for heatup and cooldown),

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

Reference Fracture Toughness = K_{Ia} or K_{Ic}, as discussed above

(Note: K_{It} is set to zero for hydrostatic and leak test calculations since these tests are performed at isothermal conditions).

The quantities E and α are temperature-dependent properties. However, to simplify the analysis, E and α are evaluated at an equivalent wall temperature at a given time:

$$T_{eqv} = \frac{2\int_{r_i}^{r_o} T(r) r \, dr}{r_o^2 - r_i^2} \qquad (2.6.1\text{-}5)$$

E and α are calculated as a function of this equivalent temperature and the Eα product in equation (2.6.1-4) is treated as a constant in the computation of hoop thermal stress.

The linear bending (σ_b) and constant membrane (σ_m) stress components of the thermal hoop stress profile are approximated by the linearization technique presented in Appendix A, to Section XI of the ASME Code[15]. These stress components are used for determining the thermal stress intensity factors, K_{It}, as described in subsections 2.6.3 and 2.6.4.

2.6.2 Steady-State Analyses

Using the calculated beltline metal temperature and the metal reference nil-ductility transition temperature, the reference stress intensity factor (K_{Ia}) is determined in Equation (2.5-1) at the 1/4t location where "t" represents the vessel wall thickness. At the 1/4t location, a 1/4 thickness flaw is assumed to originate at the vessel inside radius.

The allowable pressure $P(T_c)$ is a function of coolant temperature, and the pressure temperature curve is calculated for the steady state case at the assumed 1/4t inside surface flaw. First, the maximum allowable membrane (pressure) stress intensity factor is determined using the factor of 2.0 from equation (2.5-2) and the following equation:

$$K_{IM(max)} = \frac{K_I * (T - RT_{NDT})_{1/4t}}{2.0} \qquad (2.6.2\text{-}1)$$

where,

K_{Ia} (T-RT_{NDT}) = allowable reference stress intensity factor as a function of T-RT_{NDT} at 1/4t. (See Sections 2.7 and 2.8 for the new approach using Code Cases N640 and N588.)

Next, the maximum allowable pressure stress is determined using an iterative process and the following three equations:

$$Q = \varphi^2 - 0.212\left(\frac{\sigma_P}{\sigma_y}\right)^2 \qquad (2.6.2\text{-}2)$$

$$\sigma_P = \frac{K_{IM(max)}}{1.1\,M_K\sqrt{\frac{\pi a}{Q}}} \quad (2.6.2\text{-}3)$$

$$K_{IP} = 1.1 M_K \sigma_P \sqrt{\frac{\pi a}{Q}} \quad (2.6.2\text{-}4)$$

where,

Q = flaw shape factor modified for plastic zone size[16],

ϕ = is the elliptical integral of the 2nd kind (ϕ = 1.11376 for the fixed aspect ratio of 3 of the code reference flaw)[16],

0.212 = plastic zone size correction factor[16],

σ_P = pressure stress,

σ_y = yield stress,

1.1 = correction factor for surface breaking flaws,

M_K = correction factor for constant membrane stress [16], M_K as function of relativeflaw depth (a/t) is shown in Figure 2.4,

a = crack depth of 1/4t,

K_{IP} = pressure stress intensity factor.

The maximum allowable pressure stress is determined by incrementing σ_p from an initial value of 0.0 psi until a pressure stress is found that computes a K_{IP} value within 1.0001 of the $K_{IM(max)}$ value. After the maximum allowable σ_p is found, the maximum allowable internal pressure is determined by

$$P(T_c) = \sigma_P \left[\frac{r_o^2 - r_i^2}{r_o^2 + r_i^2}\right] \quad (2.6.2\text{-}5)$$

where,

$P(T_c)$ = calculated allowable pressure as a function of coolant temperature.

2.6.3 Finite Cooldown Rate Analyses

For each cooldown rate the pressure-temperature curve is calculated at the inside 1/4t location. First, the thermal stress intensity factor is calculated for a coolant temperature at a given time using the following equation from the Welding Research Council[16]:

$$K_{It} = [\sigma_m 1.1 M_K + \sigma_b M_B]\sqrt{\frac{\pi a}{Q}} \qquad (2.6.3\text{-}1)$$

where,

σ_m = constant membrane stress component from the linearized thermal hoop stress distribution,

σ_b = linear bending stress component from the linearized thermal hoop stress distribution,

M_X = correction factor for membrane stress[16] (see Figure 2.4),

M_B = correction factor for bending stress[16], M_B as a function of relative flaw depth (a/t) is shown in Figure 2.5.

The flaw shape factor Q in equation (2.6.2-6) is calculated from[16]

$$Q = \phi^2 - 0.212\left(\frac{\sigma_m + \sigma_b}{\sigma_y}\right)^2 \qquad (2.6.3\text{-}2)$$

Once K_{It} is computed, the maximum allowable membrane (pressure) stress intensity factor is determined using the factor of 2.0 from equation (2.5-2) and the following equation:

$$K_{IM(max)} = \frac{K_I * (T - RT_{NDT})_{1/4t} - K_{It}(T_c)_{1/4t}}{2.0} \qquad (2.6.3\text{-}3)$$

From $K_{IM(max)}$, the maximum allowable pressure is determined using the iterative process described above and equations (2.6.2-2) through (2.6.2-5).

The steady-state pressure-temperature curve of Section 2.6.2 is compared to the cooldown curves for the 1/4t inside surface flaw at each cooldown rate. At any time, the allowable pressure is the lesser of the two values, and the resulting curve is called the composite cooldown limit curve.

Finally, the 10 CFR Part 50[4] requirement for the closure flange region is incorporated into the cooldown composite curve as described in Section 2.5.

2.6.4 Finite Heatup Rate Analyses

Using the calculated beltline metal temperature and the metal reference nil-ductility transition temperature, the reference stress intensity factor (K_{Ia}) is determined in Equation (2.5-1) or (2.5-2) at both the 1/4t and 3/4t locations where "t" represents the vessel wall thickness. At the 1/4t location, a 1/4 thickness flaw is assumed to originate at the vessel inside radius. At the 3/4t location, a 1/4t flaw is assumed to originate on the outside of the vessel.

For each heatup rate a pressure-temperature curve is calculated at the 1/4t and 3/4t locations. First, the thermal stress intensity factor is calculated at the 1/4t and 3/4t locations for a coolant temperature at a given time using Option 1 or 2 from Section 2.6.3.

Once K_{It} is computed, the maximum allowable membrane (pressure) stress intensity factors at the 1/4t and 3/4t locations are determined using the following equations:

$$\text{At 1/4t,} \quad K_{IM(max)1/4t} = \frac{K_I * (T - RT_{NDT})_{1/4t} - K_{It}(T_c)_{1/4t}}{2.0} \qquad (2.6.4\text{-}1)$$

$$\text{At 3/4t,} \quad K_{IM(max)3/4t} = \frac{K_I * (T - RT_{NDT})_{3/4t} - K_{It}(T_c)_{3/4t}}{2.0} \qquad (2.6.4\text{-}2)$$

From $K_{IM(max)1/4t}$ and $K_{IM(max)3/4t}$, the maximum allowable pressure at both the 1/4t and 3/4t locations is determined using the iterative process described in Section 2.6.2 and equations (2.6.2-2) through (2.6.2-5).

As was done with the cooldown case, the steady state pressure-temperature curve of Section 2.6.2 is compared with the 1/4t and 3/4t location heatup curves for each heatup rate, with the lowest of the three being used to generate the composite heatup limit curve. The composite curve is then adjusted for the 10 CFR Part 50[4] rule for closure flange requirements, as discussed in Section 2.5.

2.6.5 Hydrostatic and Leak Test Curve Analyses

The minimum inservice hydrostatic leak test curve is determined by calculating the minimum allowable temperature at two pressure values (pressure values of 2000 psig and 2485 psig, approximately 110% of operating pressure, are generally used). The curve is generated by drawing a line between the two pressure-temperature data points. The governing equation for generating the hydrostatic leak test pressure-temperature limit curve is defined in Appendix G, Section X1, of the ASME Code[5] as follows:

$$1.5\, K_{IM} < K_{Ia} \qquad (2.6.5\text{-}1)$$

where, K_{IM} is the stress intensity factor caused by the membrane (pressure) stress and K_{Ia} is the reference stress intensity factor as defined in equation (2.5-1). Note that the thermal stress intensity factor is neglected (i.e., $K_{It} = 0$) since the hydrostatic leak test is performed at isothermal conditions.

The pressure stress is determined by,

$$\sigma_P = \left[\frac{r_o^2 + r_i^2}{r_o^2 - r_i^2}\right] P \qquad (2.6.5\text{-}2)$$

where,

P = the input pressure (generally 2000 and 2485 psig)

Next, the pressure stress intensity factor is calculated for a 1/4t flaw by,

$$K_{IM} = \left[1.1 M_K \sqrt{\frac{\pi a}{Q}}\right] \sigma_P \qquad (2.6.5\text{-}3)$$

The K_{IM} result is multiplied by the 1.5 factor of equation (2.5-2) and divided by 1000,

$$K_{HYD} = \frac{1.5 K_{IM}}{1000} \qquad (2.6.5\text{-}4)$$

Finally, the minimum allowable temperature is determined by setting K_{HYD} to K_{Ia} in equation (2.5-1) and solving for temperature T:

$$T = \frac{\ell n \left[\frac{(K_{HYD} - 26.78)}{1.223}\right]}{0.0145} + RT_{NDT} - 160.0 \qquad (2.6.5\text{-}5)$$

The 1983 Amendment to 10CFR50[4] has a rule which addresses the test temperature for hydrostatic pressure tests. This rule states that, when there is no fuel in the reactor vessel during hydrostatic pressure tests or leak tests, the minimum allowable test temperature must be 60°F above the adjusted reference temperature of the beltline region material that is controlling. If fuel is present in the reactor vessel during hydrostatic pressure tests or leak tests, the requirements of this section and Section 2.5 must be met.

2.7 1996 ADDENDA TO ASME SECTION XI, APPENDIX G METHODOLOGY

ASME Section XI, Appendix G was updated in 1996 to incorporate the most recent elastic solutions for K_I due to pressure and radial thermal gradients. The new solutions are based on finite element analyses for inside surface flaws performed at Oak Ridge National Laboratories and sponsored by the NRC, and work published for outside surface flaws. These solutions provide results that are very similar to those obtained by using solutions previously developed by Raju and Newman.

This revision provides consistent computational methods for pressure and thermal K_I, for thermal gradients through the vessel wall at any time during the transient. Consistent with the original version of

Appendix G, no contribution for crack face pressure is included in the K_I due to pressure, and cladding effects are neglected.

Using these elastic solutions in the low temperature region will provide some relief to restrictions associated with reactor operation at relatively low temperatures. Although the relief is relatively small in terms of the absolute allowable pressure, the benefits are substantial, because even a small increase in the allowable pressure can be a significant percentage increase in the operating window at relatively low temperatures. Implementing this revision results in a safety benefit (reduced likelihood of lifting COMS relief valves), with no reduction in vessel integrity.

The following revisions were made to ASME Section XI, Appendix G:

G-2214.1 Membrane Tension:

$$K_{Im} = M_m \times (pR_i / t) \qquad (2.7\text{-}1)$$

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

M_m = 1.85 for $\sqrt{t} < 2$,

M_m = $0.926\sqrt{t}$ for $2 \le \sqrt{t} \le 3.464$,

M_m = 3.21 for $\sqrt{t} > 3.464$

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

M_m = 1.77 for $\sqrt{t} < 2$,

M_m = $0.893\sqrt{t}$ for $2 \le \sqrt{t} \le 3.464$,

M_m = 3.09 for $\sqrt{t} > 3.464$

where,

p = internal pressure,

Ri = vessel inner radius, and

t = vessel wall thickness.

For Bending Stress, the K_I corresponding to bending stress for the postulated defect is:

$K_{Ib} = M_b$ * maximum bending stress, where $M_b = 0.667\ M_m$

For the Radial Thermal Gradient, the maximum K_I produced by radial thermal gradient for the postulated inside surface defect is:

$$K_{It} = 0.953 \times 10^{-3}\ CR\ t^{2.5} \qquad (2.7\text{-}2)$$

where:

CR = the cooldown rate in °F/hr.

For the Radial Thermal Gradient, the maximum K_I produced by radial thermal gradient for the postulated outside surface defect is:

$$K_{It} = 0.753 \times 10^{-3}\ HU\ t^{2.5} \qquad (2.7\text{-}3)$$

where:

HU = the heatup rate in °F/hr.

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

1. The maximum thermal K_I relationship and the temperature relationship in Figure G-2214-1 are applicable only for the conditions given in G-2214.3(a)(1) and (2) of Appendix G to ASME Section XI.

2. Alternatively, the K_I for radial thermal gradient can be calculated for any thermal stress distribution and at any specified time during cooldown for a ¼-thickness inside surface defect using the relationship:

$$K_{It} = (1.0359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3) * \sqrt{\pi a} \qquad (2.7\text{-}4)$$

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

$$K_{It} = (1.043C_0 + 0.630C_1 + 0.481C_2 + 0.401C_3) * \sqrt{\pi a} \qquad (2.7\text{-}5)$$

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

$$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3 \qquad (2.7\text{-}6)$$

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

Once K_{Ia} (As calculated via Equation 2.5-1) is known, the pressure can be solved using Equation 2.5-3 with the newly calculated K_{It} and new equation for K_{IM}.

$$C * [M_m \times (pR_i / t)] + K_{It} < K_{Ia}$$

where:

C = 2.0 for Level A and Level B service limits (for heatup and cooldown),

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

This results in a pressure equation as follows:

$$p = \frac{[K_{It} - K_{Ia}]}{C * M_m * (R_i / t)} \quad (2.7\text{-}7)$$

Note that K_{It} is equal to zero for steady state and hydrostatic leak test conditions. In addition, K_{Ia} and K_{It} must be calculated individually for inside and outside flaw locations (i.e., the ¼T and ¾T wall locations) and the minimum pressure must be used from these two locations. [Note: K_{Ia} for ¼ T steady state is not the same as K_{Ia} for ¼T thermal conditions since the wall temperature is equal to the water temperature in steady state, but is not the case under thermal conditions.]

2.7 CODE CASES N-640 FOR K_{Ic} and N-588 FOR CIRCUMFERENTIAL WELD FLAWS

2.8.1 ASME Code Case N-640

In February of 1999, the ASME Code approved Code Case N-640 which allows the use of the reference fracture toughness curve K_{Ic}, as found in Appendix A of Section XI, in lieu of Figure G-2110-1 in Appendix G for the development of pressure-temperature limit curves. (This is also described in Section 2.5 herein). Thus, when developing pressure-temperature limit curves, it is acceptable to calculate the reference stress intensity via Equation 2.5-2, in lieu of Equation 2.5-1. In addition, the K_{Ic} can be substituted for K_{Ia} in Equations 2.5-3, 2.6.2-1, 2.6.3-3, 2.6.4-1, 2.6.4-2, 2.6.5-1 and 2.7-7. An exemption request must be submitted and approved by the NRC if ASME Code Case N-640 is not contained in the edition of the ASME Code included in the unit licensing basis.

2.8.2 ASME Code Case N-588

In 1997, ASME Section XI, Appendix G was revised to add a methodology for the use of circumferential flaws when considering circumferential welds in developing pressure-temperature limit curves. This change was also implemented in a separate Code Case, N-588. An exemption request must be submitted and approved by the NRC if Code Case N-588 is not contained in the edition of the ASME Code included in the unit licensing basis.

The original ASME Section XI, Appendix G approach mandated the postulation of an axial flaw in circumferential welds for the purposes of calculating pressure-temperature limits. Postulating the Appendix G reference flaw in a circumferential weld is physically unrealistic because the length of the reference flaw is 1.5 times the vessel thickness and is much longer than the width of the vessel girth welds. In addition, historical experience, with repair weld indications found during pre-service inspection and data taken from destructive examination of actual vessel welds, confirms that any flaws are small, laminar in nature and are not oriented transverse to the weld bead orientation. Because of this, any defects potentially introduced during fabrication process (and not detected during subsequent

non-destructive examinations) should only be oriented along the direction of the weld fabrication. Thus, for circumferential welds, any postulated defect should be in the circumferential orientation.

The revision to Section XI, Appendix G now eliminates additional conservatism in the assumed flaw orientation for circumferential welds. The following revisions were made to ASME Section XI, Appendix G:

G-2214.1 Membrane Tension...

The K_I corresponding to membrane tension for the postulated circumferential defect of G-2120 is

$$K_{IM} = M_m \times (PR_i/t)$$

Where, M_m for an inside surface flaw is given by:

M_m = 0.89 for $\sqrt{t} < 2$,

M_m = 0.443 $\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$,

M_m = 1.53 for $\sqrt{t} > 3.464$

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

M_m = 0.89 for $\sqrt{t} < 2$,

M_m = 0.443 $\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$,

M_m = 1.53 for $\sqrt{t} > 3.464$

Note, that the only change relative to the OPERLIM computer code was the addition of the constants for M_m in a circumferential weld limited condition. No other changes were made to the OPERLIM computer code with regard to P-T calculation methodology.

2.9 CLOSURE HEAD/VESSEL FLANGE REQUIREMENTS

10 CFR Part 50, Appendix G contains the requirements for the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure flange regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the pre-service hydrostatic test pressure (3106 psig), which is 621 psig for a typical Westinghouse reactor vessel design.

This requirement was originally based on concerns about the fracture margin in the closure flange region. During the boltup process, stresses in this region typically reach over 70 percent of the steady-state stress, without being at steady-state temperature. The margin of 120°F and the pressure limitation of 20 percent of hydrotest pressure were developed using the K_{Ia} fracture toughness, in the mid 1970s.

Improved knowledge of fracture toughness and other issues which affect the integrity of the reactor vessel have led to the recent change to allow the use of K_{Ic} in the development of pressure-temperature curves,

as contained in Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1."

The discussion given in WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants," concluded that the integrity of the closure head/vessel flange region is not a concern for any of the operating plants using the K_{Ic} toughness. Furthermore, there are no known mechanisms of degradation for this region, other than fatigue. The calculated design fatigue usage for this region is less than 0.1, so it may be concluded that flaws are unlikely to initiate in this region. It is therefore clear that no additional boltup requirements are necessary, and therefore the requirement of 10 CFR Part 50, Appendix G, can be eliminated from the Pressure-Temperature Curves, once the requirements of 10CFR50 Appendix G are changed. However, until 10CFR50 Appendix G is revised to eliminate the flange requirement, it must be included in the P-T limits, unless an exemption request is submitted and approved by the NRC.

2.10 MINIMUM BOLTUP TEMPERATURE

The minimum boltup temperature is equal to the material RT_{NDT} of the stressed region. The RT_{NDT} is calculated in accordance with the methods described in Branch Technical Position MTEB 5-2. The Westinghouse position is that the minimum boltup temperature be no lower than 60°F. Thus, the minimum boltup temperature should be 60°F or the material RT_{NDT} whichever is higher.

reference temperature for weld or base metal in the beltline region at a distance one-fourth of the vessel section thickness from the vessel inside surface, as determined by Regulatory Guide 1.99, Revision 2.

3.4 ENABLE TEMPERATURE FOR COMS

The enable temperature is the temperature below which the COMS system is required to be operable. The definition of the enabling temperature currently approved and supported by the NRC is described in Branch Technical Position RSB 5-2[18]. This position defines the enable temperature for LTOP systems as the water temperature corresponding to a metal temperature of at least RT_{NDT} + 90°F at the beltline location (1/4t or 3/4t) that is controlling in the Appendix G limit calculations. This definition is very conservative, and is mostly based on material properties and fracture mechanics, with the understanding that material temperatures of RT_{NDT} + 90°F at the critical location will be well up the transition curve from brittle to ductile properties, and therefore brittle fracture of the vessel is not expected.

The ASME Code Case N-514 supports an enable temperature of RT_{NDT} + 50°F or 200°F, whichever is greater as described in Section 3.3.

A significant improvement in the enable temperature can be obtained by application of code case N641. This code case incorporates the benefits of code cases N588, and N640. The resulting enable temperatures for the Westinghouse designs obtained using code case N641 are listed below.

The use of Code Case N641 has not yet been approved by the NRC, and therefore the use of this Code Case will require approval of an exemption request, as discussed in under 10CFR50.60 paragraph (b), pertaining to proposed alternatives to the requirements of Appendices G and H.

Vessel Type	Axial Flaw	Circumferential Flaw
2 – loop	RT_{NDT} + 23F	Any temperature
3 – loop	RT_{NDT} + 30F	RT_{NDT} – 174F
4 – loop	RT_{NDT} + 34F	RT_{NDT} – 110F

The RCS cold leg temperature limitation for starting an RCP is the same value as the COMS enable temperature to ensure that the basis of the heat injection transient is not violated. The Standard Technical Specifications (STS) prohibit starting an RCP when any RCS cold leg temperatures is less than or equal to the COMS enable temperature unless the secondary side water temperature of each steam generator is less than or equal to 50°F above each of the RCS cold leg temperatures.

Table A-1	Status of ASME Nuclear Code Cases Associated with the P-T Limit Curve/COMS Methodology			
Code Case	**Title**	**Approved by ASME**	**Section XI of the ASME Code**	**Exemption Request Granted**
514	Low Temperature Overpressure Protection	2/12/92	1995 Edition through the 1996 Addenda	Yes
588	Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessel	12/12/97	1998 Edition through the 2000 Addenda	Yes
640	Alternative Reference Fracture Toughness for Development of P-T Limit Curves	2/26/99	1998 Edition through the 2000 Addenda	Yes
641	Alternative Pressure Temperature Relationship and Low Temperature Overpressure Protection System Requirement	1/17/00	1998 Edition through the 2000 Addenda	Yes

Attachment 3

Revisions to Section 2.2 "Neutron Fluence Methodology" of WCAP-14040, Revision 3

2.2 NEUTRON FLUENCE METHODOLOGY

The methodology used to provide neutron exposure evaluations for the reactor pressure vessel is based on the requirements provided in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence."[6]. The vessel exposure projections are based on the results of plant specific neutron transport calculations that are validated by benchmarking of the analytical approach, comparison with industry wide power reactor data bases, and finally, by comparison to plant specific surveillance capsule and reactor cavity dosimetry data. In the validation process, the measurement data are used solely to confirm the accuracy of the transport calculations. The measurements are not used in any way to modify the results of the transport calculations.

2.2.1 Plant Specific Transport Calculations

In the application of the methodology to the fast neutron exposure evaluations for the surveillance capsules and reactor vessel, plant specific forward transport calculations are carried out on a fuel cycle specific basis using the following three-dimensional flux synthesis technique:

$$\phi(r,\theta,z) = [\phi(r,\theta)] * [\phi(r,z)]/[\phi(r)]$$

where:

$\phi(r,\theta,z)$ is the synthesized three-dimensional neutron flux distribution,

$\phi(r,\theta)$ is the transport solution in r,θ geometry,

$\phi(r,z)$ is the two-dimensional solution for a cylindrical reactor model using the actual axial core power distribution, and

$\phi(r)$ is the one-dimensional solution for a cylindrical reactor model using the same source per unit height as that used in the r,θ two-dimensional calculation.

All of the transport calculations are carried out using the DORT discrete ordinates code Version 3.1[7] and the BUGLE-96 cross-section library[11]. The BUGLE-96 library provides a 67 group coupled neutron-gamma ray cross-section data set produced specifically for light water reactor application. In these analyses, anisotropic scattering is treated with a P_5 legendre expansion and the angular discretization is modeled with an S_{16} order of angular quadrature. Energy and space dependent core power distributions as well as system operating temperatures are treated on a fuel cycle specific basis. The synthesis procedure combining the $\phi(r,\theta)$, $\phi(r,z)$, and $\phi(r)$ transport solutions into the three-dimensional flux/fluence maps within the reactor geometry is accomplished by post-processing the output files generated by the [r,θ], [r,z], and [r] DORT calculations.

In some extreme cases where part length poisons or shielded fuel assemblies have been inserted into the reactor core to reduce the fluence locally in the vicinity of key vessel materials, the calculational approach may be modified to use either a multi-channel synthesis approach or a fully three-dimensional technique. For the full three-dimensional analysis, the TORT[7] three-dimensional discrete ordinates transport code is used in conjunction with either the BUGLE-96 ENDF/B-VI based library to provide a complete solution without recourse to the use of flux synthesis techniques.

In developing an analytical model of the reactor geometry, nominal design dimensions are normally employed for the various structural components. In some cases as-built dimensions are available; and, in those instances, the more accurate as-built data are used for model development. However, for the most part, as built dimensions of the components in the beltline region of the reactor are not available, thus, dictating the use of design dimensions. Likewise, water temperatures and, hence, coolant density in the reactor core and downcomer regions of the reactor are normally taken to be representative of full power operating conditions. The reactor core itself is treated as a homogeneous mixture of fuel, cladding, water, and miscellaneous core structures such as fuel assembly grids, guide tubes, etc.

The spatial mesh description used in the transport models depends on the overall size of the reactor and on the complexity required to model the core periphery, the in-vessel surveillance capsules, and the details of the reactor cavity. Mesh sizes are chosen to assure that proper convergence of the inner iterations is achieved on a pointwise basis. The pointwise inner iteration flux convergence criterion utilized in the r,θ calculations is set at a value of 0.001.

The mesh selection process results in a smaller spatial mesh in regions exhibiting steep gradients, in material zones of high cross-section (Σ_t), and at material interfaces. In the modeling of in-vessel surveillance capsules, a minimum set of 3 radial by 3 azimuthal mesh are employed within the test specimen array to assure that sufficient information is produced for use in the assessment of fluence gradients within the materials test specimens, as well as in the determination of gradient corrections for neutron sensors. Additional radial and azimuthal mesh are employed to model the capsule structure surrounding the materials test specimen array. In modeling the stainless steel baffle region at the periphery of the core, a relatively fine spatial mesh is required to adequately describe this rectilinear component in r,θ geometry. In performing this x,y to r,θ transition, care is taken to preserve both the thickness and volume of the steel region in order to accurately address the shielding effectiveness of the component.

The spatial variation of the neutron source is generally obtained from a burnup weighted average of the respective power distributions from individual fuel cycles. These spatial distributions include pinwise gradients for all fuel assemblies located at the periphery of the core and typically include a uniform or flat distribution for fuel assemblies interior to the core. The spatial component of the neutron source is transposed from x,y to [r,θ], [r,z], and [r] geometry by overlaying the mesh schematic to be used in the transport calculation on the pin by pin array and then computing the appropriate relative source applicable to each spatial interval within the reactor core.

These x,y to [r,θ], [r,z], and [r] transpositions are accomplished by first defining a fine mesh working array. The sizes of the fine mesh are usually chosen so that there is at least a 10x10 array of fine mesh over the area of each fuel pin at the core periphery. The coordinates of the center of each fine mesh interval and its associated relative source strength are assigned to the fine mesh based on the pin that is coincident with the center of the fine mesh. In the limit as the sizes of the fine mesh approach zero, this technique becomes an exact transformation.

Each space mesh in the transport geometry is checked to determine if it lies totally within the area of a particular fine working mesh. If it does, the relative source of that fine mesh is assigned to the transport space mesh. If, on the other hand, the transport space mesh covers a part of one or more fine mesh, then the relative source assigned to the transport mesh is determined by an area weighting process as follows:

$$P_m = \frac{\sum_i A_i P_i}{\sum_i A_i}$$

where:

P_m = the relative source assigned to transport mesh m.

A_i = the area of fine working mesh i within transport mesh m.

P_i = the relative source within fine working mesh i.

The energy distribution of the source is determined on a fuel assembly specific basis by selecting a fuel assembly burnup representative of conditions averaged over each fuel cycle and an initial enrichment characteristic for each assembly. From this average burnup and initial enrichment, a fission split by isotope including ^{235}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, and ^{241}Pu is derived; and, from that fission split, composite values of energy release per fission, neutron yield per fission, and fission spectrum are determined for each fuel assembly. These composite values are then combined with the spatial distribution to produce the overall absolute neutron source for use in the transport calculations.

2.2.2 Validation of the Transport Calculations

The validation of the methodology described in Section 2.2.1 is based on the guidance provided in Regulatory Guide 1.190. In particular, the validation consists of the following stages:

1. Comparisons of calculations with benchmark measurements from the Pool Critical Assembly (PCA) simulator at the Oak Ridge National Laboratory (ORNL)[12].

2. Comparisons of calculations with surveillance capsule and reactor cavity measurements from the H. B. Robinson power reactor benchmark experiment[22].

3. An analytical sensitivity study addressing the uncertainty components resulting from important input parameters applicable to the plant specific transport calculations used in the exposure assessments.

4. Comparisons of calculations with a measurements data base obtained from a large number of surveillance capsules withdrawn from a variety of pressurized water reactors.

At each subsequent application of the methodology, comparisons are made with plant specific dosimetry results to demonstrate that the plant specific transport calculations are consistent with the uncertainties derived from the methods qualification.

The first stage of the methods validation addresses the adequacy of basic transport calculation and dosimetry evaluation techniques and associated cross-sections. This stage, however, does not test the accuracy of commercial core neutron source calculations nor does it address uncertainties in operational or geometric variables that impact power reactor calculations. The second stage of the validation addresses uncertainties that are primarily methods related and would tend to apply generically to all fast neutron exposure evaluations. The third stage of the validation identifies the potential uncertainties introduced into the overall evaluation due to calculational methods approximations, as well as to a lack of knowledge relative to various plant specific parameters. The overall calculational uncertainty is established from the results of these three stages of the validation process.

The following summarizes the uncertainties determined from the results of the first three stages of the validation process:

PCA Benchmark Comparisons	3%
H. B. Robinson Benchmark Comparisons	3%
Analytical Sensitivity Studies	11%
Internals Dimensions	3%
Vessel Inner Radius	5%
Water Temperature	4%
Peripheral Assembly Source Strength	5%
Axial Power Distribution	5%
Peripheral Assembly Burnup	2%
Spatial Distribution of the Source	4%
Other Factors	5%

The category designated "Other Factors" is intended to attribute an additional uncertainty to other geometrical or operational variables that individually have an insignificant impact on the overall uncertainty, but collectively should be accounted for in the assessment.

The uncertainty components tabulated above represent percent uncertainty at the 1σ level. In the tabulation, the net uncertainty of 11% from the analytical sensitivity studies has been broken down into its individual components. When the four uncertainty values listed above (3%, 3%, 11%, and 5%) are combined in quadrature, the resultant overall 1σ calculational uncertainty is estimated to be 13%.

To date the methodology described in Section 2.2.1 coupled with the BUGLE-96 cross-section library has been used in the evaluation of dosimetry sets from 82 surveillance capsules from 23 pressurized water reactors. These capsule withdrawals included 2-5 capsules from individual reactors. The comparisons of the plant specific calculations with the results of the capsule dosimetry are used to further validate the calculational methodology within the context of a 1σ calculational uncertainty of 13%.

This 82 capsule data base includes all surveillance capsule dosimetry sets analyzed by Westinghouse using the Bugle-96 cross-section library and the synthesis approach described in Section 2.2.1. No surveillance capsule dosimetry sets were excluded from the M/C data base. As additional capsules are

analyzed using the synthesis approach with the BUGLE-96 cross-section library the M/C comparisons will be added to the database.

The comparisons between the plant specific calculations and the data base measurements are provided on two levels. In the first instance, measurement to calculation (M/C) ratios for each fast neutron sensor reaction rate from the surveillance capsule irradiations are listed. This tabulation provides a direct comparison, on an absolute basis, of measurement and calculation. The results of this comparison for the surveillance capsule data base are as follows:

REACTION	M/C	STD DEV
$^{63}Cu(n\alpha)^{60}Co$	1.09	7.9%
$^{54}Fe(n,p)^{54}Mn$	0.99	8.4%
$^{58}Ni(n,p)^{58}Co$	0.99	8.9%
$^{238}U(n,f)^{137}Cs$	1.01	11.8%
$^{237}Np(n,f)^{137}Cs$	1.06	11.3%
Linear Average	1.03	9.8%

These comparisons show that the calculations and measurements for the surveillance capsule data base fall well within the 13% calculational uncertainty for all of the fast neutron reactions.

The second comparison of calculations with the data base is based on the least squares adjustment of the individual surveillance capsule data sets. The least squares adjustment procedure provides a weighting of the individual sensor measurements based on spectral coverage and allows a comparison of the neutron flux ($E > 1.0$ MeV) before and after adjustment. The neutron flux/fluence ($E > 1.0$ MeV) is the primary parameter of interest in the overall pressure vessel exposure evaluations.

The least squares evaluations of the 82 surveillance capsule dosimetry sets followed the guidance provided in Section 1.4.2 of Regulatory Guide 1.190 and in ASTM Standard E944-96, "Standard Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance."

The application of the least squares methodology requires the following input:

1. The calculated neutron energy spectrum and associated uncertainties at the measurement location.

2. The measured reaction rates and associated uncertainty for each sensor contained in the multiple foil set.

3. The energy dependent dosimetry reaction cross-sections and associated uncertainties for each sensor contained in the multiple foil sensor set.

For the data base comparisons, the calculated neutron spectra were obtained from the results of plant specific neutron transport calculations applicable to each of the 82 surveillance capsules. The sensor reaction rates and dosimetry cross-sections were the same as those used in the direct M/C comparisons noted above.

The results of this latter comparison expressed in terms of the ratio of adjusted flux to calculated flux (A/C) are summarized as follows for the 82 capsule data base:

PARAMETER	A/C	STD DEV
$\phi(E > 1.0\ MeV)$	1.00	7.3%

As with the comparisons based on the linear average of reaction rate M/C ratios, the comparisons of the least squares adjusted results with the plant specific transport calculations demonstrate that the calculated results are essentially unbiased with an uncertainty well within the 20% criterion established in Regulatory Guide 1.190.

2.3 FRACTURE TOUGHNESS PROPERTIES

The fracture toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the requirements of Appendix G, 10 CFR Part 50[4], as augmented by the additional requirements in subsection NB-2331 of Section III of the ASME B&PV Code[8]. These fracture toughness requirements are also summarized in Branch Technical Position MTEB 5-2 ("Fracture Toughness Requirements")[9] of the NRC Regulatory Standard Review Plan.

These requirements are used to determine the value of the reference nil-ductility transition temperature (RT_{NDT}) for unirradiated material (defined as initial RT_{NDT}, IRT_{NDT}) and to calculate the adjusted reference temperature (ART) as described in Section 2.4. Two types of tests are required to determine a material's value of IRT_{NDT}: Charpy V-notch impact (C_v) tests and drop-weight tests. The procedure is as follows:

1. Determine a temperature T_{NDT} that is at or above the nil-ductility transition temperature by drop weight tests.

2. At a temperature not greater than T_{NDT} + 60°F, each specimen of the C_v test shall exhibit at least 35 mils lateral expansion and not less than 50 ft-lb absorbed energy. When these requirements are met, T_{NDT} is the reference temperature RT_{NDT}.

3. If the requirements of (2) above are not met, conduct additional C_v tests in groups of three specimens to determine the temperature T_{Cv} at which they are met. In this case the reference temperature $RT_{NDT} = T_{Cv}$ - 60°F. Thus, the reference temperature RT_{NDT} is the higher of T_{NDT} and (T_{Cv} - 60°F).

4. If the C_v test has not been performed at T_{NDT} + 60°F, or when the C_v test at T_{NDT} + 60°F does not exhibit a minimum of 50 ft-lb and 35 mils lateral expansion, a temperature representing a minimum of 50 ft-lb and 35 mils lateral expansion may be obtained from a full C_v impact curve developed from the minimum data points of all the C_v tests performed as shown in Figure 2.1.

Plants that do not follow the fracture toughness guidelines in Branch Technical Position MTEB 5-2 to determine IRT_{NDT} can use alternative procedures. However, sufficient technical justification and special circumstances per the criteria of 10CFR50.12(a)(2) must be provided for an exemption from the regulations to be granted by the NRC.

4.0 REFERENCES

1. NUREG 1431, "Standard Technical Specifications for Westinghouse Plants," Revision 2.

2. U.S. Nuclear Regulatory Commission, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," Generic Letter 88-16, October, 1988.

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

4. Code of Federal Regulations, Title 10, Part 50, "Fracture Toughness Requirements for Light-Water Nuclear Power Reactors," Appendix G, Fracture Toughness Requirements.

5. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Appendix G, Fracture Toughness Criteria For Protection Against Failure.

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

7. RSICC Computer Code Collection CCC-650, "DOORS 3.1, One-, Two-, and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System," August 1996.

8. ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Power Plant Components," Division 1, Subsection NB: Class 1 Components.

9. Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements," NUREG-0800 Standard Review Plan 5.3.2, Pressure-Temperature Limits, July 1981, Rev. 1.

10. ASTM E-208, Standard Test Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels, ASTM Standards, Section 3, American Society for Testing and Materials.

11. RSICC Data Library Collection DLC-185, "BUGLE-96, Coupled 47 Neutron, 20 Gamma-Ray Group Cross-Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications," March 1996.

12. I. Remec and F. B. K. Kam, "Pool Critical Assembly Pressure Vessel Benchmark," NUREG/CR-6454 (ORNL/TM-13205), July 1997.

13. Code of Federal Regulations, Title 10, Part 50, "Fracture Toughness Requirements for Light-Water Nuclear Power Reactors," Appendix H, Reactor Vessel Material Surveillance Program Requirements.

14. Timoshenko, S. P. and Goodier, J. N., Theory of Elasticity, Third Edition, McGraw-Hill Book Co., New York, 1970.

15. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Appendix A, Analysis of Flaws, Article A3000, Method For K_I Determination.

16. WRC Bulletin No. 175, "PVRC Recommendations on Toughness Requirements for Ferritic Materials," Welding Research Council, New York, August 1972.

17. ASME Boiler and Pressure Vessel Code Case N-514, Section X1, Division 1, "Low Temperature Overpressure Protection," Approval date: February 12, 1992.

18. Branch Technical Position RSB 5-2, "Overpressurization Protection of Pressurized Water Reactors While Operating at Low Temperatures," NUREG-0800 Standard Review Plan 5.2.2, Overpressure Protection, November 1988, Rev. 2.

19. ASME Boiler and Pressure Vessel Code Case N640, Section XI, Division 1, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves," February 26, 1996.

20. ASME Boiler and Pressure Vessel Code Case N588, Section XI, Division 1, "Alternative to Reference Flow Orientation of Appendix G for Circumferential Welds in Reactor Vessels," December 12, 1997.

21. ASME Boiler and Pressure Vessel Code Case N641, Section XI, Division 1, "Alternative Pressure-Temperature Relationship and Low Temperature Overpressure Protection System Requirements," January 17, 2000.

22. I. Remec and F. B. K. Kam, "H. B. Robinson Pressure Vessel Benchmark," NUREG/CR-6453 (ORNL/TM-13204), February 1998.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 18, 2003

Mr. Gordon Bischoff, Project Manager
Westinghouse Owners Group
Westinghouse Electric Company
Mail Stop ECE 5-16
P.O. Box 355
Pittsburgh, PA 15230-0355

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION - WCAP-14040, REVISION 3, "METHODOLOGY USED TO DEVELOP COLD OVERPRESSURE MITIGATING SYSTEM SETPOINTS AND RCS HEATUP AND COOLDOWN CURVES" (TAC NO. MB5754)

Dear Mr. Bischoff:

By letter dated May 23, 2002, the Westinghouse Owners Group submitted for staff review Topical Report WCAP-14040, Revision 3, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Curves." The staff has completed its preliminary review of WCAP-14040, Revision 3, and has identified a number of items for which additional information is needed to continue its review. This was discussed in a telephone conversation with Mr. Ken Vavrek of your staff on June 5, 2003, and it was agreed that a response would be provided within 30 days of receipt of this letter.

If you have any questions, please call me at (301) 415-1436.

Sincerely,

Drew Holland, Project Manager, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 694

Enclosure: Request for Additional Information

cc w/encl: See next page

RECEIVED
JUN 24 2003
H. A. SEPP, JR.

Westinghouse Owners Group Project No. 694

cc:
Mr. H. A. Sepp, Manager
Regulatory and Licensing Engineering
Westinghouse Electric Company
P.O. Box 355
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REQUEST FOR ADDITIONAL INFORMATION

WCAP-14040, REVISION 3, "METHODOLOGY USED TO DEVELOP COLD OVERPRESSURE MITIGATING SYSTEM SETPOINTS AND RCS HEATUP AND COOLDOWN CURVES"

WESTINGHOUSE OWNERS GROUP

PROJECT NO. 694

Please address the following NRC staff issues pertaining to the review of this topical report.

1. Section 2.3, page 2-5, Branch Technical Position MTEB 5-2 does not give fracture toughness "requirements." Revise WCAP-14040, Revision 3, to refer to the information in MTEB 5-2 as "guidelines" rather than "requirements."

2. Section 2.4, page 2-6, when referring to the "A_i" term in Equation 2.4-3, revise your definition which refers to it as the "measured value of ΔRT_{NDT}" – instead call it the "measured shift in the Charpy V-notch 30 ft-lb energy level between the unirradiated condition and the irradiated condition, f_i."

3. Section 2.4, page 2-7, revise the sentence which reads, "[i]f the measured value exceeds the predicted value ($\Delta RT_{NDT} + 2\sigma_{\Delta}$), a supplement to the PTLR must be provided to demonstrate how the results affect the approved methodology," to state "[i]f the measured value exceeds the predicted value ($\Delta RT_{NDT} + 2\sigma_{\Delta}$), a supplement to the PTLR methodology must be provided for NRC staff review and approval to demonstrate how the results affect the approved methodology."

4. Section 2.5, page 2-7, it is stated that K_{Ia} is the reference fracture toughness curve in Appendix G to Section XI of the ASME Code. Clarify this to note that this refers to Editions of the Code through the 1995 Edition/1996 Addenda. The most recent Edition and Addenda of the Code (1998 Edition through 2000 Addenda) incorporated by reference into 10 CFR 50.55a, however, uses K_{Ic} as the reference fracture toughness curve.

5. Section 2.5, page 2-8, the "note" regarding the use of a 1.223 vs. 1.233 coefficient in the K_{Ia} equation is meaningless and confusing unless one also explains that there was a typographical error in the 1989 Edition of Section XI, Appendix G (i.e., where the 1.233 was used). Revise WCAP-14040, Revision 3, to either eliminate this note or revise the note to offer additional explanation regarding the historical basis for the 1.223 vs. 1.233 issue.

6. Section 2.5, page 2-8, when discussing ASME Code Case N-640, it is not correct to say that an exemption is required to implement N-640 because the NRC has not "endorsed" the Code Case. "Endorsement" implies that it has been included in Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability -- ASME Section XI, Division 1." Code Case N-640 would have to be included in the edition of the ASME Code which the licensee has adopted in their facility's licensing basis in order to comply with 10 CFR 50.55a before an exemption is no longer required.

- 2 -

7. The statement in Section 2.5, page 2-10, regarding need for an exemption relative to modifying existing 10 CFR Part 50, Appendix G flange requirements should, for consistency be repeated in Section 2.8.

8. Section 2.6.1, page 2-12, it is stated "[t]hese stress components are used for determining the thermal stress intensity factors, K_{It}, as described in the following subsection." The following subsection is 2.6.2, "Steady-State Analyses," and it does not address the calculation of K_{It}. Revise WCAP-14040, Revision 3, to address this apparent inconsistency.

9. Section 2.6.2, page 2-14, and Section 2.6.5, page 2-15, M_m factors of 1.84, 0.918, and 3.18 are given for various reactor pressure vessel wall thickness ranges to be used when steady-state analyses are performed. It is unclear as to where these M_m factors come from (unable to locate them in any edition of ASME Section XI, Appendix G). Further, they are not consistent with what should be the same M_m factors cited on page 2-15. Revise WCAP-14040, Revision 3, to address this apparent inconsistency in the cited M_m factors.

10. Section 2.7, page 2-19, it should be noted that an exemption is required when a licensee wishes to make use of ASME Code Case N-588. Revise WCAP-14040, Revision 3, accordingly.

OG-02-018
May 23, 2002

WCAP-14040, Rev. 3
Project Number 694

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

Attention: Chief, Information Management Branch,
Division of Inspection and Support Programs

Subject: Westinghouse Owners Group
<u>Transmittal of WCAP-14040, Rev. 3, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," (MUHP-3073)</u>

Reference: 1) Westinghouse Owners Group Letter, R. Bryan to Document Control Desk, "Transmittal of WCAP-15315, Rev. 1, 'Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants'," OG-02-019, May 23, 2002.

This letter transmits five copies of the WCAP-14040, Rev. 3, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," for NRC review and approval. WCAP-14040-A, Rev. 2, was approved by the NRC on October 16, 1995, and contains a methodology for developing Reactor Coolant System (RCS) Pressure-Temperature (P-T) limit curves and Cold Overpressure Mitigation System (COMS) setpoints and enable temperature that can be referenced by licensees in the Administrative Controls Section of the Technical Specifications when relocating P-T limit curves, COMS setpoints and COMS enable temperature to a Pressure and Temperature Limits Report (PTLR).

Several ASME Nuclear Code Cases (N-588, N-640, and N-641) associated with the development of P-T limit curves and the COMS enable temperature have been approved by the ASME subsequent to the approval of WCAP-14040-NP-A, Rev. 2 in October 1995. Exemption requests have been approved by the NRC to allow the use of these ASME Nuclear Code Cases in the development of P-T limit curves.

WCAP-14040, Rev. 3 has been revised to incorporate these approved ASME Nuclear Code Cases into the methodology used to develop the P-T limit curves and COMS enable temperature that is contained in WCAP-NP-A, Rev. 2.

OG-02-018
May 23, 2002

WCAP-14040, Rev. 3 also contains an option to develop the P-T limit curves without the flange requirement, currently required by 10CFR50 Appendix G. The option to develop P-T limit curves without the flange requirement would require NRC approval of an exemption request, or rulemaking to eliminate the requirement. A Petition for Rulemaking to eliminate the flange requirement of 10CFR50 Appendix G from the P-T limit curves was submitted by Westinghouse Electric Co. in November 1999.

The technical justification for eliminating the flange requirement is contained in WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants," Rev. 0. WCAP-15315, Rev. 0 was submitted to the NRC with the Petition for Rulemaking to eliminate the flange requirement of 10CFR50 Appendix G by Westinghouse Electric Co., in November 1999. WCAP-15315, Rev. 1 contains the additional information for eliminating the flange requirement as requested by the NRC during a meeting between Westinghouse and the NRC on August 28, 2001. WCAP-15315, Rev. 1 is also being submitted for NRC review as justification for eliminating the flange requirement of 10CFR50 Appendix G (Reference 1).

The WOG is submitting WCAP-14040, Rev. 3 under the NRC licensing topical report program for review and acceptance for referencing in licensing actions. The objective is that once approved, each WOG member can reference a single methodology in the Administrative Controls Section of the Technical Specifications when relocating or revising P-T limit curves and COMS setpoints and enable temperature in a PTLR.

The WOG requests that the NRC complete the review of WCAP-14040, Rev. 3, by September 30, 2002. Consistent with the Office of Nuclear Reactor Regulation, Office Instruction LIC-500, "Processing Request for Reviews of Topical Reports," the WOG requests that the NRC provide an estimate of the review hours, and target dates for any Request(s) for Additional Information and for completion of the Safety Evaluation for WCAP-14040, Rev. 3.

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OG-02-018
May 23, 2002

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Very truly yours,

Robert H Bryan

Robert H. Bryan, Chairman
Westinghouse Owners Group

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OG-02-018
May 23, 2002

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May 23, 2002

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Attachment 4

<u>WCAP-18363-NP, REVISION 1, "NORTH ANNA UNITS 1 AND 2 HEATUP AND COOLDOWN LIMIT CURVES FOR NORMAL OPERATION"</u>

Virginia Electric and Power Company
(Dominion Energy Virginia)
North Anna Power Station Units 1 and 2

WCAP-18363-NP
Revision 1

March 2020

North Anna Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation

Westinghouse

WCAP-18363-NP
Revision 1

North Anna Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation

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RECORD OF REVISION

Revision 0: Original Issue

Revision 1: Section 7 was revised to address the reevaluation of the instrument uncertainties and pressure corrections. Appendix K was added to provide justification for the use of PWROG-17090-NP-A, consistent with the NRC Safety Evaluation.

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EXECUTIVE SUMMARY

This report provides the methodology and results of the generation of heatup and cooldown pressure-temperature (P-T) limit curves for normal operation of the North Anna Units 1 and 2 reactor vessels through the Subsequent License Renewal (SLR) period of operation, also known as the Subsequent Period of Extended Operation (SPEO). The heatup and cooldown P-T limit curves were generated using the limiting Adjusted Reference Temperature (ART) values for North Anna Units 1 and 2. The limiting ART values were those of the North Anna Unit 2 Lower Shell Forging 03 at both the 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations. Note that the limiting material for the current EOLE P-T limit curves contained in the Technical Specifications is also North Anna Unit 2 Lower Shell Forging 03.

The P-T limit curves were generated for 72 effective full-power years (EFPY) using the K_{Ic} methodology detailed in the 1998 Edition through 2000 Addenda of the ASME Code, Section XI, Appendix G. The P-T limit curve generation methodology is consistent with the NRC-approved methodology documented in WCAP-14040-A, Revision 4. Heatup rates of 20, 40, and 60°F/hr, and cooldown rates of -100, -60, -40, -20, and 0°F/hr (steady-state) were used to generate the P-T limit curves, with the flange requirements and without margins for instrumentation errors. The North Anna Units 1 and 2 SLR period of operation corresponding to 80 years of operation is 72 EFPY. The SLR P-T limit curves can be found in Figures 6-1 and 6-2. As concluded in Section 7, the new 72 EFPY P-T limit curves are bounded by the current North Anna Power Station P-T limit curves. Thus, continued use of the current North Anna Power Station P-T limit curves is justified through 72 EFPY.

Appendix A contains the thermal stress intensity factors for the maximum heatup and cooldown rates at 72 EFPY based on the Section 6 P-T limit curves.

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

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

Appendix D contains the determination of the Low Temperature Overpressure Protection (LTOP) system minimum enable temperature at 72 EFPY.

Appendix E contains an evaluation of the North Anna Units 1 and 2 surveillance data credibility.

Appendix F contains an evaluation of the North Anna Units 1 and 2 Upper-Shelf Energy (USE) at 72 EFPY.

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

Appendix H contains an evaluation of Master Curve data relevant to North Anna Unit 1 Lower Shell Forging 03.

Appendix I contains a summary of the North Anna licensing basis related to selection of chemistry factors (CFs) when surveillance data is available.

Appendix J contains the current Technical Specifications P-T limit curves (referred to herein as the "current" P-T limits).

Appendix K contains the justification for the use of PWROG-17090-NP-A per the stipulations in the NRC's Safety Evaluation.

1 INTRODUCTION

The purpose of this report is to present the calculations and the development of the North Anna Units 1 and 2 heatup and cooldown P-T limit curves for 72 EFPY. This report documents the calculated Adjusted Reference Temperature (ART) values and the development of the P-T limit curves for normal operation through 80 years of operation.

Heatup and cooldown P-T limit curves are calculated using the adjusted RT_{NDT} (reference nil-ductility temperature) corresponding to the limiting beltline region material of the reactor vessel. The adjusted RT_{NDT} of the limiting material in the core region of the reactor vessel is determined by using the unirradiated reactor vessel material fracture toughness properties, estimating the radiation-induced ΔRT_{NDT}, and adding a margin. The unirradiated RT_{NDT} ($RT_{NDT(U)}$) values were redefined in PWROG-18005-NP (Reference 9) to take advantage of the most up-to-date methodologies and data available; therefore, the values utilized herein supersede those utilized in the previous P-T limit curves developed in WCAP-15112 (Reference 17). The redefined $RT_{NDT(U)}$ is designated as the higher of either the drop weight nil-ductility transition temperature (NDTT) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (normal to the major working direction) minus 60°F. In instances where insufficient data is available to determine $RT_{NDT(U)}$ using ASME Code methods, alternate estimation methods such as Branch Technical Position (BTP) 5-3 are applied.

RT_{NDT} increases as the material is exposed to fast-neutron radiation. Therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the unirradiated RT_{NDT}. The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and nickel) present in reactor vessel steel. The U.S. Nuclear Regulatory Commission (NRC) has published a method for predicting radiation embrittlement in Regulatory Guide 1.99, Revision 2 (Reference 1). Regulatory Guide 1.99, Revision 2 is used for the calculation of ART values ($RT_{NDT(U)}$ + ΔRT_{NDT} + margins for uncertainties) at the 1/4T and 3/4T locations, where T is the thickness of the vessel at the beltline region measured from the clad/base metal interface. The calculated ART values for 72 EFPY are documented in Section 5 of this report. The fluence projections used in calculation of the ART values are taken from WCAP-18015-NP (Reference 8) which identifies the materials projected to exceed a neutron fluence of 1.0E+17 n/cm^2 (E > 1.0 MeV) including the inlet and outlet nozzle forgings, as applicable. A description of the fluence analysis is provided in Section 2 of this report.

The heatup and cooldown P-T limit curves documented in this report were generated using the most limiting ART values and the NRC-approved methodology documented in WCAP-14040-A, Revision 4 (Reference 2). Specifically, the "Axial Flaw" methodology of the 1998 Edition through 2000 Addenda of ASME Code, Section XI, Appendix G (Reference 3) was used, which makes use of the K_{Ic} methodology. The K_{Ic} curve is a lower bound static fracture toughness curve obtained from test data gathered from several different heats of pressure vessel steel. The limiting material is indexed to the K_{Ic} curve so that allowable stress intensity factors can be obtained for the material as a function of temperature. Allowable operating limits are then determined using the allowable stress intensity factors.

The P-T limit curves presented herein were generated without instrumentation errors. The North Anna Power Station Technical Specifications (Reference 18) P-T limit curves are currently provided with instrumentation errors as discussed in Appendix J. The reactor vessel flange requirements of 10 CFR 50, Appendix G (Reference 4) have been incorporated in the P-T limit curves. The P-T limit curves generated

in Section 6 are compared to the current North Anna Units 1 and 2 P-T limit curves, contained in the North Anna Power Station Technical Specifications (Reference 18), in Section 7 to determine if adequate margin exists to justify continued use of the North Anna Units 1 and 2 current P-T limits through the Subsequent License Renewal (SLR) period of operation.

The P-T limit curves generated in Section 6 bound the P-T limit curves for the reactor vessel inlet and outlet nozzles generated in Appendix B for North Anna Units 1 and 2 at 72 EFPY. Discussion of the other ferritic RCPB components relative to P-T limits is contained in Appendix C. Appendix D contains a calculation of the Low Temperature Overpressure Protection (LTOP) system enable temperature. Appendix E provides a credibility evaluation of the North Anna Units 1 and 2 surveillance data. Appendix F contains an evaluation of the North Anna Units 1 and 2 Upper-Shelf Energy (USE) values at 72 EFPY. Appendix G contains a comparison of the material property input values used in this evaluation and those used in past evaluations as well as the Updated Final Safety Analysis Report (UFSAR). Appendix H contains an evaluation of Master Curve data relevant to North Anna Unit 1 Lower Shell Forging 03. Appendix I contains a summary of the North Anna licensing basis related to selection of CFs when surveillance data is available. Appendix J contains the current Technical Specifications P-T limit curves (referred to herein as the "current" P-T limits). Appendix K contains the justification for the use of PWROG-17090-NP-A per the stipulations in the NRC's Safety Evaluation.

2 CALCULATED NEUTRON FLUENCE

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

Estimated RPV beltline and extended beltline fast neutron (E > 1.0 MeV) fluences at the end of 80 years of operation were calculated for North Anna Units 1 and 2. The analyses methodologies used to calculate the North Anna Units 1 and 2 RPV fluences satisfy the guidance set forth in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" (Reference 5). These methodologies have been approved by the U.S. NRC for the beltline region, i.e. materials directly surrounding the core and adjacent materials per 10 CFR 50, Appendix G (Reference 4), which are projected to experience the highest fluence. The methodologies, along with the NRC safety evaluation, are contained in detail in WCAP-14040-A (Reference 2). For North Anna Units 1 and 2, the beltline region has traditionally included the upper, intermediate, and lower shell forgings, and the circumferential welds between these components. Note that while a consistent approach is applied to the extended beltline, there is, at present, no generically-approved methodology for performing neutron fluence evaluations of the reactor vessel extended beltline. The traditional beltline and extended beltline materials are identified by heat numbers in Tables 2-3 and 2-4 and Figures 2-1 and 2-2.

Materials exceeding a fast neutron (E > 1.0 MeV) fluence of 1.0×10^{17} n/cm^2 at the end of the SLR period are evaluated for changes in fracture toughness. RPV materials that are not traditionally plant-limiting because of low levels of neutron radiation must now be evaluated to determine the accumulated fluence at SLR. Therefore, fast neutron (E > 1.0 MeV) fluence calculations were performed for the North Anna Units 1 and 2 RPV circumferential welds (lower shell to lower vessel head, intermediate shell to lower shell, and upper shell to intermediate shell), centerline of the inlet and outlet nozzle forging to vessel shell welds at the lowest extent, 1/4T flaw location in the inlet and outlet nozzle, and forgings (lower shell, intermediate shell, and upper shell), to determine if they will exceed a fast neutron (E > 1.0 MeV) fluence of 1.0×10^{17} n/cm^2 at SLR. The materials that exceed the 1.0×10^{17} n/cm^2 fast neutron (E > 1.0 MeV) fluence threshold, and were not evaluated in past analyses of record as part of the traditional beltline, are referred to as extended beltline materials in this report and are evaluated to determine the effect of neutron irradiation embrittlement during the SLR period. The need to evaluate these extended beltline material was previously identified during Dominion submittal and NRC review of the P-T limit curves with vacuum refill (Reference 19).

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

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

where $\varphi(r,\theta,z)$ is the synthesized 3D neutron fluence rate distribution, $\varphi(r,\theta)$ is the transport solution in r,θ geometry, $\varphi(r,z)$ is the two-dimensional solution for a cylindrical reactor model using the actual axial core power distribution, and $\varphi(r)$ is the one-dimensional solution for a cylindrical reactor model using the

same source per unit height as that used in the r,θ two-dimensional calculation. This synthesis procedure was carried out for each operating cycle at North Anna Units 1 and 2.

All of the transport calculations were carried out using the DORT discrete ordinates code (Reference 6) with the BUGLE-96 cross-section library (Reference 7). The BUGLE-96 library provides a coupled 47-neutron-, 20-gamma-ray-group cross-section data set produced specifically for light water reactor applications. In these analyses, anisotropic scattering was treated with a P_5 Legendre expansion and the angular discretization was modeled with an S_{16} order of angular quadrature. Energy- and space-dependent core power distributions, as well as system operating temperatures, were treated on a fuel-cycle-specific basis.

The calculations for fuel Cycles 1 through 24 for North Anna Unit 1 and fuel Cycles 1 through 23 for North Anna Unit 2 determine the neutron exposure of the pressure vessel and surveillance capsules based on completed fuel cycles. For North Anna Unit 1, projections for Cycle 25 and beyond, up to and including EOLE (50.3 EFPY) and SLR (conservatively set to 72 EFPY), were based on the uprated core power level of 2940 MWt and the uprated Cycle 24. For North Anna Unit 2, projections for Cycle 24 and beyond, up to and including EOLE (52.3 EFPY) and SLR (conservatively set to 72 EFPY), were based on the uprated core power level of 2940 MWt and the uprated Cycle 23. These projections are used to perform the reactor vessel integrity evaluation contained herein. Projected results will remain valid as long as future plant operation is consistent with these conservative inputs.

Table 2-1 gives the North Anna Unit 1 calculated fast neutron ($E > 1.0$ MeV) fluences at the capsule locations including all withdrawn surveillance capsules (Capsules V, U, and W). Table 2-2 gives the North Anna Unit 2 calculated fast neutron ($E > 1.0$ MeV) fluences at the capsule locations including all withdrawn surveillance capsules (Capsules V, U, and W). These fast neutron ($E > 1.0$ MeV) fluences were calculated using methodologies that follow the guidance of Regulatory Guide 1.190.

Selected results for the pressure vessel from the neutron transport analyses are provided in Tables 2-3 and 2-4 for North Anna Units 1 and 2, respectively. Calculated fast neutron ($E > 1.0$ MeV) fluence results for reactor vessel materials, on the pressure vessel clad/base metal interface, are provided for the nominal end of cycle (EOC) 24 for North Anna Unit 1 (29.7 EFPY) and nominal EOC 23 for North Anna Unit 2 (28.1 EFPY), as well as projected fluence results up to 72 EFPY, which corresponds to the North Anna Units 1 and 2 80-year plant life.

From Table 2-3 it is observed that one outlet nozzle and two inlet nozzles have fast neutron ($E > 1.0$ MeV) fluence greater than 1.0×10^{17} n/cm^2 at the nozzle forging to vessel shell weld centerline and one inlet nozzle has a fast neutron ($E > 1.0$ MeV) fluence greater than 1.0×10^{17} n/cm^2 at the postulated 1/4T nozzle flaw location at 72 EFPY for North Anna Unit 1. From Table 2-4, it is observed that one outlet nozzle and two inlet nozzles have fast neutron ($E > 1.0$ MeV) fluence greater than 1.0×10^{17} n/cm^2 at the nozzle forging to vessel shell weld centerline and one inlet nozzle has a fast neutron ($E > 1.0$ MeV) fluence greater than 1.0×10^{17} n/cm^2 at the postulated 1/4T nozzle flaw location at 72 EFPY for North Anna Unit 2. Tables 2-3 and 2-4 indicate that the lower shell to lower vessel head circumferential weld will remain below 1.0×10^{17} n/cm^2 through SLR for both North Anna Units 1 and 2.

Figure 2-1 shows the axial boundary of the 1.0×10^{17} n/cm^2 fluence threshold (at 50.3 EFPY and 72 EFPY) as a function of azimuthal position (Z versus θ) for North Anna Unit 1, whereas Figure 2-2 shows the same information (at 52.3 EFPY and 72 EFPY) for North Anna Unit 2. It is noted that the nozzle materials located above the nozzle centerline remain below 1.0×10^{17} n/cm^2 through 72 EFPY. Likewise, the lower shell to lower head circumferential weld remains out of the beltline region through 72 EFPY. The data used

to generate Figures 2-1 and 2-2 are tabulated in Appendices A and B of WCAP-18015-NP (Reference 8), respectively.

Table 2-1 Calculated Fast Neutron Fluence (E > 1.0 MeV) at the Surveillance Capsule Center for North Anna Unit 1[(a)]

Cycle	Cumulative Time [EFPY]	Surveillance Capsules [n/cm²]						Clad/Base Metal Interface
		15°	25°	35°	45°	35°/15°[(e)]	35°/25°[(f)]	
1	1.1	3.06E+18[(b)]	2.01E+18	1.37E+18	1.07E+18	1.37E+18	1.37E+18	1.90E+18
2	1.9	5.45E+18	3.54E+18	2.38E+18	1.84E+18	2.38E+18	2.38E+18	3.39E+18
3	2.9	7.65E+18	5.01E+18	3.34E+18	2.58E+18	3.34E+18	3.34E+18	4.78E+18
4	3.8	1.00E+19	6.39E+18	4.20E+18	3.24E+18	4.20E+18	4.20E+18	6.31E+18
5	4.8	1.18E+19	7.59E+18	5.03E+18	3.91E+18	5.03E+18	5.03E+18	7.42E+18
6	5.9	1.40E+19	9.14E+18[(c)]	6.06E+18	4.70E+18	6.06E+18	6.06E+18	8.75E+18
7	7.1	1.64E+19	1.08E+19	7.19E+18	5.58E+18	7.19E+18	7.19E+18	1.01E+19
8	8.4	1.89E+19	1.25E+19	8.41E+18	6.54E+18	8.41E+18	8.41E+18	1.14E+19
9	9.8	2.14E+19	1.42E+19	9.59E+18	7.46E+18	9.59E+18	9.59E+18	1.29E+19
10	11.1	2.37E+19	1.59E+19	1.08E+19	8.40E+18	1.08E+19	1.08E+19	1.41E+19
11	12.4	2.59E+19	1.75E+19	1.19E+19	9.33E+18	1.19E+19	1.19E+19	1.54E+19
12	13.5	2.79E+19	1.90E+19	1.29E+19	1.01E+19	1.29E+19	1.29E+19	1.65E+19
13	14.8	3.02E+19	2.05E+19[(d)]	1.40E+19	1.09E+19	1.40E+19	1.40E+19	1.77E+19
14	16.2	3.26E+19	2.23E+19	1.52E+19	1.19E+19	1.52E+19	1.52E+19	1.90E+19
15	17.5	3.50E+19	2.41E+19	1.65E+19	1.29E+19	1.76E+19	1.70E+19	2.03E+19
16	18.9	3.74E+19	2.58E+19	1.77E+19	1.38E+19	2.01E+19	1.88E+19	2.16E+19
17	20.2	3.98E+19	2.76E+19	1.89E+19	1.48E+19	2.24E+19	2.05E+19	2.28E+19
18	21.6	4.22E+19	2.94E+19	2.02E+19	1.58E+19	2.49E+19	2.23E+19	2.41E+19
19	23.0	4.47E+19	3.11E+19	2.14E+19	1.68E+19	2.73E+19	2.41E+19	2.55E+19
20	24.4	4.71E+19	3.29E+19	2.27E+19	1.78E+19	2.98E+19	2.58E+19	2.68E+19
21	25.8	4.95E+19	3.47E+19	2.39E+19	1.88E+19	3.21E+19	2.76E+19	2.81E+19
22	26.9	5.13E+19	3.60E+19	2.49E+19	1.96E+19	3.39E+19	2.89E+19	2.90E+19
23	28.3	5.36E+19	3.75E+19	2.60E+19	2.05E+19	3.62E+19	3.04E+19	3.03E+19
24	29.7	5.59E+19	3.92E+19	2.73E+19	2.16E+19	3.86E+19	3.22E+19	3.16E+19
Projected	50.3	9.11E+19	6.48E+19	4.64E+19	3.74E+19	7.37E+19	5.77E+19	5.13E+19
Projected	54.0	9.74E+19	6.94E+19	4.98E+19	4.03E+19	8.00E+19	6.23E+19	5.48E+19
Projected	72.0	1.28E+20	9.17E+19	6.65E+19	5.41E+19	1.11E+20	8.46E+19	7.20E+19

Notes:

(a) Information taken from WCAP-18015-NP (Reference 8).

(b) Capsule V was withdrawn at the end-of-cycle 1.

(c) Capsule U was withdrawn at the end-of-cycle 6.

(d) Capsule W was withdrawn at the end-of-cycle 13.

(e) Capsule Z was moved at the end-of-cycle 14.

(f) Capsule T was moved at the end-of-cycle 14.

Table 2-2 Calculated Fast Neutron Fluence (E > 1.0 MeV) at the Surveillance Capsule Center for North Anna Unit 2[a]

Cycle	Cumulative Time [EFPY]	Surveillance Capsules [n/cm²]						Clad/Base Metal Interface
		15°	25°	35°	45°	35°/15°[e]	35°/25°[f]	
1	1.0	2.86E+18[b]	1.87E+18	1.27E+18	9.96E+17	1.27E+18	1.27E+18	1.78E+18
2	1.6	4.68E+18	3.05E+18	2.06E+18	1.61E+18	2.06E+18	2.06E+18	2.92E+18
3	2.7	6.99E+18	4.86E+18	3.34E+18	2.62E+18	3.34E+18	3.34E+18	4.20E+18
4	3.8	9.44E+18	6.53E+18	4.51E+18	3.57E+18	4.51E+18	4.51E+18	5.68E+18
5	5.0	1.19E+19	8.20E+18	5.68E+18	4.48E+18	5.68E+18	5.68E+18	7.07E+18
6	6.2	1.42E+19	9.85E+18[c]	6.80E+18	5.35E+18	6.80E+18	6.80E+18	8.43E+18
7	7.5	1.65E+19	1.14E+19	7.88E+18	6.21E+18	7.88E+18	7.88E+18	9.71E+18
8	8.7	1.87E+19	1.30E+19	8.97E+18	7.06E+18	8.97E+18	8.97E+18	1.10E+19
9	9.9	2.09E+19	1.45E+19	1.01E+19	8.00E+18	1.01E+19	1.01E+19	1.22E+19
10	11.3	2.30E+19	1.60E+19	1.12E+19	8.91E+18	1.12E+19	1.12E+19	1.34E+19
11	12.5	2.52E+19	1.75E+19	1.23E+19	9.71E+18	1.23E+19	1.23E+19	1.46E+19
12	13.8	2.75E+19	1.91E+19	1.33E+19	1.06E+19	1.33E+19	1.33E+19	1.59E+19
13	15.1	2.99E+19	2.08E+19[d]	1.45E+19	1.15E+19	1.45E+19	1.45E+19	1.71E+19
14	16.5	3.22E+19	2.25E+19	1.57E+19	1.24E+19	1.68E+19	1.62E+19	1.84E+19
15	17.7	3.44E+19	2.41E+19	1.68E+19	1.32E+19	1.91E+19	1.78E+19	1.96E+19
16	19.0	3.65E+19	2.56E+19	1.78E+19	1.41E+19	2.35E+19	1.93E+19	2.07E+19
17	20.3	3.90E+19	2.75E+19	1.91E+19	1.50E+19	2.60E+19	2.12E+19	2.20E+19
18	21.6	4.15E+19	2.93E+19	2.03E+19	1.59E+19	2.85E+19	2.30E+19	2.33E+19
19	22.9	4.37E+19	3.09E+19	2.14E+19	1.68E+19	3.06E+19	2.46E+19	2.44E+19
20	24.3	4.59E+19	3.25E+19	2.26E+19	1.77E+19	3.29E+19	2.62E+19	2.56E+19
21	25.5	4.80E+19	3.41E+19	2.36E+19	1.86E+19	3.50E+19	2.78E+19	2.67E+19
22	26.8	5.03E+19	3.57E+19	2.48E+19	1.95E+19	3.72E+19	2.94E+19	2.80E+19
23	28.1	5.26E+19	3.73E+19	2.59E+19	2.03E+19	3.96E+19	3.10E+19	2.92E+19
Projected	52.3	9.75E+19	6.82E+19	4.63E+19	3.59E+19	8.44E+19	6.19E+19	5.36E+19
Projected	54.0	1.01E+20	7.04E+19	4.78E+19	3.70E+19	8.76E+19	6.41E+19	5.53E+19
Projected	72.0	1.34E+20	9.33E+19	6.30E+19	4.85E+19	1.21E+20	8.70E+19	7.34E+19

Notes:

(a) Information taken from WCAP-18015-NP (Reference 8).

(b) Capsule V was withdrawn at the end-of-cycle 1.

(c) Capsule U was withdrawn at the end-of-cycle 6.

(d) Capsule W was withdrawn at the end-of-cycle 13.

(e) Capsule Z was moved at the end-of-cycle 13.

(f) Capsule T was moved at the end-of-cycle 13.

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

Material	Region	Neutron Fluence [n/cm^2]			
		29.7 EFPY	50.3 EFPY	54 EFPY	72 EFPY[(a)]
Postulated 1/4T Flaw in Inlet Nozzle					
Nozzle 09 (Ht. # 990290-11)	Extended Beltline[(h)]	1.65E+16	2.92E+16	3.15E+16	4.25E+16
Nozzle 10[(b)] (Ht. # 990290-12)	Extended Beltline	6.13E+16	1.04E+17	1.11E+17	1.48E+17
Nozzle 11 (Ht. # 990268-11)	Extended Beltline[(h)]	2.29E+16	3.94E+16	4.24E+16	5.68E+16
Postulated 1/4T Flaw in Outlet Nozzle					
Nozzle 12 (Ht. # 990290-31)	Extended Beltline[(h)]	3.62E+16	6.12E+16	6.57E+16	8.75E+16
Nozzle 13 (Ht. # 990290-22)	Extended Beltline[(h)]	9.74E+15	1.72E+16	1.86E+16	2.51E+16
Nozzle 14 (Ht. # 990290-21)	Extended Beltline[(h)]	1.35E+16	2.33E+16	2.50E+16	3.35E+16
Centerline of the Inlet Nozzle Forging to Vessel Shell Welds – Lowest Extent[(i)]					
Nozzle 09	Extended Beltline[(h)]	3.50E+16	6.17E+16	6.65E+16	8.98E+16
Nozzle 10[(d)]	Extended Beltline	1.30E+17	2.19E+17	2.35E+17	3.13E+17
Nozzle 11[(e)]	Extended Beltline	4.85E+16	8.33E+16	8.95E+16	1.20E+17
Centerline of the Outlet Nozzle Forging to Vessel Shell Welds – Lowest Extent[(i)]					
Nozzle 12[(c)]	Extended Beltline	7.53E+16	1.27E+17	1.37E+17	1.82E+17
Nozzle 13	Extended Beltline[(h)]	2.03E+16	3.59E+16	3.87E+16	5.22E+16
Nozzle 14	Extended Beltline[(h)]	2.82E+16	4.84E+16	5.20E+16	6.97E+16
Upper Shell (Ht. # 990286 / 295213)	Beltline	1.30E+18	2.15E+18	2.30E+18	3.04E+18
Upper Shell to Intermediate Shell Circumferential Weld (Ht. # 25295 & 4278)	Beltline	1.50E+18	2.48E+18	2.66E+18	3.51E+18
Intermediate Shell (Ht. # 990311 / 298244)	Beltline	3.11E+19	5.03E+19	5.39E+19	7.07E+19
Intermediate Shell to Lower Shell Circumferential Weld (Ht. # 25531)	Beltline	3.09E+19	5.02E+19	5.36E+19	7.04E+19
Lower Shell (Ht. # 990400 / 292332)	Beltline	3.16E+19	5.13E+19	5.48E+19	7.20E+19
Lower Shell to Lower Vessel Head Circumferential Weld[(g)(i)]	Outside Beltline	< 1.00E+17	< 1.00E+17	< 1.00E+17	< 1.00E+17

Notes:

(a) Corresponds to 80 years of life.

(b) 1/4 T Flaw in Inlet Nozzle Inlet 10 is projected to reach 1.0E+17 n/cm^2 at approximately 48.5 EFPY; which corresponds to December 26, 2034[(f)].

(c) Outlet Nozzle 12 is projected to reach 1.0E+17 n/cm^2 at approximately 39.5 EFPY; which corresponds to June 6, 2025[(f)].

(d) Inlet Nozzle 10 reached 1.0E+17 n/cm^2 at approximately 22.4 EFPY, which occurred during Cycle 19.

(e) Inlet Nozzle 11 is projected to reach 1.0E+17 n/cm^2 at approximately 60.3 EFPY; which corresponds to May 1, 2047[(f)].

(f) Note, the dates provided in notes b, c, and e are approximations based on an 18 month cycle and average outage time of 25 days.

(g) The lower shell to lower vessel head circumferential weld is not modeled, it is known to be below the 1E+17 n/cm^2 fast neutron fluence threshold due to the fact that: it is 32 cm further from the core midplane than the above-core threshold location at 72 EFPY, and that the coolant below the core is cooler than the coolant above the core, which increases the density and shielding effects, reducing the fluence below the core relative to above the core.

(h) Component is conservatively included in the "Extended Beltline" even though its projected SLR fluence is less than 1E+17 n/cm^2 (E > 1.0 MeV) because, either a component at the same axial elevation meets the "Extended Beltline" fluence criterion, or the same component meets the fluence criterion at a lower elevation.

(i) The specific heat numbers of these welds could not be identified in the available information.

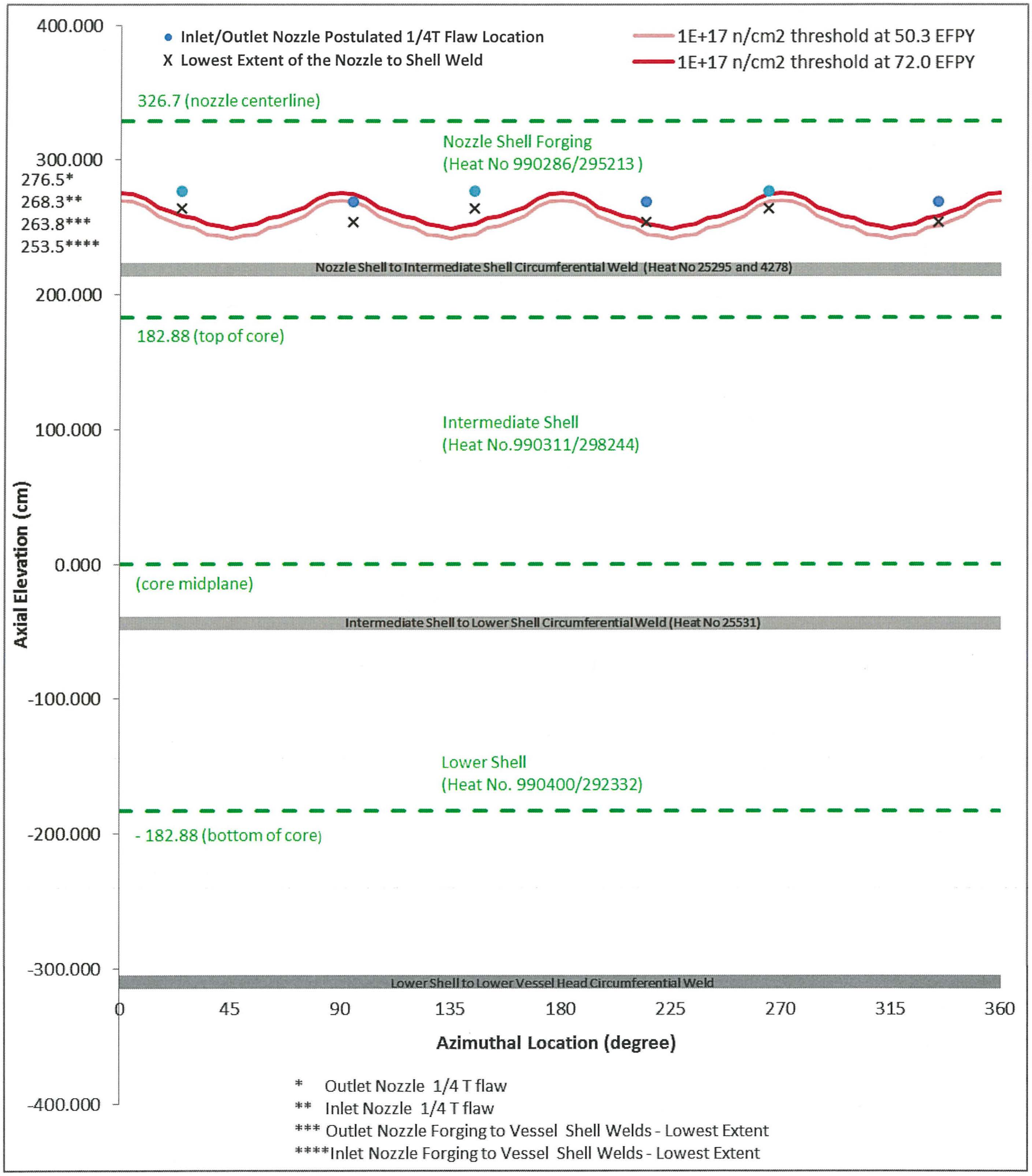

Figure 2-1 North Anna Unit 1 - Axial Boundary of the 1.0E+17 n/cm^2 Fast Neutron (E > 1.0 MeV) Fluence Threshold in the +Z Direction (at 50.3 EFPY and 72 EFPY)

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

Material	Region	Neutron Fluence [n/cm^2]			
		28.1 EFPY	52.3 EFPY	54 EFPY	72 EFPY[(a)]
Postulated 1/4T Flaw in Inlet Nozzle					
Nozzle 09 (Ht. # 990426)	Extended Beltline[(h)]	1.56E+16	2.85E+16	2.94E+16	3.90E+16
Nozzle 10[(b)] (Ht. # 54567-2)	Extended Beltline	5.69E+16	1.07E+17	1.11E+17	1.48E+17
Nozzle 11 (Ht. # 54590-2)	Extended Beltline[(h)]	2.17E+16	4.05E+16	4.19E+16	5.59E+16
Postulated 1/4T Flaw in Outlet Nozzle					
Nozzle 12 (Ht. # 990426-22)	Extended Beltline[(h)]	3.36E+16	6.33E+16	6.54E+16	8.75E+16
Nozzle 13 (Ht. # 990426-31)	Extended Beltline[(h)]	9.18E+15	1.68E+16	1.73E+16	2.30E+16
Nozzle 14 (Ht. # 791291)	Extended Beltline[(h)]	1.28E+16	2.39E+16	2.47E+16	3.30E+16
Centerline of the Inlet Nozzle Forging to Vessel Shell Welds – Lowest Extent (Ht. # 8816, 20459, & 27622)					
Nozzle 09	Extended Beltline[(h)]	3.29E+16	6.03E+16	6.22E+16	8.26E+16
Nozzle 10[(d)]	Extended Beltline	1.21E+17	2.27E+17	2.35E+17	3.14E+17
Nozzle 11[(e)]	Extended Beltline	4.60E+16	8.58E+16	8.86E+16	1.18E+17
Centerline of the Outlet Nozzle Forging to Vessel Shell Welds – Lowest Extent (Ht. # 8816, 20459, & 27622)					
Nozzle 12[(c)]	Extended Beltline	6.99E+16	1.32E+17	1.36E+17	1.82E+17
Nozzle 13	Extended Beltline[(h)]	1.91E+16	3.50E+16	3.61E+16	4.79E+16
Nozzle 14	Extended Beltline[(h)]	2.67E+16	4.98E+16	5.14E+16	6.87E+16
Upper Shell (Ht. # 990598 / 291396)	Beltline	1.20E+18	2.23E+18	2.30E+18	3.07E+18
Upper Shell to Intermediate Shell Circumferential Weld (Ht. # 4278 & 801)	Beltline	1.38E+18	2.58E+18	2.66E+18	3.55E+18
Intermediate Shell (Ht. # 990496 / 292424)	Beltline	2.87E+19	5.25E+19	5.42E+19	7.20E+19
Intermediate Shell to Lower Shell Circumferential Weld (Ht. #716126)	Beltline	2.86E+19	5.24E+19	5.41E+19	7.18E+19
Lower Shell (Ht. # 990533 / 297355)	Beltline	2.92E+19	5.36E+19	5.53E+19	7.34E+19
Lower Shell to Lower Vessel Head Circumferential Weld[(g)] (Ht. # 716126)	Outside Beltline	< 1.00E+17	< 1.00E+17	< 1.00E+17	< 1.00E+17

Notes:

(a) Corresponds to 80 years of life.

(b) Postulated 1/4T Flaw in Inlet Nozzle Inlet 10 is projected to reach 1.0E+17 n/cm^2 at approximately 48.8 EFPY; which corresponds to May 27, 2036[(f)].

(c) Outlet Nozzle 12 is projected to reach 1.0E+17 n/cm^2 at approximately 39.8 EFPY; which corresponds to February 4, 2027[(f)].

(d) Inlet Nozzle 10 reached 1.0E+17 n/cm^2 at approximately 23.1 EFPY, which occurred during Cycle 20.

(e) Inlet Nozzle 11 is projected to reach 1.0E+17 n/cm^2 at approximately 60.9 EFPY; which corresponds to February 12, 2049[(f)].

(f) Note, the dates provided in notes b, c, and e are approximations based on an 18 month cycle and average outage time of 25 days.

(g) The lower shell to lower vessel head circumferential weld is not modeled, it is known to be below the 1E+17 n/cm^2 fast neutron fluence threshold due to the fact that: it is 32 cm further from the core midplane than the above-core threshold location at 72 EFPY, and that the coolant below the core is cooler than the coolant above the core, which increases the density and shielding effects, reducing the fluence below the core relative to above the core.

(h) Component is conservatively included in the "Extended Beltline" even though its projected SLR fluence is less than 1E+17 n/cm^2 (E > 1.0 MeV) because, either a component at the same axial elevation meets the "Extended Beltline" fluence criterion, or the same component meets the fluence criterion at a lower elevation.

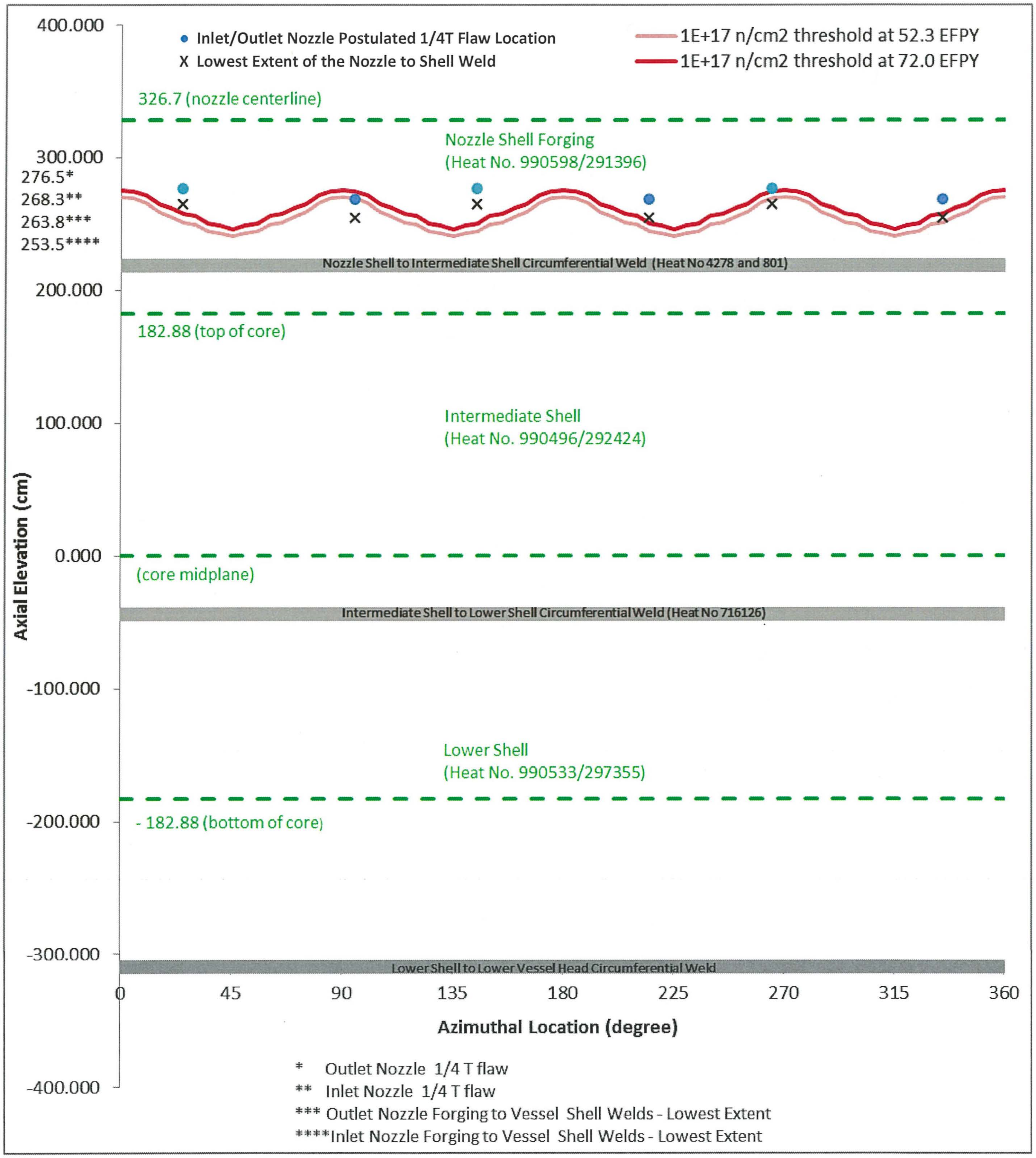

Figure 2-2 North Anna Unit 2 - Axial Boundary of the 1.0E+17 n/cm^2 Fast Neutron (E > 1.0 MeV) Fluence Threshold in the +Z Direction (at 52.3 EFPY and 72 EFPY)

3 MATERIAL PROPERTY INPUT

The requirements for P-T limit curve development are specified in 10 CFR 50, Appendix G (Reference 4). The beltline region of the reactor vessel is defined as the following in 10 CFR 50, Appendix G:

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

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

The North Anna Units 1 and 2 beltline materials consist of one (1) Intermediate Shell (IS) Forging, one (1) Lower Shell (LS) Forging, one (1) Upper Shell (US) Forging (also termed nozzle shell forging), and two (2) circumferential welds: the IS to LS Circumferential Weld and the US to IS Circumferential Weld. The reactor vessel (RV) forgings and weld materials are shown in Figure 3-1 for North Anna Units 1 and 2. Used in conjunction with the fluence data in Tables 2-3 and 2-4, and Figures 2-1 and 2-2, the beltline and extended beltline materials are identified as shown in Tables 3-1 and 3-2.

The North Anna Unit 1 surveillance forging material was made from reactor vessel Lower Shell Forging 03, Heat # 990400 / 292332. The North Anna Unit 1 reactor vessel beltline IS to LS Circumferential Weld was fabricated using weld wire Heat # 25531, Flux Type SMIT 89, Flux Lot Number 1211. The weld material in the North Anna Unit 1 surveillance program was fabricated with the same material heat, flux type, and flux lot number as reactor vessel beltline IS to LS Circumferential Weld. The outer diameter (OD) 94% of the US to IS Circumferential Weld was fabricated with weld wire Heat # 25295, Flux Type SMIT 89, Flux Lot Number 1170 and the inner diameter (ID) 6% was fabricated with weld wire Heat # 4278, Flux Type SMIT 89, Flux Lot Number 1211. Surveillance data does not exist for Heat # 25295 or Heat # 4278 in the North Anna Unit 1 reactor vessel surveillance program; however weld wire Heat # 25295 or Heat # 4278 were included in the surveillance programs of other plants as summarized in Table 3-5.

The North Anna Unit 2 surveillance forging material was made from reactor vessel Intermediate Shell Forging 04, Heat # 990496 / 292424. The North Anna Unit 2 reactor vessel beltline IS to LS Circumferential Weld was fabricated using weld wire Heat # 716126, LW320 Flux Type, Flux Lot Number 26. The weld material in the North Anna Unit 2 surveillance program was fabricated with the same material heat, flux type, and flux lot number as reactor vessel beltline IS to LS Circumferential Weld. The OD 94% of US to IS Circumferential Weld was fabricated with weld wire Heat # 4278, Flux Type SMIT 89, Flux Lot Number 1211. Surveillance data does not exist for Heat # 4278 in the North Anna Unit 2 reactor vessel surveillance program; however, as previously stated, it was included in the surveillance programs of other plants, as summarized in Table 3-5. The remaining 6% of the US to IS Circumferential Weld was fabricated from weld wire Heat # 801, SMIT 89 Flux Type, Flux Lot Number 1211. Surveillance data does not exist for Heat # 801.

Based on the results of Section 2 of this report, the materials that exceed the 1 x 10^{17} n/cm^2 (E > 1.0 MeV) threshold at 72 EFPY are considered to be the North Anna Units 1 and 2 extended beltline materials and are evaluated to determine their impact on the proposed SLR period of operation. The North Anna Units 1 and 2 reactor vessels contain three (3) Inlet Nozzles, three (3) Outlet Nozzles, three (3) Inlet Nozzle to US Welds, and three (3) Outlet Nozzle to US Welds per Unit. Only the forgings and welds corresponding to the North Anna Units 1 and 2 Inlet Nozzles 10, Inlet Nozzles 11, and Outlet Nozzles 12 are predicted to experience neutron fluence greater than 1.0 x 10^{17} n/cm^2 at SLR. Only those materials with a fluence greater than 1 x 10^{17} n/cm^2 (E > 1.0 MeV) at SLR require the effects of embrittlement to be included when evaluating the reactor vessel integrity.

For the Unit 1 Inlet/Outlet Nozzle to US Welds, the heat numbers, flux type, and flux lot numbers of these welds could not be identified in the available information; however, these welds were fabricated at the Rotterdam Dockyard Company (Rotterdam). Therefore, conservative generic/bounding properties from PWROG-17090-NP-A (Reference 12) are used. The Unit 2 Inlet/Outlet Nozzle to US Welds were fabricated using Heat # 8816, Flux Type LW320, Lot Number 28; Heat # 20459, Flux Type LW320, Lot Number 26; and Heat # 27622, Flux Type LW320, Lot Numbers 26 & 28. The records do not identify which weld heats are associated with which specific nozzles. Therefore, the bounding material properties (which consider all available data, as documented in PWROG-18005-NP [Reference 9]) will be conservatively associated with all North Anna Unit 2 nozzle welds. Justification for the use of PWROG-17090-NP-A, consistent with the NRC Safety Evaluation, is presented in Appendix K.

The unirradiated material property inputs used for the RV integrity evaluations herein are contained in PWROG-18005-NP (Reference 9). PWROG-18005-NP defined or redefined many of the material properties and chemistry values using the most up-to-date methodologies and all available data; therefore, the values utilized herein supersede previously documented values. The sources and methods used in the determination of the chemistry factors and the fracture toughness properties are summarized below.

Chemical Compositions

The best-estimate copper (Cu) and nickel (Ni) chemical compositions for the North Anna Units 1 and 2 beltline and extended beltline materials are presented in Tables 3-1 and 3-2. The best-estimate weight percent copper and nickel values for the beltline and extended beltline materials were previously reported in PWROG-18005-NP.

Fracture Toughness Properties

The initial fracture toughness properties (initial RT_{NDT} and initial USE) of most of the beltline and extended beltline forging materials were originally determined using NUREG-0800, BTP 5-3 Position 1.1 (Reference 10) methodology. The exceptions are the North Anna Units 1 and 2 IS Forging 04, and LS Forging 03 which were determined using the ASME Code, Section III (Reference 11) methods. Many of the beltline and extended beltline fracture toughness properties were updated per ASME Section III and NUREG-0800, BTP 5-3 Position 1.1 methodologies, as described in PWROG-18005-NP. The most up-to-date initial RT_{NDT} and initial USE values are documented in PWROG-18005-NP for North Anna Units 1 and 2. The beltline and extended beltline material properties of the North Anna Units 1 and 2 reactor vessels are presented in Tables 3-1 and 3-2 herein.

The initial RT_{NDT} values of the reactor vessel flange and closure head serve as input to the P-T limit curves "flange-notch" per 10 CFR 50, Appendix G (Reference 4). Since North Anna Units 1 and 2 share P-T Limit curves for operation, materials for both plants must be considered concerning input acceptability. The closure heads at both North Anna Units 1 and 2 have been replaced, and the initial RT_{NDT} values of the North Anna Units 1 and 2 flange materials were confirmed in PWROG-18005-NP (Reference 9). The North Anna Unit 1 replacement closure head flange has an initial RT_{NDT} value of -76°F, determined per ASME Code Section III, NB-2300. The North Anna Unit 1 reactor vessel flange has an initial RT_{NDT} of -22°F, calculated using the BTP 5-3 methodology. The North Anna Unit 2 replacement head flange has an initial RT_{NDT} value of -49°F, determined per ASME Code Section III, NB-2300. The North Anna Unit 2 reactor vessel flange has an initial RT_{NDT} of -22°F, calculated using the BTP 5-3 methodology. See Table 3-3 for a summary of the initial RT_{NDT} values for these two components at each plant.

It is also noted that direct fracture toughness Master Curve data is available for North Anna Unit 1 Lower Shell Forging 03, as described in Appendix H.

Chemistry Factor Values

The chemistry factor (CF) values were calculated using Positions 1.1 and 2.1 of Regulatory Guide 1.99, Revision 2 (Reference 1). Position 1.1 uses Tables 1 and 2 from the Regulatory Guide along with the best-estimate copper and nickel weight percent values (contained in Tables 3-1 and 3-2). Position 2.1 uses the surveillance capsule data from all capsules tested to date and surveillance data from other plants, as applicable. Credibility evaluations of the North Anna Units 1 and 2 surveillance data are provided in Appendix E of this report. The calculated capsule fluence values are provided in Tables 2-1 and 2-2 and are used to determine the Position 2.1 CFs as shown in Tables 3-4 and 3-6 for North Anna Units 1 and 2, respectively. In addition, North Anna Units 1 and 2 utilize weld materials which are included in the Sequoyah Units 1 and 2 surveillance programs. Table 3-5 calculates the Position 2.1 CFs from the Sequoyah Units 1 and 2 surveillance weld materials for use in North Anna Units 1 and 2 calculations. The credibility evaluations of the Sequoyah Units 1 and 2 surveillance data are contained in WCAP-17539-NP (Reference 15). Tables 3-7 and 3-8 summarize the Positions 1.1 and 2.1 CF values determined for the North Anna Units 1 and 2 RPV beltline and extended beltline materials, respectively. Appendix I contains a description of the North Anna licensing basis relative to selection of CFs when surveillance data is available.

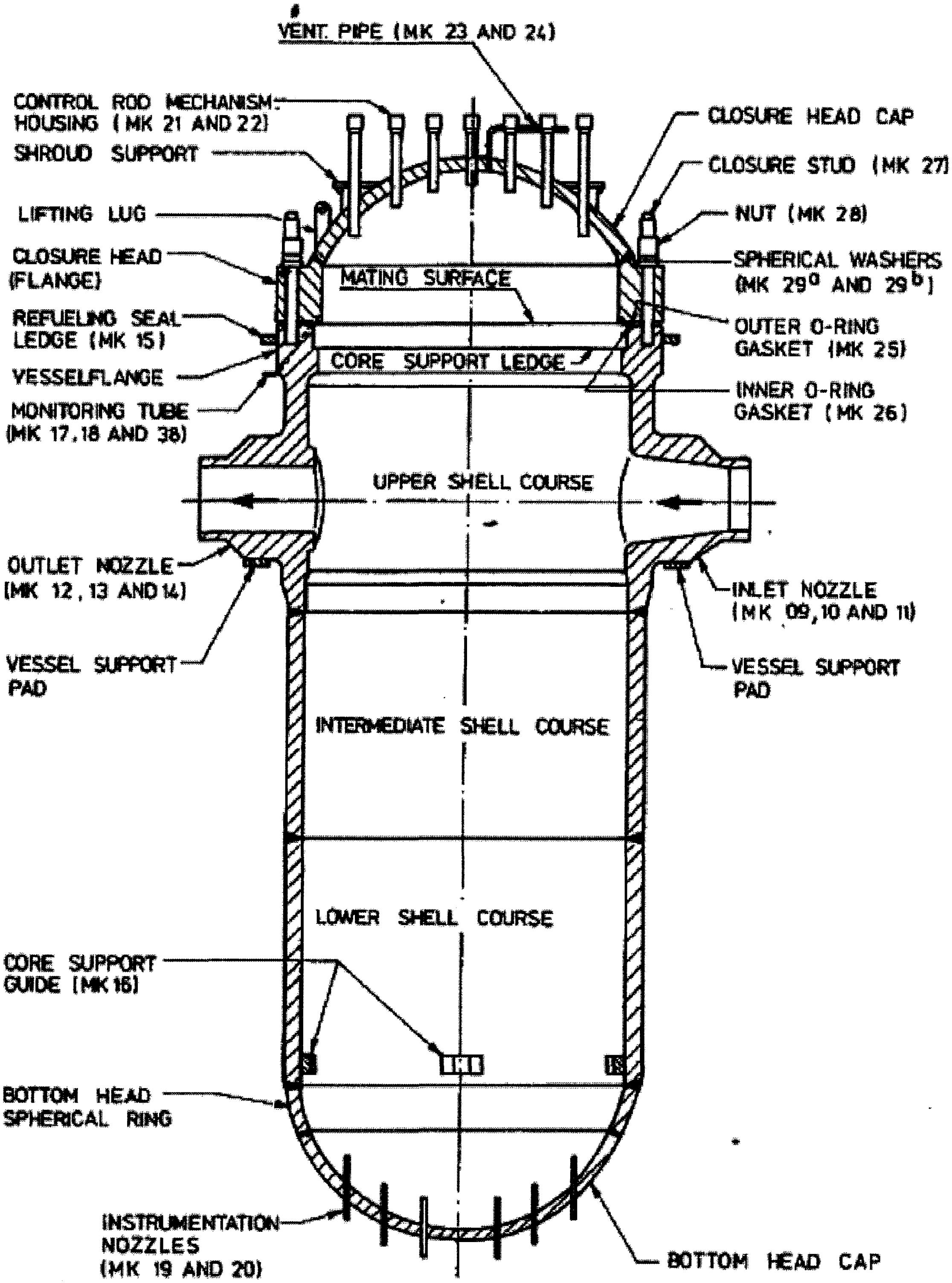

Figure 3-1 RPV Base Metal Material Identifications for North Anna Units 1 and 2
*Note: This figure is representative of the RPV with the original RPV closure heads.

Table 3-1 Best-Estimate Cu and Ni Weight Percent Values, Initial RT_{NDT} Values, and Initial USE Values for the North Anna Unit 1 RPV Beltline and Extended Beltline Materials[(a)]

Material Description	Heat Number	Flux Type (Lot)	Wt. % Cu	Wt. % Ni	$RT_{NDT(U)}$[(b)] (°F)	Initial USE (ft-lbs)
Reactor Vessel Beltline Materials						
Upper Shell Forging 05	990286 / 295213	-	0.16	0.74	1	72
Upper to Intermediate Shell Circumferential Weld (94% OD)	25295	SMIT 89 (1170)	0.352	0.125	-40	112
Upper to Intermediate Shell Circumferential Weld (6% ID)	4278	SMIT 89 (1211)	0.12	0.11	-4	105
Intermediate Shell Forging 04	990311 / 298244	-	0.12	0.82	-6	91
Intermediate to Lower Shell Circumferential Weld	25531	SMIT 89 (1211)	0.098	0.124	-2	95
Lower Shell Forging 03	990400 / 292332	-	0.156	0.817	33	85
Reactor Vessel Extended Beltline Materials						
Inlet/Outlet Nozzle to Upper Shell Welds	Rotterdam[(d)]		0.35[(c)]	1.13[(c)]	30[(d)]	72[(c)]
Inlet Nozzle Forging 09	990290-11	-	0.13	0.80	-14	≥ 71
Inlet Nozzle Forging 10	990290-12	-	0.13	0.79	-10	≥ 58
Inlet Nozzle Forging 11	990268-21	-	0.18	0.78	8	56[(c)]
Outlet Nozzle Forging 12	990290-31	-	0.13	0.80	-6	≥ 66
Outlet Nozzle Forging 13	990290-22	-	0.13	0.81	-7	≥ 59
Outlet Nozzle Forging 14	990290-21	-	0.13	0.81	8	≥ 59
Reactor Vessel Surveillance Program Materials[(e)]						
Lower Shell Forging 03	990400 / 292332	-	0.158	0.823	-	-
Intermediate to Lower Shell Circumferential Weld	25531	SMIT 89 (1211)	0.098	0.124	-	-

Notes:
(a) Unless otherwise noted, the information is extracted from PWROG-18005-NP (Reference 9). Dashes indicate when a category is not applicable to the material.
(b) All $RT_{NDT(U)}$ values are based on measured data which are used in conjunction with ASME Code Section III (Reference 11) and/or BTP 5-3 (Reference 10) methods; thus, a σ_I value of 0°F can be used with these $RT_{NDT(U)}$ values per WCAP-14040-A, Revision 4 (Reference 2).
(c) Generic value developed in PWROG-17090-NP-A (Reference 12). Justification for the use of these values, consistent with the NRC Safety Evaluation, are presented in Appendix K.
(d) The specific heat, flux type, and flux lot numbers of these welds could not be identified in the available information; therefore, conservative generic numbers will be used to describe these welds. The $RT_{NDT(U)}$ value was determined using ASME Code Section III minimum criteria at the time of fabrication and BTP 5-3 (Reference 10), Position 1.1(4) guidance. Since this is a maximum possible value based on measured data that satisfied the ASME requirements, the σ_I associated with this $RT_{NDT(U)}$ is zero.
(e) The reactor vessel surveillance material data is taken from Dominion Energy calculation SM-1008 (Reference 16).

Table 3-2 Best-Estimate Cu and Ni Weight Percent Values, Initial RT_{NDT} Values, and Initial USE Values for the North Anna Unit 2 RPV Beltline and Extended Beltline Materials[a]

Material Description	Heat Number	Flux Type (Lot)	Wt. % Cu	Wt. % Ni	$RT_{NDT(U)}$[b] (°F)	Initial USE (ft-lbs)
Reactor Vessel Beltline Materials						
Upper Shell Forging 05	990598 / 291396	-	0.08	0.77	8	72
Upper to Intermediate Shell Circumferential Weld (94% OD)	4278	SMIT 89 (1211)	0.12	0.11	-4	105
Upper to Intermediate Shell Circumferential Weld (6% ID)	801	SMIT 89 (1211)	0.18	0.11	10	75[c]
Intermediate Shell Forging 04	990496 / 292424	-	0.107	0.857	69	72
Intermediate to Lower Shell Circumferential Weld	716126	LW320 (26)	0.066	0.046	-67	109
Lower Shell Forging 03	990533 / 297355	-	0.13	0.83	37	80
Reactor Vessel Extended Beltline Materials						
Inlet/Outlet Nozzle to Upper Shell Welds	8816	LW320 (28)	0.23[c]	0.56[c]	30[d]	75[c]
	20459	LW320 (26)				
	27622	LW320 (26)				
	27622	LW320 (28)				
Inlet Nozzle Forging 09	990426	-	0.19	0.82	11	56[c]
Inlet Nozzle Forging 10	54567-2	-	0.14	0.79	5	≥ 77
Inlet Nozzle Forging 11	54590-2	-	0.155	0.77	-31	≥ 75
Outlet Nozzle Forging 12	990426-22	-	0.19	0.80	8	≥ 60
Outlet Nozzle Forging 13	990426-31	-	0.19	0.79	1	56[c]
Outlet Nozzle Forging 14	791291	-	0.12	0.82	-22	≥ 74
Reactor Vessel Surveillance Program Materials[e]						
Intermediate Shell Forging 04	990496 / 292424	-	0.116	0.886	-	-
Intermediate to Lower Shell Circumferential Weld	716126	LW320 (26)	0.067	0.052	-	-

Notes contained on the following page.

Notes:

(a) Unless otherwise noted, the information is extracted from PWROG-18005-NP (Reference 9). Dashes indicate when a category is not applicable to the material.

(b) All $RT_{NDT(U)}$ values are based on measured data which are used in conjunction with ASME Code Section III (Reference 11) and/or BTP 5-3 (Reference 10) methods; thus, a σ_I value of 0°F can be used with these $RT_{NDT(U)}$ values per WCAP-14040-A, Revision 4 (Reference 2).

(c) Generic value developed in PWROG-17090-NP-A (Reference 12). Justification for the use of these values, consistent with the NRC Safety Evaluation, are presented in Appendix K.

(d) The records do not identify which weld heats are associated with which specific nozzle welds. Therefore, the bounding material properties will be conservatively associated with all Unit 2 nozzle welds. The $RT_{NDT(U)}$ value was determined using ASME Code Section III minimum criteria at the time of fabrication and BTP 5-3 (Reference 10), Position 1.1(4) guidance. Since this is a maximum possible value based on measured data that satisfied the ASME requirements, the σ_I associated with this $RT_{NDT(U)}$ is zero.

(e) The reactor vessel surveillance material data is taken from Dominion Energy calculation SM-1008 (Reference 16).

Table 3-3 Initial RT_{NDT} Values for the North Anna Units 1 and 2 Replacement Reactor Vessel Closure Head and Vessel Flange Materials[(a)]

Reactor Vessel Material	Unit 1 Initial RT_{NDT} (°F)	Unit 2 Initial RT_{NDT} (°F)
Replacement Closure Head	-76	-49
Vessel Flange	-22	-22

Note:

(a) The information is extracted from PWROG-18005-NP (Reference 9).

Table 3-4 Calculation of Position 2.1 CF Values for North Anna Unit 1 Surveillance Materials

Material	Capsule	Capsule Fluence[(a)] (x 10^{19} n/cm^2, E > 1.0 MeV)	FF[(b)]	ΔRT_{NDT}[(c)] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Lower Shell Forging 03 (Tangential)	V	0.306	0.675	51	34.44	0.456
	U	0.914	0.975	116	113.08	0.950
	W	2.05	1.196	93	111.19	1.429
Lower Shell Forging 03 (Axial)	V	0.306	0.675	29	19.59	0.456
	U	0.914	0.975	72	70.19	0.950
	W	2.05	1.196	96	114.77	1.429
				SUM:	463.25	5.671
	$CF_{\text{Surveillance Forging}} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma (FF)^2 = (463.25) \div (5.671) = \mathbf{81.68°F}$					
Surveillance Weld Metal (Heat # 25531)	V	0.306	0.675	88	59.43	0.456
	U	0.914	0.975	30	29.24	0.950
	W	2.05	1.196	86	102.82	1.429
				SUM:	191.50	2.836
	$CF_{\text{Surveillance Weld}} = \Sigma (FF * \Delta RT_{NDT}) \div \Sigma (FF)^2 = (191.50) \div (2.836) = \mathbf{67.53°F}$					

Notes:

(a) The fluence values are taken from Table 2-1 of this report.

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

(c) ΔRT_{NDT} values are extracted from BAW-2356 (Reference 13). Chemistry adjustments are not performed because the beltline and surveillance materials are identical and/or not adjusting for chemical composition is conservative.

Table 3-5 Calculation of Position 2.1 CF Values for North Anna Units 1 and 2 Welds with Data from Other Plant Surveillance Programs

Material	Capsule	Capsule Fluence[(a)] (x 10^{19} n/cm², E > 1.0 MeV)	FF[(b)]	ΔRT_{NDT}[(a)(c)] (°F)	FF*ΔRT_{NDT} (°F)	FF²
Sequoyah 1 Surveillance Weld Material (Heat # 25295)	T	0.241	0.615	123.79 (127.79)	76.11	0.378
	U	0.693	0.897	140.92 (144.92)	126.43	0.805
	X	1.16	1.041	155.02 (159.02)	161.44	1.085
	Y	1.97	1.185	159.80 (163.80)	189.39	1.405
				SUM:	553.37	3.672
	$CF_{Surveillance\ Weld} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma (FF)^2 = (553.37) \div (3.672) =$ **150.69°F**					
Sequoyah 2 Surveillance Weld Material (Heat # 4278)[(d)]	T	0.244	0.618	70.56 (74.56)	43.60	0.382
	U	0.654	0.881	126.38 (130.38)	111.34	0.776
	X	1.16	1.041	40.22 (44.22)	41.89	1.085
	Y	2.02	1.192	82.91 (86.91)	98.81	1.420
				SUM:	295.63	3.663
	$CF_{Surveillance\ Weld} = \Sigma (FF * \Delta RT_{NDT}) \div \Sigma (FF)^2 = (295.63) \div (3.663) =$ **80.71°F**[(d)]					

Notes:

(a) Sequoyah Units 1 and 2 surveillance data are taken from WCAP-17539-NP (Reference 15).

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

(c) The surveillance weld ΔRT_{NDT} values have been decreased by 4°F (547°F - 551°F) to account for the difference in the operating temperature between the Sequoyah and North Anna units. Pre-adjusted values are listed in parentheses. Chemistry adjustments are not performed since the Sequoyah Units 1 and 2 surveillance weld CF values are 178.7°F and 67.9°F, respectively. This results in Position 1.1 CF ratios less than 1; therefore, not adjusting for chemical composition is conservative.

(d) Since North Anna Units 1 and 2 have same vessel weld CF and inlet temperature, the results apply to both units.

Table 3-6 Calculation of Position 2.1 CF Values for North Anna Unit 2 Surveillance Materials

Material	Capsule	Capsule Fluence(a) (x 10^{19} n/cm^2, E > 1.0 MeV)	FF(b)	ΔRT_{NDT}(c) (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Intermediate Shell Forging 04 (Tangential)	V	0.286	0.658	19	12.50	0.433
	U	0.985	0.996	33	32.86	0.992
	W	2.08	1.199	86	103.14	1.438
Intermediate Shell Forging 04 (Axial)	V	0.286	0.658	21	13.82	0.433
	U	0.985	0.996	66	65.72	0.992
	W	2.08	1.199	65	77.96	1.438
				SUM:	306.00	5.726
	$CF_{\text{Surveillance Forging}} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma (FF)^2 = (306.00) \div (5.726) = \mathbf{53.44°F}$					
Surveillance Weld Metal (Heat # 716126)	V	0.286	0.658	18	11.84	0.433
	U	0.985	0.996	8	7.97	0.992
	W	2.08	1.199	47	56.37	1.438
				SUM:	76.18	2.863
	$CF_{\text{Surveillance Weld}} = \Sigma (FF * \Delta RT_{NDT}) \div \Sigma (FF)^2 = (76.18) \div (2.863) = \mathbf{26.61°F}$					

Notes:

(a) The fluence values are taken from Table 2-2 of this report.

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

(c) ΔRT_{NDT} values are extracted from BAW-2376 (Reference 14). Chemistry adjustments are not performed because the beltline and surveillance materials are identical and/or not adjusting for chemical composition is conservative.

Table 3-7 Summary of the North Anna Unit 1 RPV Beltline, Extended Beltline, and Surveillance Material CF Values based on Regulatory Guide 1.99, Revision 2, Position 1.1 and Position 2.1

Material	Heat Number	Flux Type (Lot)	Chemistry Factor Position 1.1[a] (°F)	Chemistry Factor Position 2.1[b] (°F)
Reactor Vessel Beltline Materials				
Upper Shell Forging 05	990286 / 295213	-	121.50	-
Upper to Intermediate Shell Circumferential Weld (OD 94%)	25295	SMIT 89 (1170)	163.25	150.69
Upper to Intermediate Shell Circumferential Weld (ID 6%)	4278	SMIT 89 (1211)	63.00	80.71
Intermediate Shell Forging 04	990311 / 298244	-	86.00	-
Intermediate to Lower Shell Circumferential Weld	25531	SMIT 89 (1211)	56.22	67.53
Lower Shell Forging 03	990400 / 292332	-	119.97	81.68
Reactor Vessel Extended Beltline Materials				
Inlet/Outlet Nozzle to Upper Shell Welds	Rotterdam	-	293.45	-
Inlet Nozzle Forging 09	990290-11	-	96.00	-
Inlet Nozzle Forging 10	990290-12	-	95.75	-
Inlet Nozzle Forging 11	990268-21	-	140.30	-
Outlet Nozzle Forging 12	990290-31	-	96.00	-
Outlet Nozzle Forging 13	990290-22	-	96.00	-
Outlet Nozzle Forging 14	990290-21	-	96.00	-
Reactor Vessel Surveillance Program Materials				
Lower Shell Forging 03	990400 / 292332	-	121.63	-
Intermediate to Lower Shell Circumferential Weld	25531	SMIT 89 (1211)	56.22	-

Notes:

(a) All values are based on Tables 1 and 2 of Regulatory Guide 1.99, Revision 2 (Position 1.1) using the Cu and Ni weight percent values given in Table 3-1 of this report. Dashes indicate when a category is not applicable to the material.

(b) Values are from Tables 3-4 and 3-5 of this report.

Table 3-8 Summary of the North Anna Unit 2 RPV Beltline, Extended Beltline, and Surveillance Material CF Values based on Regulatory Guide 1.99, Revision 2, Position 1.1 and Position 2.1

Material	Heat Number	Flux Type (Lot)	Chemistry Factor	
			Position 1.1[(a)] (°F)	Position 2.1[(b)] (°F)
Reactor Vessel Beltline Materials				
Upper Shell Forging 05	990598 / 291396	-	51.00	-
Upper to Intermediate Shell Circumferential Weld (OD 94%)	4278	SMIT 89 (1211)	63.00	80.71
Upper to Intermediate Shell Circumferential Weld (ID 6%)	801	SMIT 89 (1211)	87.80	-
Intermediate Shell Forging 04	990496 / 292424	-	74.00	53.44
Intermediate to Lower Shell Circumferential Weld	716126	LW320 (26)	36.09	26.61
Lower Shell Forging 03	990533 / 297355	-	96.00	-
Reactor Vessel Extended Beltline Materials				
Inlet/Outlet Nozzle to Upper Shell Welds	8816 20459 27622	LW320 (26 & 28)	163.20	-
Inlet Nozzle Forging 09	990426	-	150.40	-
Inlet Nozzle Forging 10	54567-2	-	104.75	-
Inlet Nozzle Forging 11	54590-2	-	118.25	-
Outlet Nozzle Forging 12	990426-22	-	150.00	-
Outlet Nozzle Forging 13	990426-31	-	149.60	-
Outlet Nozzle Forging 14	791291	-	86.00	-
Reactor Vessel Surveillance Program Materials				
Intermediate Shell Forging 04	990496 / 292424	-	82.40	-
Intermediate to Lower Shell Circumferential Weld	716126	LW320 (26)	37.08	-

Notes:

(a) All values are based on Tables 1 and 2 of Regulatory Guide 1.99, Revision 2 (Position 1.1) using the Cu and Ni weight percent values given in Table 3-2 of this report. Dashes indicate when a category is not applicable to the material.

(b) Values are from Tables 3-5 and 3-6 of this report.

4 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

4.1 OVERALL APPROACH

The ASME (American Society of Mechanical Engineers) approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{Ic}, for the metal temperature at that time. K_{Ic} is obtained from the reference fracture toughness curve, defined in the 1998 Edition through 2000 Addenda of Section XI, Appendix G of the ASME Code (Reference 3). The K_{Ic} curve is given by the following equation:

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

where,

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

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

4.2 METHODOLOGY FOR PRESSURE-TEMPERATURE LIMIT CURVE DEVELOPMENT

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

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

where,

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

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

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

C = 2.0 for Level A and Level B service limits

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

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

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

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

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

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

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

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

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

where,

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

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

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

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

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

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

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

where HU is the heatup rate in °F/hr.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Following the generation of pressure-temperature curves for the steady-state and finite heatup rate situations, the final limit curves are produced by constructing a composite curve based on a point-by-point

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

4.3 LOWEST SERVICE TEMPERATURE REQUIREMENTS

North Anna Units 1 and 2 are Westinghouse-designed plants; thus, the primary Reactor Coolant System (RCS) piping is stainless steel. Therefore, the lowest service temperature requirements of Paragraph NB-2332 of ASME Code Section III (Reference 11) do not apply to the North Anna Units 1 and 2 reactor vessels. See Appendix C for additional details.

4.4 CLOSURE HEAD/VESSEL FLANGE REQUIREMENTS

10 CFR Part 50, Appendix G (Reference 4) addresses the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure head regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the preservice hydrostatic test pressure, which is calculated to be 621 psig. The initial RT_{NDT} values of the reactor vessel closure head and vessel flange are documented in Table 3-3. The limiting unirradiated RT_{NDT} of -22°F is associated with the North Anna Units 1 and 2 reactor vessel flange materials, so the minimum allowable temperature of this region is 98°F at pressures greater than 621 psig (without margins for instrument uncertainties). This limit is shown in Tables 6-1 and 6-3, as well as Figures 6-1 and 6-2.

4.5 BOLTUP TEMPERATURE REQUIREMENTS

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

5 CALCULATION OF ADJUSTED REFERENCE TEMPERATURE

The current P-T limit curves implemented in the North Anna Units 1 and 2 Technical Specifications (Reference 18) 3.4.3 control plant operation through EOLE, 50.3 EFPY and 52.3 EFPY for Units 1 and 2, respectively. SLR will extend the operation of North Anna Units 1 and 2 to 72 EFPY. In order develop P-T limit curves for SLR, new adjusted reference temperature (ART) values are calculated herein using the NRC methodology in Regulatory Guide 1.99, Revision 2 (Reference 1).

From Regulatory Guide 1.99, Revision 2 (Reference 1), the ART for each material in the beltline region is given by the following expression:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \tag{10}$$

Initial RT_{NDT} is the reference temperature for the unirradiated material as defined in Paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code (Reference 11). If measured values of the initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may be used, provided there are sufficient test results to establish a mean and standard deviation for the class.

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

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

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

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

where x inches (reactor vessel cylindrical shell beltline thickness is 7.677 inches) is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then placed in Equation 11 to calculate the ΔRT_{NDT} at the specific depth. It is noted that the previous P-T limits analysis in WCAP-15112 (Reference 17) utilized a thickness of 7.705 inches. This difference in thickness is negligible, and therefore the design dimension of 7.677 inches is utilized herein.

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

Tables 5-2 and 5-3 contain the surface fluence values at 72 EFPY, which were used for the development of the P-T limit curves contained in this report. Tables 5-2 and 5-3 also contain the 1/4T and 3/4T calculated fluence values and fluence factors (FFs), per Regulatory Guide 1.99, Revision 2, as applicable. The values in this table will be used to calculate the 72 EFPY ART values for the North Anna Units 1 and 2 reactor vessel materials.

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

The 1/4T and 3/4T ART calculations for North Anna Unit 1 are shown in Tables 5-4 and 5-5, respectively. The 1/4T and 3/4T ART calculations for North Anna Unit 2 are shown in Tables 5-7 and 5-8, respectively.

The ART values for the extended beltline are conservatively calculated using surface fluence values. Therefore, the 1/4T and 3/4T ART calculations for the North Anna Units 1 and 2 exclude the inlet and outlet nozzle forging and weld materials. Instead the ART values for the nozzle forging and weld materials are contained in Tables 5-6 and 5-9. North Anna Units 1 and 2 Inlet Nozzle 10 have projected fluence values that exceed the 1 x 10^{17} n/cm^2 (E > 1.0 MeV) fluence threshold at the postulated 1/4T flaw location at 72 EFPY per Tables 2-3 and 2-4. Therefore, neutron radiation embrittlement should be considered herein for these nozzle forging materials. For all other forging and weld materials with a fluence value less than 1 x 10^{17} n/cm^2 (E > 1.0 MeV) the embrittlement effects can be neglected and the FF is reduced to 0.

The limiting ART values for North Anna Units 1 and 2 are summarized in Table 5-1 as well as those ART values used in the current North Anna Units 1 and 2 Technical Specifications P-T limit curves. The 1/4T and 3/4T limiting ART values at 72 EFPY are less than the 1/4T and 3/4T ART values used in the current Technical Specifications. This decrease is driven by the reduction in the initial RT_{NDT} of the limiting material, i.e. the Unit 2 Lower Shell Forging 03 initial RT_{NDT} reduced from 56°F to 37°F. A comparison of the material property input values is provided in Appendix G.

Table 5-1
Summary of the Limiting ART Values Used in Generation of the North Anna Units 1 and 2 Reactor Vessel Heatup and Cooldown Curves at 72 EFPY

	1/4T Location	3/4T Location
Limiting ART[(a)] **(°F)**	205[(b)]	184[(b)]
	Limiting Material: Unit 2 Lower Shell Forging 03 (developed using Position 1.1 data)	
ART in Current Technical Specifications (°F)	218.5	195.6
	Limiting Material: Unit 2 Lower Shell Forging 03 (developed using Position 1.1 data)	

Notes:

(a) The ART values to be used for P-T limit curve development are the limiting 72 EFPY ART values from Tables 5-4 through 5-9. The values have been rounded up for conservatism.

(b) Note that the ART values calculated for Unit 1 Lower Shell Forging 03 and Unit 2 Intermediate Shell Forging 04 per Regulatory Guide 1.99, Revision 2, Position 1.1, are higher. However, the use of the lesser of the Regulatory Guide 1.99, Revision 2, Position 1.1 and 2.1 CFs with non-credible data and a full margin term is justified since none of the surveillance data are more than two times sigma-delta above the Position 1.1 CF trend line. This determination is documented in Appendix I.

Table 5-2 Fluence Values and Fluence Factors for the Vessel Surface, 1/4T, and 3/4T Locations for the North Anna Unit 1 Reactor Vessel Materials at 72 EFPY

Material	Surface Fluence[(a)] (x 10^{19} n/cm², E> 1.0 MeV)	Surface FF[(b)]	1/4T Fluence[(a)] (x 10^{19} n/cm², E> 1.0 MeV)	1/4T FF[(b)]	3/4T Fluence[(a)] (x 10^{19} n/cm², E> 1.0 MeV)	3/4T FF[(b)]
Reactor Vessel Beltline Materials						
Upper Shell Forging 05	0.304	0.674	0.192	0.559	0.0763	0.365
Upper to Intermediate Shell Circumferential Weld	0.351	0.711	0.221	0.594	0.0881	0.392
Intermediate Shell Forging 04	7.07	1.464	4.46	1.379	1.78	1.158
Intermediate to Lower Shell Circumferential Weld	7.04	1.464	4.44	1.378	1.77	1.157
Lower Shell Forging 03	7.20	1.467	4.54	1.383	1.81	1.163
Reactor Vessel Extended Beltline Materials						
Inlet Nozzle Forging 09 to Upper Shell Weld	0.00898	0[(d)]	See Note (c)			
Inlet Nozzle Forging 10 to Upper Shell Weld	0.0313	0.225				
Inlet Nozzle Forging 11 to Upper Shell Weld	0.0120	0.124				
Outlet Nozzle Forging 12 to Upper Shell Weld	0.0182	0.162				
Outlet Nozzle Forging 13 to Upper Shell Weld	0.00522	0[(d)]				
Outlet Nozzle Forging 14 to Upper Shell Weld	0.00697	0[(d)]				
Inlet Nozzle Forging 09	0.00425	0[(d)]				
Inlet Nozzle Forging 10	0.0148	0.142				
Inlet Nozzle Forging 11	0.00568	0[(d)]				
Outlet Nozzle Forging 12	0.00875	0[(d)]				
Outlet Nozzle Forging 13	0.00251	0[(d)]				
Outlet Nozzle Forging 14	0.00335	0[(d)]				

Notes:

(a) 72 EFPY surface fluence values for the reactor vessel materials were taken from Table 2-3 of this report. 1/4T and 3/4T fluence values were calculated from the surface fluence, the reactor vessel beltline thickness (7.677 inches), and equation $f = f_{surf} * e^{-0.24\,(x)}$ from Regulatory Guide 1.99, Revision 2, where x = the depth into the vessel wall (inches).

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

(c) The fluence values for the nozzle forgings are taken at the postulated 1/4T flaw axial location. The fluence values for the inlet/outlet nozzle to upper shell welds are taken at the lowest extent of the nozzle weld centerline. Analysis of the nozzle forgings and associated welds are conservatively performed using the surface fluence, neglecting attenuation through the reactor vessel wall.

(d) Because the fluence is less than 10^{17} n/cm² (E > 1.0 MeV), the FF is reduced to 0. Embrittlement effects only need to be considered if the fluence is greater than 10^{17} n/cm² (E > 1.0 MeV).

Table 5-3 Fluence Values and Fluence Factors for the Vessel Surface, 1/4T, and 3/4T Locations for the North Anna Unit 2 Reactor Vessel Materials at 72 EFPY

Material	Surface Fluence[(a)] (x 10^{19} n/cm^2, E> 1.0 MeV)	Surface FF[(b)]	1/4T Fluence[(a)] (x 10^{19} n/cm^2, E> 1.0 MeV)	1/4T FF[(b)]	3/4T Fluence[(a)] (x 10^{19} n/cm^2, E> 1.0 MeV)	3/4T FF[(b)]
Reactor Vessel Beltline Materials						
Upper Shell Forging 05	0.307	0.676	0.194	0.562	0.0771	0.367
Upper to Intermediate Shell Circumferential Weld	0.355	0.714	0.224	0.597	0.0891	0.394
Intermediate Shell Forging 04	7.20	1.467	4.54	1.383	1.81	1.163
Intermediate to Lower Shell Circumferential Weld	7.18	1.467	4.53	1.382	1.80	1.162
Lower Shell Forging 03	7.34	1.470	4.63	1.387	1.84	1.168
Reactor Vessel Extended Beltline Materials						
Inlet Nozzle Forging 09 to Upper Shell Weld	0.00826	0[(d)]	See Note (c)			
Inlet Nozzle Forging 10 to Upper Shell Weld	0.0314	0.226				
Inlet Nozzle Forging 11 to Upper Shell Weld	0.0118	0.123				
Outlet Nozzle Forging 12 to Upper Shell Weld	0.0182	0.162				
Outlet Nozzle Forging 13 to Upper Shell Weld	0.00479	0[(d)]				
Outlet Nozzle Forging 14 to Upper Shell Weld	0.00687	0[(d)]				
Inlet Nozzle Forging 09	0.00390	0[(d)]				
Inlet Nozzle Forging 10	0.0148	0.142				
Inlet Nozzle Forging 11	0.00559	0[(d)]				
Outlet Nozzle Forging 12	0.00875	0[(d)]				
Outlet Nozzle Forging 13	0.00230	0[(d)]				
Outlet Nozzle Forging 14	0.00330	0[(d)]				

Notes:

(a) 72 EFPY surface fluence values for the reactor vessel materials were taken from Table 2-4 of this report. 1/4T and 3/4T fluence values were calculated from the surface fluence, the reactor vessel beltline thickness (7.677 inches), and equation $f = f_{surf} * e^{-0.24\,(x)}$ from Regulatory Guide 1.99, Revision 2, where x = the depth into the vessel wall (inches).

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

(c) The fluence values for the nozzle forgings are taken at the postulated 1/4T flaw axial location. The fluence values for the inlet/outlet nozzle to upper shell welds are taken at the lowest extent of the nozzle weld centerline. Analysis of the nozzle forgings and associated welds are conservatively performed using the surface fluence, neglecting attenuation through the reactor vessel wall.

(d) Because the fluence is less than 10^{17} n/cm^2 (E > 1.0 MeV), the FF is reduced to 0. Embrittlement effects only need to be considered if the fluence is greater than 10^{17} n/cm^2 (E > 1.0 MeV).

Table 5-4 Calculation of the North Anna Unit 1 ART Values at the 1/4T Location for the Reactor Vessel Beltline Materials at 72 EFPY[(a)]

Material	Heat Number	Flux Type (Lot)	R.G. 1.99, Rev. 2 Position	CF[(b)]	1/4T Fluence[(c)] (x 10^{19} n/cm^2, E > 1.0 MeV)	1/4T FF[(c)]	$RT_{NDT(U)}$[(d)] (°F)	Predicted ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ[(e)] (°F)	M (°F)	ART (°F)
Upper Shell Forging 05	990286 / 295213	-	1.1	121.50	0.192	0.559	1	68.0	0.0	17.0	34.0	103.0
Upper to Intermediate Shell Circumferential Weld (OD 94%)	25295	SMIT 89 (1170)	1.1	163.25	0.221	0.594	-40	97.0	0.0	28.0	56.0	113.0
Using credible surveillance data[(f)]			2.1	150.69	0.221	0.594	-40	89.5	0.0	14.0	28.0	77.5
Upper to Intermediate Shell Circumferential Weld (ID 6%)[(i)]	4278	SMIT 89 (1211)	1.1	63.00	0.221	0.594	-4	37.4	0.0	18.7	37.4	70.8
Using non-credible surveillance data[(g)]			2.1	80.71	0.221	0.594	-4	47.9	0.0	24.0	47.9	91.9
Intermediate Shell Forging 04	990311 / 298244	-	1.1	86.00	4.46	1.379	-6	118.6	0.0	17.0	34.0	146.6
Intermediate to Lower Shell Circumferential Weld	25531	SMIT 89 (1211)	1.1	56.22	4.44	1.378	-2	77.5	0.0	28.0	56.0	131.5
Using non-credible surveillance data[(h)]			2.1	67.53	4.44	1.378	-2	93.1	0.0	28.0	56.0	147.1
Lower Shell Forging 03	990400 / 292332	-	1.1	119.97	4.54	1.383	33	165.9	0.0	17.0	34.0	232.9
Using non-credible surveillance data[(h)]			2.1	81.68	4.54	1.383	33	113.0	0.0	17.0	34.0	**180.0**

Notes contained on the following page.

Notes:

(a) The Regulatory Guide 1.99, Revision 2 (Reference 1) methodology was utilized in the calculation of the ART values. Dashes indicate when a category is not applicable to the material.

(b) Chemistry factors are taken from Table 3-7 of this report.

(c) Fluence and Fluence Factors are taken from Table 5-2 of this report.

(d) $RT_{NDT(U)}$ values are taken from Table 3-1 of this report.

(e) Per the guidance of Regulatory Guide 1.99, Revision 2 (Reference 1), the base metal σ_Δ = 17°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and σ_Δ = 8.5°F for Position 2.1 with credible surveillance data. Also, per Regulatory Guide 1.99, Revision 2, the weld metal σ_Δ = 28°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and σ_Δ = 14°F for Position 2.1 with credible surveillance data. However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$ for either forgings or welds with or without surveillance data.

(f) The surveillance data for weld Heat # 25295 from the Sequoyah Unit 1 surveillance program were deemed credible per WCAP-17539-NP (Reference 15), Appendix A.

(g) The surveillance data for weld Heat # 4278 from the Sequoyah Unit 2 surveillance program were deemed non-credible per WCAP-17539-NP (Reference 15), Appendix A.

(h) The credibility evaluation for the North Anna Unit 1 surveillance data in Appendix E of this report determined that the Lower Shell Forging 03 and weld Heat # 25531 surveillance data are deemed non-credible.

(i) Since this inner diameter (ID) weld is only 6% of the vessel thickness, the weld is not present at the 1/4T location; hence, it is not applicable to this calculation. It is presented for information only.

Table 5-5 Calculation of the North Anna Unit 1 ART Values at the 3/4T Location for the Reactor Vessel Beltline Materials at 72 EFPY(a)

Material	Heat Number	Flux Type (Lot)	R.G. 1.99, Rev. 2 Position	CF(b)	3/4T Fluence(c) (x 10^{19} n/cm², E > 1.0 MeV)	3/4T FF(c)	$RT_{NDT(U)}$(d) (°F)	Predicted ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ(e) (°F)	M (°F)	ART (°F)
Upper Shell Forging 05	990286 / 295213	-	1.1	121.50	0.0763	0.365	1	44.4	0.0	17.0	34.0	79.4
Upper to Intermediate Shell Circumferential Weld (OD 94%)	25295	SMIT 89 (1170)	1.1	163.25	0.0881	0.392	-40	64.0	0.0	28.0	56.0	80.0
Using credible surveillance data(f)			2.1	150.69	0.0881	0.392	-40	59.1	0.0	14.0	28.0	47.1
Upper to Intermediate Shell Circumferential Weld (ID 6%)(i)	4278	SMIT 89 (1211)	1.1	63.00	0.0881	0.392	-4	24.7	0.0	12.4	24.7	45.4
Using non-credible surveillance data(g)			2.1	80.71	0.0881	0.392	-4	31.6	0.0	15.8	31.6	59.3
Intermediate Shell Forging 04	990311 / 298244	-	1.1	86.00	1.78	1.158	-6	99.6	0.0	17.0	34.0	127.6
Intermediate to Lower Shell Circumferential Weld	25531	SMIT 89 (1211)	1.1	56.22	1.77	1.157	-2	65.0	0.0	28.0	56.0	119.0
Using non-credible surveillance data(h)			2.1	67.53	1.77	1.157	-2	78.1	0.0	28.0	56.0	132.1
Lower Shell Forging 03	990400 / 292332	-	1.1	119.97	1.81	1.163	33	139.5	0.0	17.0	34.0	206.5
Using non-credible surveillance data(h)			2.1	81.68	1.81	1.163	33	95.0	0.0	17.0	34.0	**162.0**

Notes contained on the following page.

Notes:

(a) The Regulatory Guide 1.99, Revision 2 (Reference 1) methodology was utilized in the calculation of the ART values. Dashes indicate when a category is not applicable to the material.

(b) Chemistry factors are taken from Table 3-7 of this report.

(c) Fluence and Fluence Factors are taken from Table 5-2 of this report.

(d) $RT_{NDT(U)}$ values are taken from Table 3-1 of this report.

(e) Per the guidance of Regulatory Guide 1.99, Revision 2 (Reference 1), the base metal σ_{Δ} = 17°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and σ_{Δ} = 8.5°F for Position 2.1 with credible surveillance data. Also, per Regulatory Guide 1.99, Revision 2, the weld metal σ_{Δ} = 28°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and σ_{Δ} = 14°F for Position 2.1 with credible surveillance data. However, σ_{Δ} need not exceed $0.5^{*}\Delta RT_{NDT}$ for either forgings or welds with or without surveillance data.

(f) The surveillance data for weld Heat # 25295 from the Sequoyah Unit 1 surveillance program were deemed credible per WCAP-17539-NP (Reference 15), Appendix A.

(g) The surveillance data for weld Heat # 4278 from the Sequoyah Unit 2 surveillance program were deemed non-credible per WCAP-17539-NP (Reference 15), Appendix A.

(h) The credibility evaluation for the North Anna Unit 1 surveillance data in Appendix E of this report determined that the Lower Shell Forging 03 and weld Heat # 25531 surveillance data are deemed non-credible.

(i) Since this inner diameter (ID) weld is only 6% of the vessel thickness, the weld is not present at the 1/4T location; hence, it is not applicable to this calculation. It is presented for information only.

Table 5-6 Calculation of the North Anna Unit 1 ART Values for the Reactor Vessel Extended Beltline Materials at 72 EFPY[(a)]

Material	Heat Number	Flux Type (Lot)	R.G. 1.99, Rev. 2 Position	CF[(b)]	Surface Fluence[(c)] (x 10^{19} n/cm^2, E > 1.0 MeV)	Surf. FF[(c)]	$RT_{NDT(U)}$[(d)] (°F)	Predicted ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ[(e)] (°F)	M (°F)	ART (°F)
Inlet Nozzle Forging 09 to Upper Shell Weld	Rotterdam	-	1.1	293.45	0.00898	0	30	0.0	0.0	0.0	0.0	30.0
Inlet Nozzle Forging 10 to Upper Shell Weld	Rotterdam	-	1.1	293.45	0.0313	0.225	30	66.1	0.0	28.0	56.0	152.1
Inlet Nozzle Forging 11 to Upper Shell Weld	Rotterdam	-	1.1	293.45	0.0120	0.124	30	36.4	0.0	18.2	36.4	102.7
Outlet Nozzle Forging 12 to Upper Shell Weld	Rotterdam	-	1.1	293.45	0.0182	0.162	30	47.6	0.0	23.8	47.6	125.2
Outlet Nozzle Forging 13 to Upper Shell Weld	Rotterdam	-	1.1	293.45	0.00522	0	30	0.0	0.0	0.0	0.0	30.0
Outlet Nozzle Forging 14 to Upper Shell Weld	Rotterdam	-	1.1	293.45	0.00697	0	30	0.0	0.0	0.0	0.0	30.0
Inlet Nozzle Forging 09	990290-11	-	1.1	96.00	0.00425	0	-14	0.0	0.0	0.0	0.0	-14.0
Inlet Nozzle Forging 10	990290-12	-	1.1	95.75	0.0148	0.142	-10	13.6	0.0	6.8	13.6	17.2
Inlet Nozzle Forging 11	990268-21	-	1.1	140.30	0.00568	0	8	0.0	0.0	0.0	0.0	8.0
Outlet Nozzle Forging 12	990290-31	-	1.1	96.00	0.00875	0	-6	0.0	0.0	0.0	0.0	-6.0
Outlet Nozzle Forging 13	990290-22	-	1.1	96.00	0.00251	0	-7	0.0	0.0	0.0	0.0	-7.0
Outlet Nozzle Forging 14	990290-21	-	1.1	96.00	0.00335	0	8	0.0	0.0	0.0	0.0	8.0

Notes contained on the following page.

Notes:

(a) The Regulatory Guide 1.99, Revision 2 (Reference 1) methodology was utilized in the calculation of the ART values. Dashes indicate when a category is not applicable to the material.

(b) Chemistry factors are taken from Table 3-7 of this report.

(c) Fluence and Fluence Factors are taken from Table 5-2 of this report. The fluence values for the nozzle forgings are taken at the postulated 1/4T flaw axial location. The fluence values for the inlet/outlet nozzle to upper shell welds are taken at the lowest extent of the nozzle weld centerline. Analysis of the nozzle forgings and associated welds are conservatively performed using the surface fluence, neglecting attenuation through the reactor vessel wall. Embrittlement effects are considered only if the fluence is greater than 10^{17} n/cm^2. For materials with fluence less than 10^{17} n/cm^2 the FF is set equal to 0.

(d) $RT_{NDT(U)}$ values are taken from Table 3-1 of this report.

(e) Per the guidance of Regulatory Guide 1.99, Revision 2 (Reference 1), the base metal σ_Δ = 17°F for Position 1.1. Also, per Regulatory Guide 1.99, Revision 2, the weld metal σ_Δ = 28°F for Position 1.1. However, σ_Δ need not exceed 0.5*ΔRT_{NDT} for either forgings or welds with or without surveillance data.

Table 5-7 Calculation of the North Anna Unit 2 ART Values at the 1/4T Location for the Reactor Vessel Beltline Materials at 72 EFPY[(a)]

Material	Heat Number	Flux Type (Lot)	R.G. 1.99, Rev. 2 Position	CF[(b)]	1/4T Fluence[(c)] (x 10^{19} n/cm^2, E > 1.0 MeV)	1/4T FF[(c)]	$RT_{NDT(U)}$[(d)] (°F)	Predicted ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ[(e)] (°F)	M (°F)	ART (°F)
Upper Shell Forging 05	990598 / 291396	-	1.1	51.00	0.194	0.562	8	28.7	0.0	14.3	28.7	65.3
Upper to Intermediate Shell Circumferential Weld (OD 94%)	4278	SMIT 89 (1211)	1.1	63.00	0.224	0.597	-4	37.6	0.0	18.8	37.6	71.2
Using non-credible surveillance data[(f)]			2.1	80.71	0.224	0.597	-4	48.2	0.0	24.1	48.2	92.3
Upper to Intermediate Shell Circumferential Weld (ID 6%)[(h)]	801	SMIT 89 (1211)	1.1	87.80	0.224	0.597	10	52.4	0.0	26.2	52.4	114.8
Intermediate Shell Forging 04	990496 / 292424	-	1.1	74.00	4.54	1.383	69	102.3	0.0	17.0	34.0	205.3
Using non-credible surveillance data[(g)]			2.1	53.44	4.54	1.383	69	73.9	0.0	17.0	34.0	176.9
Intermediate to Lower Shell Circumferential Weld	716126	LW320 (26)	1.1	36.09	4.53	1.382	-67	49.9	0.0	24.9	49.9	32.8
Using credible surveillance data[(g)]			2.1	26.61	4.53	1.382	-67	36.8	0.0	14.0	28.0	-2.2
Lower Shell Forging 03	990533 / 297355	-	1.1	96.00	4.63	1.387	37	133.2	0.0	17.0	34.0	**204.2**

Notes contained on the following page.

Notes:

(a) The Regulatory Guide 1.99, Revision 2 (Reference 1) methodology was utilized in the calculation of the ART values. Dashes indicate when a category is not applicable to the material.

(b) Chemistry factors are taken from Table 3-8 of this report.

(c) Fluence and Fluence Factors are taken from Table 5-3 of this report.

(d) $RT_{NDT(U)}$ values are taken from Table 3-2 of this report.

(e) Per the guidance of Regulatory Guide 1.99, Revision 2 (Reference 1), the base metal σ_Δ = 17°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and σ_Δ = 8.5°F for Position 2.1 with credible surveillance data. Also, per Regulatory Guide 1.99, Revision 2, the weld metal σ_Δ = 28°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and σ_Δ = 14°F for Position 2.1 with credible surveillance data. However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$ for either forgings or welds with or without surveillance data.

(f) The surveillance data for weld Heat # 4278 from the Sequoyah Unit 2 surveillance program were deemed non-credible per WCAP-17539-NP (Reference 15), Appendix A.

(g) The credibility evaluation for the North Anna Unit 2 surveillance data in Appendix E of this report determined that the Intermediate Shell Forging 04 surveillance data are deemed non-credible; however, the weld Heat # 716126 surveillance data are deemed credible.

(h) Since this inner diameter (ID) weld is only 6% of the vessel thickness, the weld is not present at the 1/4T location; hence, it is not applicable to this calculation. It is presented for information only.

Table 5-8 Calculation of the North Anna Unit 2 ART Values at the 3/4T Location for the Reactor Vessel Beltline Materials at 72 EFPY[(a)]

Material	Heat Number	Flux Type (Lot)	R.G. 1.99, Rev. 2 Position	CF[(b)]	3/4T Fluence[(c)] (x 10^{19} n/cm^2, E > 1.0 MeV)	3/4T FF[(c)]	$RT_{NDT(U)}$[(d)] (°F)	Predicted ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ[(e)] (°F)	M (°F)	ART (°F)
Upper Shell Forging 05	990598 / 291396	-	1.1	51.00	0.0771	0.367	8	18.7	0.0	9.4	18.7	45.4
Upper to Intermediate Shell Circumferential Weld (OD 94%)	4278	SMIT 89 (1211)	1.1	63.00	0.0891	0.394	-4	24.8	0.0	12.4	24.8	45.7
Using non-credible surveillance data[(f)]			2.1	80.71	0.0891	0.394	-4	31.8	0.0	15.9	31.8	59.6
Upper to Intermediate Shell Circumferential Weld (ID 6%)[(h)]	801	SMIT 89 (1211)	1.1	87.80	0.0891	0.394	10	34.6	0.0	17.3	34.6	79.2
Intermediate Shell Forging 04	990496 / 292424	-	1.1	74.00	1.81	1.163	69	86.0	0.0	17.0	34.0	189.0
Using non-credible surveillance data[(g)]			2.1	53.44	1.81	1.163	69	62.1	0.0	17.0	34.0	165.1
Intermediate to Lower Shell Circumferential Weld	716126	LW320 (26)	1.1	36.09	1.80	1.162	-67	41.9	0.0	21.0	41.9	16.9
Using credible surveillance data[(g)]			2.1	26.61	1.80	1.162	-67	30.9	0.0	14.0	28.0	-8.1
Lower Shell Forging 03	990533 / 297355	-	1.1	96.00	1.84	1.168	37	112.1	0.0	17.0	34.0	**183.1**

Notes contained on the following page.

Notes:

(a) The Regulatory Guide 1.99, Revision 2 (Reference 1) methodology was utilized in the calculation of the ART values. Dashes indicate when a category is not applicable to the material.

(b) Chemistry factors are taken from Table 3-8 of this report.

(c) Fluence and Fluence Factors are taken from Table 5-3 of this report.

(d) $RT_{NDT(U)}$ values are taken from Table 3-2 of this report.

(e) Per the guidance of Regulatory Guide 1.99, Revision 2 (Reference 1), the base metal σ_Δ = 17°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and σ_Δ = 8.5°F for Position 2.1 with credible surveillance data. Also, per Regulatory Guide 1.99, Revision 2, the weld metal σ_Δ = 28°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and σ_Δ = 14°F for Position 2.1 with credible surveillance data. However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$ for either forgings or welds with or without surveillance data.

(f) The surveillance data for weld Heat # 4278 from the Sequoyah Unit 2 surveillance program were deemed non-credible per WCAP-17539-NP (Reference 15), Appendix A.

(g) The credibility evaluation for the North Anna Unit 2 surveillance data in Appendix E of this report determined that the Intermediate Shell Forging 04 surveillance data are deemed non-credible; however, the weld Heat # 716126 surveillance data are deemed credible.

(h) Since this inner diameter (ID) weld is only 6% of the vessel thickness, the weld is not present at the 1/4T location; hence, it is not applicable to this calculation. It is presented for information only.

Table 5-9 Calculation of the North Anna Unit 2 ART Values for the Reactor Vessel Extended Beltline Materials at 72 EFPY[(a)]

Material	Heat Number	Flux Type (Lot)	R.G. 1.99, Rev. 2 Position	CF[(b)]	Surface Fluence[(c)] (x 10^{19} n/cm^2, E > 1.0 MeV)	Surf. FF[(c)]	$RT_{NDT(U)}$[(d)] (°F)	Predicted ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ[(e)] (°F)	M (°F)	ART (°F)
Inlet Nozzle Forging 09 to Upper Shell Weld	8816 20459 27622	LW320 (26 & 28)	1.1	163.20	0.00826	0	30	0.0	0.0	0.0	0.0	30.0
Inlet Nozzle Forging 10 to Upper Shell Weld			1.1	163.20	0.0314	0.226	30	36.8	0.0	18.4	36.8	103.6
Inlet Nozzle Forging 11 to Upper Shell Weld			1.1	163.20	0.0118	0.123	30	20.0	0.0	10.0	20.0	70.0
Outlet Nozzle Forging 12 to Upper Shell Weld			1.1	163.20	0.0182	0.162	30	26.5	0.0	13.2	26.5	82.9
Outlet Nozzle Forging 13 to Upper Shell Weld			1.1	163.20	0.00479	0	30	0.0	0.0	0.0	0.0	30.0
Outlet Nozzle Forging 14 to Upper Shell Weld			1.1	163.20	0.00687	0	30	0.0	0.0	0.0	0.0	30.0
Inlet Nozzle Forging 09	990426	-	1.1	150.40	0.00390	0	11	0.0	0.0	0.0	0.0	11.0
Inlet Nozzle Forging 10	54567-2	-	1.1	104.75	0.0148	0.142	5	14.9	0.0	7.4	14.9	34.8
Inlet Nozzle Forging 11	54590-2	-	1.1	118.25	0.00559	0	-31	0.0	0.0	0.0	0.0	-31.0
Outlet Nozzle Forging 12	990426-22	-	1.1	150.00	0.00875	0	8	0.0	0.0	0.0	0.0	8.0
Outlet Nozzle Forging 13	990426-31	-	1.1	149.60	0.00230	0	1	0.0	0.0	0.0	0.0	1.0
Outlet Nozzle Forging 14	791291	-	1.1	86.00	0.00330	0	-22	0.0	0.0	0.0	0.0	-22.0

Notes contained on the following page.

Notes:

(a) The Regulatory Guide 1.99, Revision 2 (Reference 1) methodology was utilized in the calculation of the ART values. Dashes indicate when a category is not applicable to the material.

(b) Chemistry factors are taken from Table 3-8 of this report.

(c) Fluence and Fluence Factors are taken from Table 5-3 of this report. The fluence values for the nozzle forgings are taken at the postulated 1/4T flaw axial location. The fluence values for the inlet/outlet nozzle to upper shell welds are taken at the lowest extent of the nozzle weld centerline. Analysis of the nozzle forgings and associated welds are conservatively performed using the surface fluence, neglecting attenuation through the reactor vessel wall. Embrittlement effects are considered only if the fluence is greater than 10^{17} n/cm^2. For materials with fluence less than 10^{17} n/cm^2 the FF is set equal to 0.

(d) $RT_{NDT(U)}$ values are taken from Table 3-2 of this report.

(e) Per the guidance of Regulatory Guide 1.99, Revision 2 (Reference 1), the base metal σ_Δ = 17°F for Position 1.1. Also, per Regulatory Guide 1.99, Revision 2, the weld metal σ_Δ = 28°F for Position 1.1. However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$ for either forgings or welds with or without surveillance data.

6 HEATUP AND COOLDOWN PRESSURE-TEMPERATURE LIMIT CURVES

Table 5-1 shows that the SLR ART values at the 1/4T and 3/4T locations remain bounded by the ART values used in the current P-T limit curves. Therefore, the P-T limit curves implemented in the North Anna Units 1 and 2 Technical Specifications will remain valid through SLR (72.0 EFPY) for the cylindrical shell materials. Appendix B demonstrates that the current P-T limits for the cylindrical beltline region bound the P-T limits for the reactor vessel inlet and outlet nozzles for North Anna Units 1 and 2 at 72 EFPY.

However, in order to evaluate the amount of margin inherent to the current Technical Specifications P-T limit curves, the pressure-temperature limit curves for normal heatup and cooldown of the primary reactor coolant system have been calculated in the reactor vessel cylindrical beltline region using the methods discussed in Sections 4 and 5 of this report and the material conditions projected at SLR presented in Table 5-1. The approved methodology is also presented in WCAP-14040-A, Revision 4 (Reference 2).

Figure 6-1 presents the limiting heatup curves without margins for possible instrumentation errors using heatup rates of 20, 40, and 60°F/hr applicable for 72 EFPY, with the flange requirements. Figure 6-2 presents the limiting cooldown curves without margins for possible instrumentation errors using cooldown rates of -100, -60, -40, -20, and 0°F/hr (steady-state) applicable for 72 EFPY, with the flange requirements. Both Figure 6-1 and Figure 6-2 use the "Axial Flaw" methodology and the limiting "Axial Flaw" ART values summarized in Table 5-1. The heatup and cooldown curves were generated using the 1998 Edition through the 2000 Addenda ASME Code Section XI, Appendix G.

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

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

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

where,

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

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

T = the minimum permissible metal temperature, and

RT_{NDT} = metal reference nil-ductility temperature.

The criticality limit curve specifies pressure-temperature limits for core operation in order to provide additional margin during actual power production. The pressure-temperature limits for core operation (except for low power physics tests) are that: 1) the reactor vessel must be at a temperature equal to or

higher than the minimum temperature required for the in-service hydrostatic test, and 2) the reactor vessel must be at least 40°F higher than the minimum permissible temperature in the corresponding pressure-temperature curve for heatup and cooldown calculated as described in Section 4 of this report. For the heatup and cooldown curves without margins for instrumentation errors, the minimum temperature for the in-service hydrostatic leak tests for the North Anna Units 1 and 2 reactor vessel at 72 EFPY is 262°F. This temperature is the minimum permissible temperature at which design pressure can be reached during a hydrostatic test per Equation (13). The vertical line drawn from these points on the pressure-temperature curve, intersecting a curve 40°F higher than the pressure-temperature limit curve, constitutes the limit for core operation for the reactor vessel.

Figures 6-1 and 6-2 define all of the preceding limits for ensuring prevention of non-ductile failure for the North Anna Units 1 and 2 reactor vessel for 72 EFPY with the flange requirements and without instrumentation uncertainties. The data points used for developing the heatup and cooldown P-T limit curves shown in Figures 6-1 and 6-2 are presented in Tables 6-1 through 6-3. The P-T limit curves shown in Figures 6-1 and 6-2 were generated based on the limiting "Axial Flaw" ART values for the cylindrical beltline and extended beltline reactor vessel materials.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: North Anna Unit 2 Lower Shell Forging 03 (Position 1.1)
LIMITING ART VALUES AT 72 EFPY: 1/4T, 205°F (Axial Flaw)
3/4T, 184°F (Axial Flaw)
Limiting Flange RT_{NDT} = -22°F.

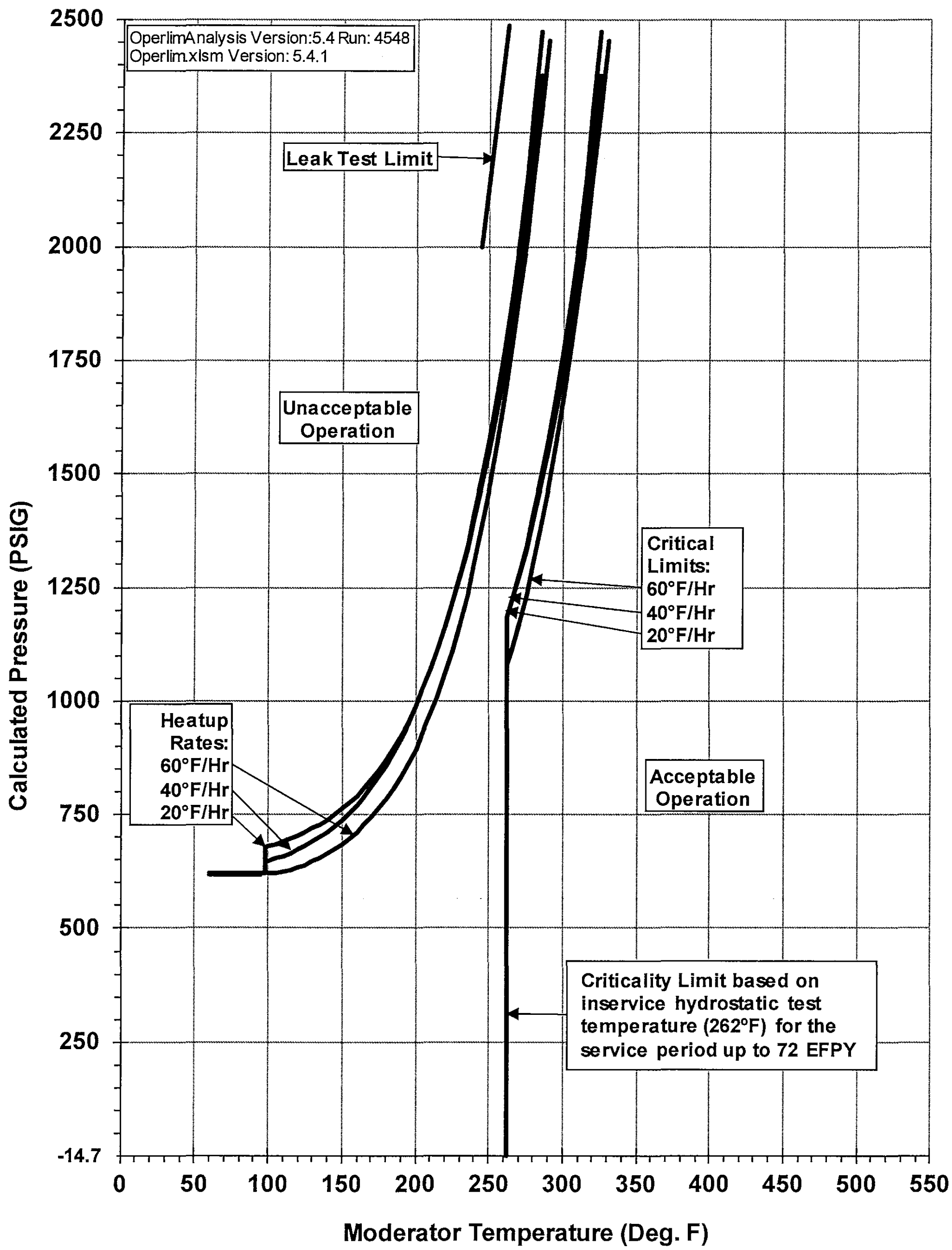

Figure 6-1 North Anna Units 1 and 2 Heatup Curves for 20, 40, and 60°F/hr Applicable to 72 EFPY Based on the K_{Ic} Methodology of the 1998 through the 2000 Addenda Edition of ASME Code, Section XI, App. G, Without Margins for Instrument Error or Pressure Correction, and With Flange Requirements

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: North Anna Unit 2 Lower Shell Forging 03 (Position 1.1)
LIMITING ART VALUES AT 72 EFPY: 1/4T, 205°F (Axial Flaw)
3/4T, 184°F (Axial Flaw)
Limiting Flange RT_{NDT} = -22°F.

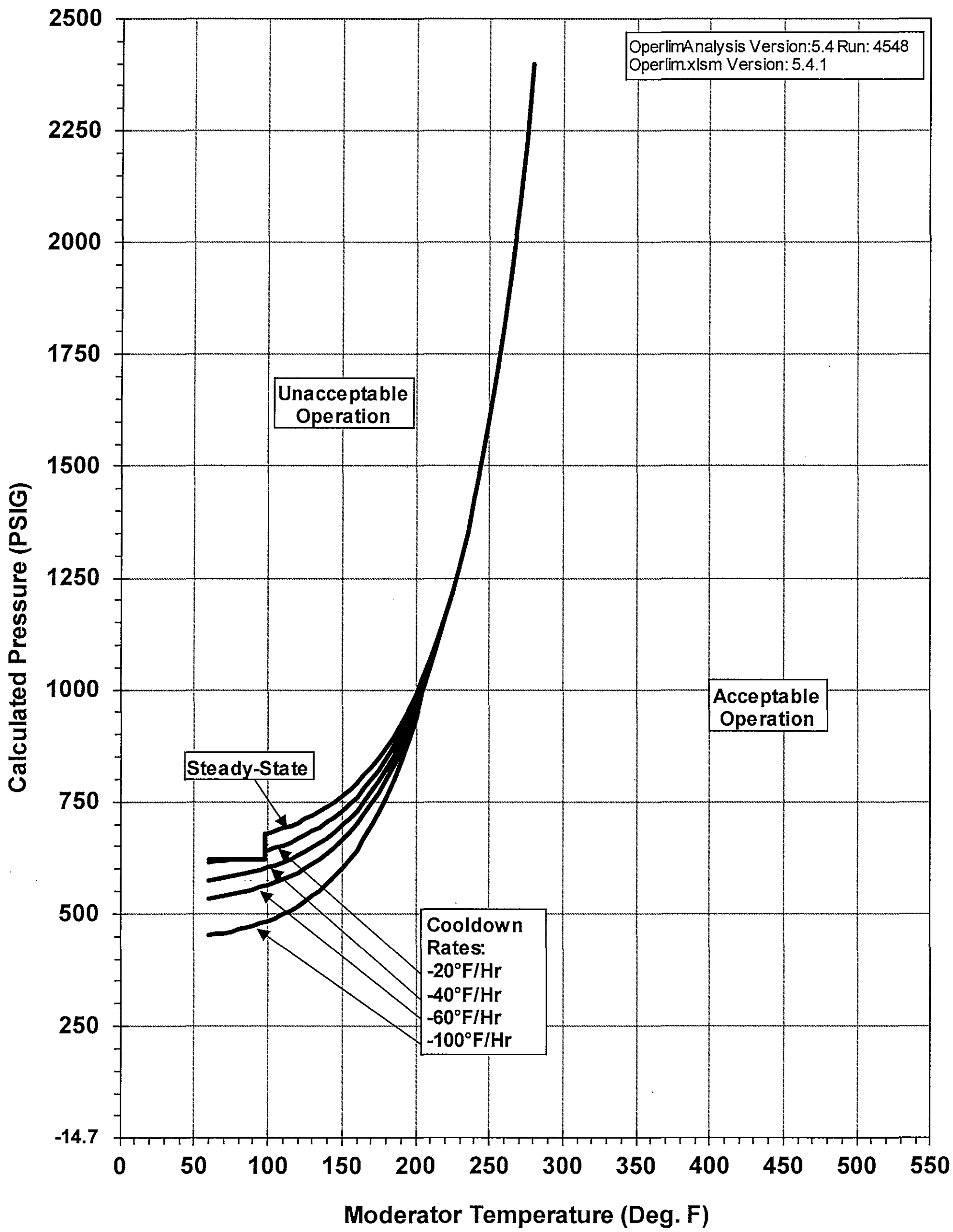

Figure 6-2 North Anna Units 1 and 2 Steady-State and Cooldown Curves for -20, -40, -60, and -100°F/hr Applicable to 72 EFPY Based on the K_{Ic} Methodology of the 1998 through the 2000 Addenda Edition of ASME Code, Section XI, App. G, Without Margins for Instrument Error or Pressure Correction, and With Flange Requirements

Table 6-1 North Anna Units 1 and 2 72 EFPY Heatup Curves Data Points using the 1998 Edition through the 2000 Addenda App. G Methodology (w/ K_{Ic}, w/ Flange Requirements, and w/o Margins for Instrumentation Errors or Pressure Correction)

20°F/hr Heatup		20°F/hr Criticality		40°F/hr Heatup		40°F/hr Criticality		60°F/hr Heatup		60°F/hr Criticality	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
60	621	262	-14.7	60	621	262	-14.7	60	619	262	-14.7
65	621	262	1186	65	621	262	1186	65	619	262	1079
70	621	265	1220	70	621	265	1220	70	619	265	1110
75	621	270	1277	75	621	270	1276	75	619	270	1168
80	621	275	1340	80	621	275	1334	80	619	275	1232
85	621	280	1409	85	621	280	1398	85	619	280	1302
90	621	285	1486	90	621	285	1469	90	619	285	1379
95	621	290	1571	95	621	290	1547	95	619	290	1465
98	621	295	1665	98	621	295	1634	100	620	295	1560
98	678	300	1769	98	647	300	1729	105	621	300	1664
100	680	305	1883	100	649	305	1835	110	624	305	1779
105	685	310	2010	105	653	310	1951	115	628	310	1901
110	690	315	2150	110	659	315	2080	120	633	315	2020
115	697	320	2304	115	665	320	2222	125	639	320	2150
120	703	325	2474	120	673	325	2378	130	646	325	2295
125	711	-	-	125	681	-	-	135	654	330	2454
130	719	-	-	130	690	-	-	140	663	-	-
135	729	-	-	135	700	-	-	145	673	-	-
140	739	-	-	140	711	-	-	150	685	-	-
145	750	-	-	145	724	-	-	155	697	-	-
150	763	-	-	150	738	-	-	160	711	-	-
155	776	-	-	155	753	-	-	165	727	-	-
160	792	-	-	160	770	-	-	170	744	-	-
165	809	-	-	165	789	-	-	175	763	-	-
170	827	-	-	170	809	-	-	180	784	-	-
175	848	-	-	175	832	-	-	185	808	-	-
180	871	-	-	180	858	-	-	190	834	-	-
185	896	-	-	185	886	-	-	195	863	-	-
190	924	-	-	190	916	-	-	200	894	-	-

Table 6-1 North Anna Units 1 and 2 72 EFPY Heatup Curves Data Points using the 1998 Edition through the 2000 Addenda App. G Methodology (w/ K_{Ic}, w/ Flange Requirements, and w/o Margins for Instrumentation Errors or Pressure Correction)

20°F/hr Heatup		20°F/hr Criticality		40°F/hr Heatup		40°F/hr Criticality		60°F/hr Heatup		60°F/hr Criticality	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
195	954	-	-	195	951	-	-	205	929	-	-
200	988	-	-	200	988	-	-	210	968	-	-
205	1026	-	-	205	1026	-	-	215	1011	-	-
210	1067	-	-	210	1067	-	-	220	1058	-	-
215	1113	-	-	215	1113	-	-	225	1110	-	-
220	1164	-	-	220	1164	-	-	230	1168	-	-
225	1220	-	-	225	1220	-	-	235	1232	-	-
230	1277	-	-	230	1276	-	-	240	1302	-	-
235	1340	-	-	235	1334	-	-	245	1379	-	-
240	1409	-	-	240	1398	-	-	250	1465	-	-
245	1486	-	-	245	1469	-	-	255	1560	-	-
250	1571	-	-	250	1547	-	-	260	1664	-	-
255	1665	-	-	255	1634	-	-	265	1779	-	-
260	1769	-	-	260	1729	-	-	270	1901	-	-
265	1883	-	-	265	1835	-	-	275	2020	-	-
270	2010	-	-	270	1951	-	-	280	2150	-	-
275	2150	-	-	275	2080	-	-	285	2295	-	-
280	2304	-	-	280	2222	-	-	290	2454	-	-
285	2474	-	-	285	2378	-	-	-	-	-	-

Table 6-2 North Anna Units 1 and 2 72 EFPY Leak Test Data Points using the 1998 Edition through the 2000 Addenda Appendix G Methodology (w/ K_{Ic}, w/ Flange Requirements, and w/o Margins for Instrumentation Errors or Pressure Correction)

Temperature (°F)	Pressure (psig)
244.5	2000
262	2485

Table 6-3 North Anna Units 1 and 2 72 EFPY Cooldown Curves Data Points using the 1998 Edition through the 2000 Addenda App. G Methodology (w/ K_{Ic}, w/ Flange Requirements, and w/o Margins for Instrumentation Errors or Pressure Correction)

Steady-State		-20°F/hr		-40°F/hr		-60°F/hr		-100°F/hr	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
60	621	60	614	60	575	60	535	60	452
65	621	65	617	65	577	65	537	65	455
70	621	70	619	70	580	70	540	70	458
75	621	75	621	75	583	75	543	75	461
80	621	80	621	80	586	80	546	80	464
85	621	85	621	85	590	85	550	85	469
90	621	90	621	90	594	90	554	90	473
95	621	95	621	95	598	95	559	95	478
98	621	98	621	100	603	100	564	100	484
98	678	98	640	105	609	105	570	105	491
100	680	100	642	110	615	110	577	110	499
105	685	105	647	115	622	115	584	115	507
110	690	110	653	120	630	120	592	120	516
115	697	115	659	125	638	125	601	125	527
120	703	120	667	130	648	130	611	130	539
125	711	125	675	135	658	135	623	135	552
130	719	130	684	140	670	140	635	140	566
135	729	135	693	145	683	145	649	145	582
140	739	140	704	150	697	150	664	150	601
145	750	145	716	155	713	155	682	155	621
150	763	150	730	160	731	160	701	160	643
155	776	155	745	165	750	165	722	165	668
160	792	160	761	170	772	170	745	170	695
165	809	165	779	175	796	175	771	175	726
170	827	170	799	180	822	180	800	180	760
175	848	175	821	185	852	185	832	185	798
180	871	180	846	190	884	190	867	190	840
185	896	185	873	195	920	195	906	195	886
190	924	190	903	200	960	200	949	200	938

Table 6-3 North Anna Units 1 and 2 72 EFPY Cooldown Curves Data Points using the 1998 Edition through the 2000 Addenda App. G Methodology (w/ K_{Ic}, w/ Flange Requirements, and w/o Margins for Instrumentation Errors or Pressure Correction)

Steady-State		-20°F/hr		-40°F/hr		-60°F/hr		-100°F/hr	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
195	954	195	936	205	1004	205	997	205	995
200	988	200	973	210	1053	210	1051	210	1051
205	1026	205	1014	215	1107	215	1107	215	1107
210	1067	210	1059	220	1163	220	1163	220	1163
215	1113	215	1108	225	1220	225	1220	225	1220
220	1164	220	1163	230	1282	230	1282	230	1282
225	1220	225	1220	235	1350	235	1350	235	1350
230	1282	230	1282	240	1426	240	1426	240	1426
235	1350	235	1350	245	1509	245	1509	245	1509
240	1426	240	1426	250	1601	250	1601	250	1601
245	1509	245	1509	255	1703	255	1703	255	1703
250	1601	250	1601	260	1816	260	1816	260	1816
255	1703	255	1703	265	1941	265	1941	265	1941
260	1816	260	1816	270	2078	270	2078	270	2078
265	1941	265	1941	275	2231	275	2231	275	2231
270	2078	270	2078	280	2399	280	2399	280	2399
275	2231	275	2231	-	-	-	-	-	-
280	2399	280	2399	-	-	-	-	-	-

7 HEATUP AND COOLDOWN LIMITS APPLICABILITY AND MARGIN ASSESSMENT

This section provides a comparison of the Heatup and Cooldown P-T limit curves currently implemented in the North Anna Units 1 and 2 Technical Specifications (Reference 18) and the Heatup and Cooldown P-T limit curves generated within this report.

The curves developed in this report (through 72 EFPY; without margins for instrumentation errors) are shown as solid lines in Figures 7-1 through 7-4, while the curves developed from the data points (through EOLE; without margins for instrumentation errors) are shown as dashed lines in Figures 7-1 through 7-4. Data from WCAP-15112 (Reference 17) was chosen for comparison because it represents the basis for the current Technical Specifications P-T limit curves (without adjustments for uncertainties or pressure correction). The color scheme in Figures 7-1 through 7-4 correlates so that the solid and dashed lines have an identical colors for each corresponding heatup or cooldown rate.

Figure 7-1 provides a side-by-side comparison of the heatup curves, with Table 7-2 providing the quantification of the margin between the two curves. Figure 7-2 shows a magnified version of Figure 7-1 in the lower pressure and temperature region. Table 7-4 contains a summary of the available margin.

Figure 7-3 provides a side-by-side comparison of the cooldown curves, with Table 7-3 providing the quantification of the margin between the two curves. Figure 7-4 shows a magnified version of Figure 7-3 in the lower pressure and temperature region. Table 7-5 contains a summary of the available margin.

Per Tables 7-4 and 7-5, the minimum pressure difference (at constant temperature) between the current P-T limit curves and the new curves developed herein is 0 psid. However, the 0 psid margin is driven by the 10 CFR 50, Appendix G flange requirements (minimum temperature = minimum flange $RT_{NDT(U)}$ + 120°F at > 20% of the hydrostatic test pressure) which are identical for both sets of curves since both curves use a limiting RT_{NDT} value of -22°F. The minimum margin associated with the ASME Section XI, Appendix G calculation is 1 psid, which applies to the 60°F/hr heatup curves below 100°F. The pressure margin is much larger in the higher temperature ranges, as demonstrated in Tables 7-3 and 7-4. Using visual comparison of the current Technical Specifications P-T limit curves and the new curves, a minimum temperature difference (at constant pressure) of no less than 10°F is identified. The curve comparisons demonstrate that the current EOLE curves are equal to or bounding at all pressure/temperature combinations.

Additionally, the minimum temperature for criticality of 541°F from Technical Specifications 3.4.2 bounds the criticality curves developed herein. Thus, no changes are required to the minimum criticality temperature.

These comparisons are made without instrument uncertainties or pressure corrections. Dominion has reevaluated the instrument uncertainties and pressure corrections. A comparison of the recalculated correction factors with those used in the Technical Specifications P-T limit curves are identified in Table 7-1.

Table 7-1 North Anna Units 1 and 2 Pressure and Temperature Correction Factors

Type	Current TS Value[(a)]	Revised Value[(b)]	Units
Pressure adjustment for head loss.	57[(c)]	59.06 (≥ 180°F) 56.1 (< 180°F)	psid
Pressure correction for instrument uncertainty.	70.1	68.58	psid
Temperature correction for instrument uncertainty.	13.5	11.2	°F

Notes:

(a) Values were taken from Reference 20.

(b) Values were taken from SM-908, Addendum E (Reference 21).

(c) This value considers one reactor coolant pump (RCP), two RCP, and three RCP operation.

These correction factor changes are generally in the conservative direction; therefore, the changes do not need to be qualified. The exception is the pressure adjustment at high temperatures, i.e. ≥ 180°F. This non-conservative change is accounted for with the margin between the two sets of P-T limit curves. Per Tables 7-4 and 7-5, at ≥ 180°F, the margin is greater than 50 psid. Therefore, there is sufficient margin to account for the 2.06 psid non-conservative increase in the pressure adjustment (59.06 psid - 57 psid). Therefore, when the margin and adjustments are considered together, the current Technical Specifications P-T limit curves remain conservative.

P-T Limit Curve Applicability Conclusion

In conclusion, the margins between the curves developed herein and the current Technical Specifications P-T limit curves illustrate that the current Technical Specifications P-T limit curves remain applicable through 72 EFPY for the beltline region. Since the P-T limits remain applicable, the current Technical Specifications P-T limit curves remain valid through SLR.

Low Temperature Overpressure Protection (LTOP) Applicability Conclusion

The maximum allowable Low Temperature Overpressure Protection System (LTOPS) pressurizer Power Operated Relief Valve (PORV) setpoint was calculated to be ≤ 400 psig when any RCS cold leg temperature is ≤ 180°F and ≤ 558 psig when any RCS cold leg temperature is ≤ 280°F for the North Anna Units 1 and 2 SLR program. The calculation was performed in accordance with the WCAP-14040-A (Reference 2) methodology using critical LTOPS input parameters provided by Dominion, updated results of the design basis mass injection (MI) and heat injection (HI) transients, and the limiting axial flaw steady state Appendix G limits from WCAP-15112 (Reference 17) that were determined to be applicable for SLR through 72 EFPY for North Anna Units 1 and 2.

The evaluation showed that the current Technical Specifications value of ≤ 375 psig when any RCS cold leg temperature is ≤ 180°F and ≤ 540 psig when any RCS cold leg temperature is ≤ 280°F maintain margin to the maximum allowable settings calculated for SLR throughout the range of LTOP applicability. Therefore, the current LTOPS settings are bounding and can be maintained for SLR through 72 EFPY for North Anna Units 1 and 2.

Summary of Conclusions

- The current P-T limit curves in the North Anna Power Station Technical Specifications (Reference 18) remain valid through 72 EFPY.
- The nozzle P-T limit curves (documented in Appendix B) are bounded by the current North Anna Power Station Technical Specifications (Reference 18) P-T limit curves through 72 EFPY, and other Reactor Coolant Pressure Boundary ferritic components have been addressed (see Appendix C).
- The current Technical Specifications PORV setpoints remain valid for SLR through 72 EFPY.

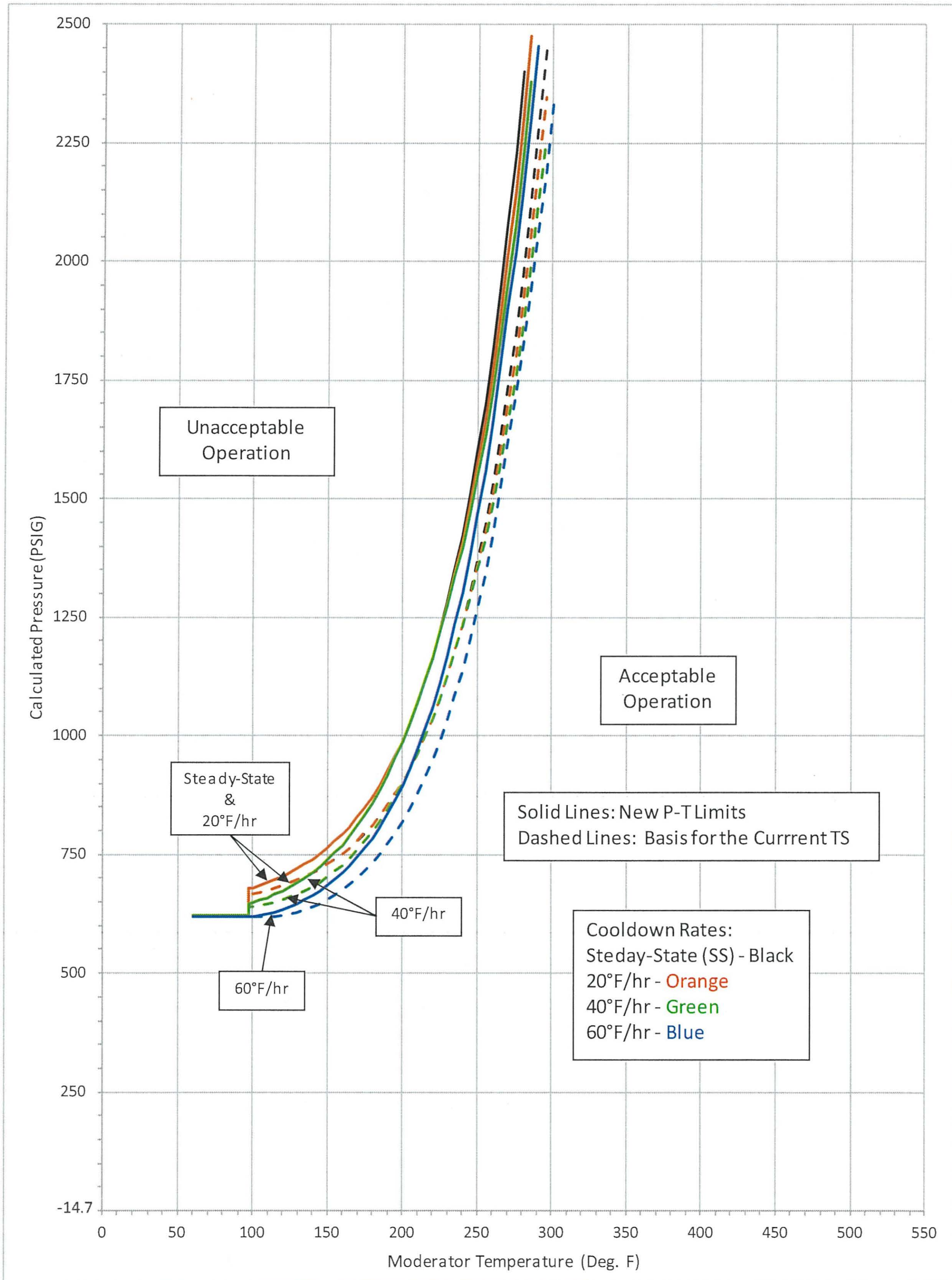

Figure 7-1 North Anna Units 1 and 2 Curve Comparison Between the Current and Newly Developed P-T Limit Curves for 20, 40, and 60°F/hr Heatup

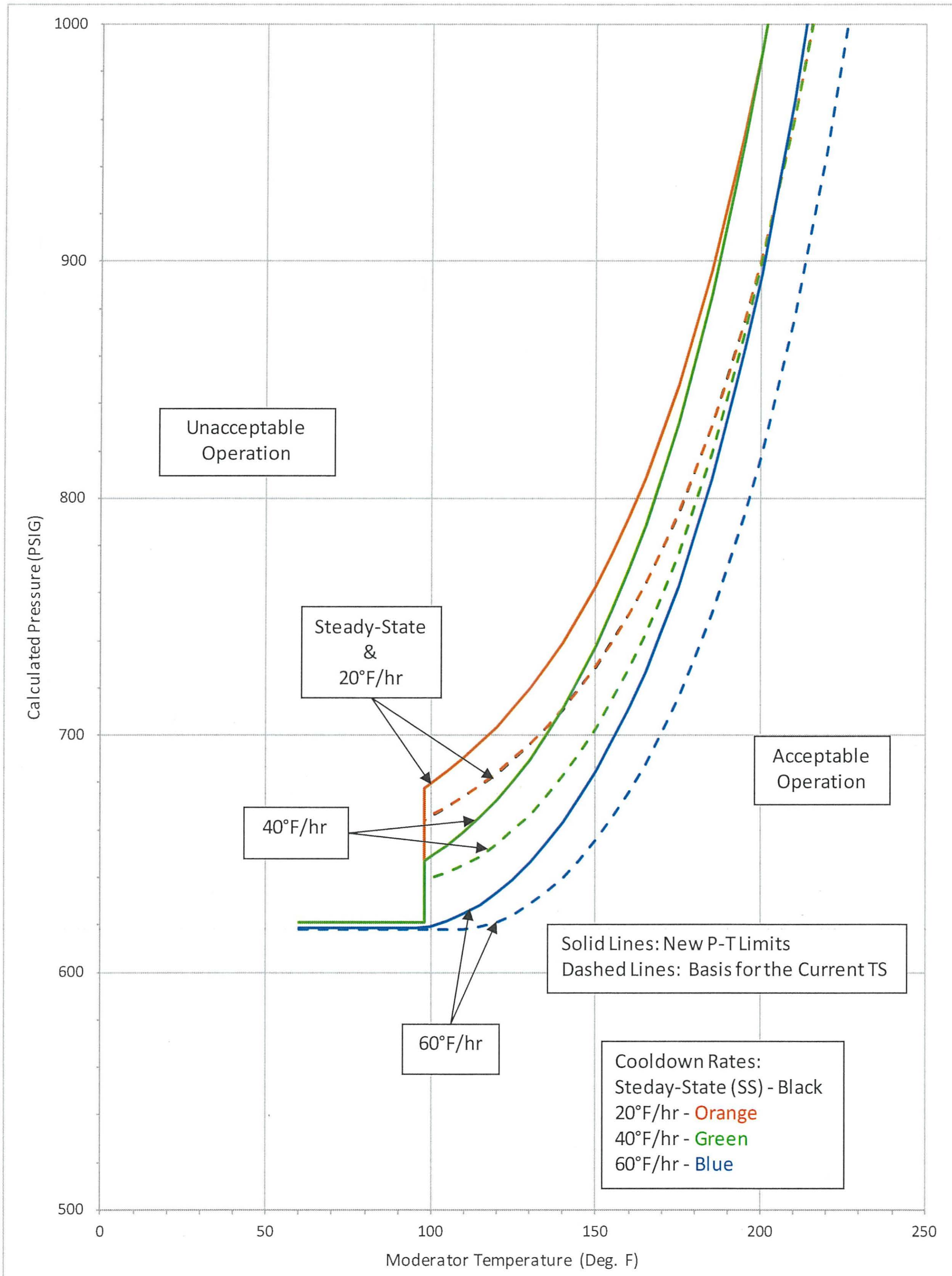

Figure 7-2 North Anna Units 1 and 2 Curve Comparison Between the Current and Newly Developed P-T Limit Curves for 20, 40, and 60°F/hr Heatup (Magnified)

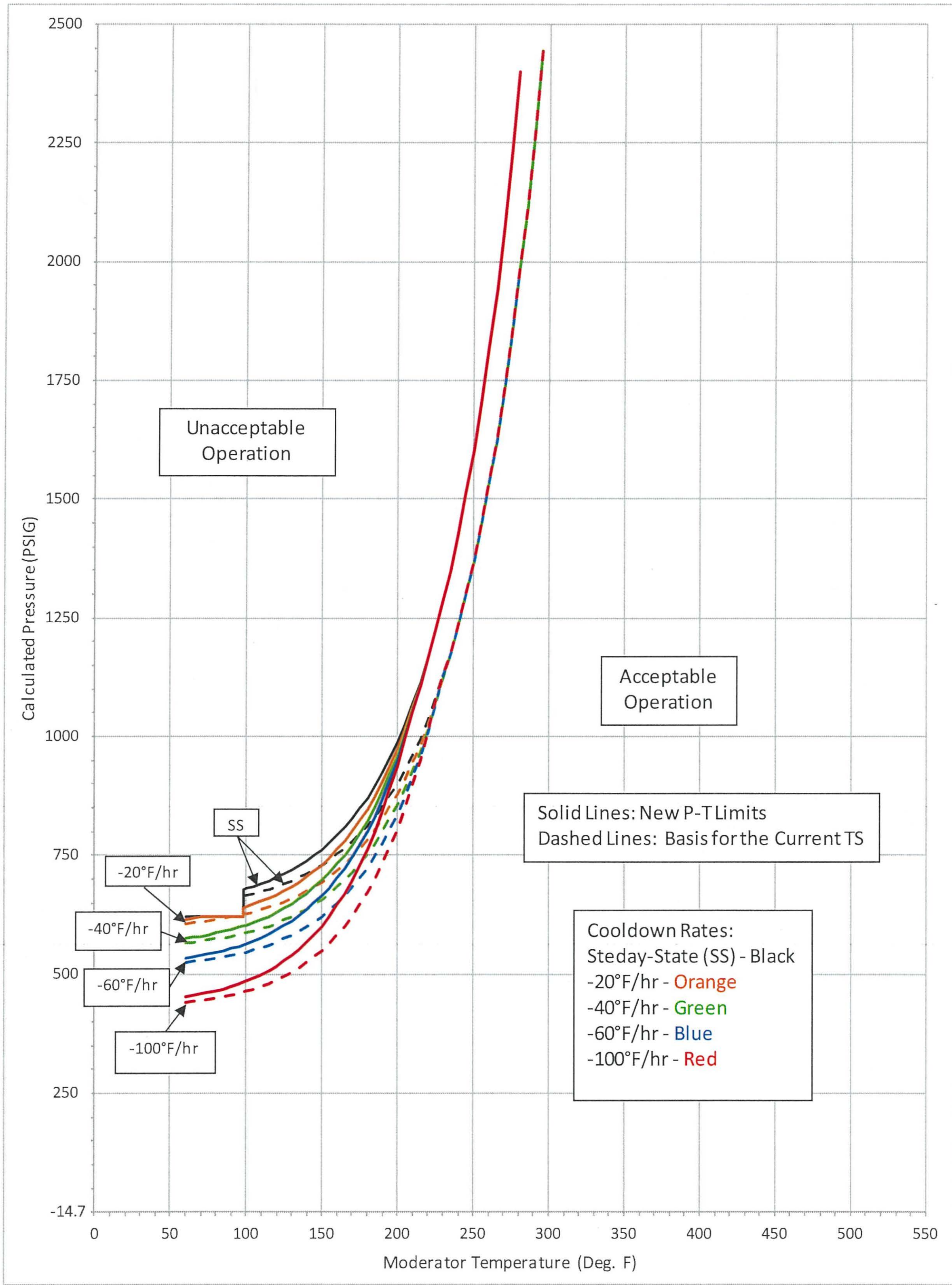

Figure 7-3 North Anna Units 1 and 2 Curve Comparison Between the Current and Newly Developed P-T Limit Curves for Steady-State and 0, -20, -40, -60, and -100°F/hr. Cooldown

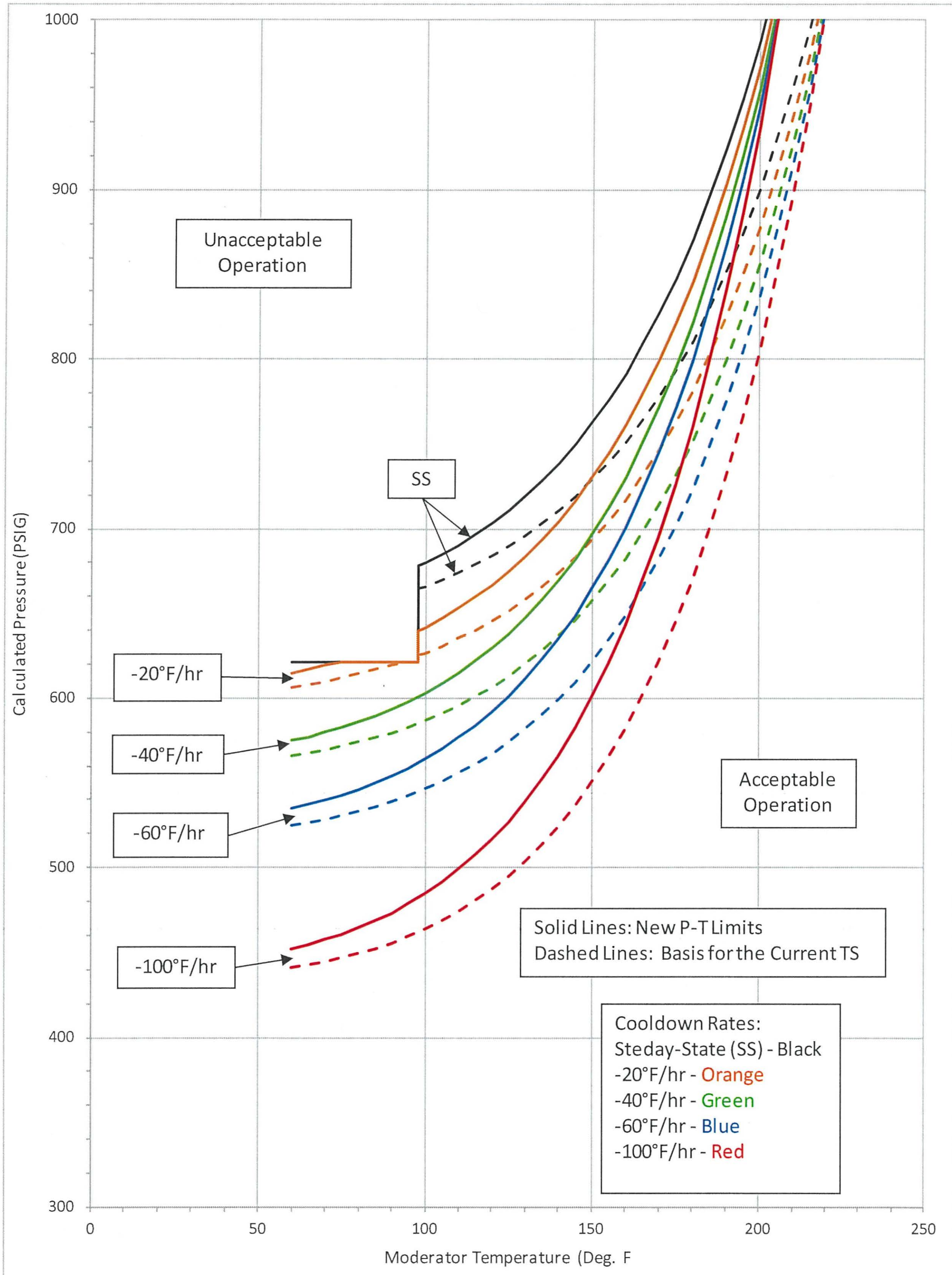

Figure 7-4 North Anna Units 1 and 2 Curve Comparison Between the Current and Newly Developed P-T Limit Curves for Steady-State and 0, -20, -40, -60, and -100°F/hr. Cooldown (Magnified)

Table 7-2 North Anna Units 1 and 2 Heatup P-T Limit Curve Pressure Margin Summary Between the Current P-T Limit Curves and the New P-T Limit Curves to 72 EFPY

Temp. (°F)	20°F/hr. Heatup			40°F/hr. Heatup			60°F/hr. Heatup		
	New[(a)] (psig)	TS[(b)] (psig)	Margin[(c)] (psid)	New[(a)] (psig)	TS[(b)] (psig)	Margin[(c)] (psid)	New[(a)] (psig)	TS[(b)] (psig)	Margin[(c)] (psid)
60	621	621	0	621	621	0	619	618	1
65	621	621	0	621	621	0	619	618	1
70	621	621	0	621	621	0	619	618	1
75	621	621	0	621	621	0	619	618	1
80	621	621	0	621	621	0	619	618	1
85	621	621	0	621	621	0	619	618	1
90	621	621	0	621	621	0	619	618	1
95	621	621	0	621	621	0	619	618	1
98	621	621	0	621	621	0	-	618	-
98	678	664	14	647	640	7	-	618	-
100	680	666	14	649	640	9	620	618	2
105	685	670	15	653	642	11	621	618	3
110	690	674	16	659	645	14	624	618	6
115	697	679	18	665	649	16	628	619	9
120	703	684	19	673	654	19	633	621	12
125	711	690	21	681	660	21	639	624	15
130	719	696	23	690	666	24	646	629	17
135	729	703	26	700	674	26	654	634	20
140	739	711	28	711	683	28	663	640	23
145	750	719	31	724	692	32	673	647	26
150	763	729	34	738	703	35	685	656	29
155	776	739	37	753	715	38	697	665	32
160	792	751	41	770	728	42	711	676	35
165	809	764	45	789	743	46	727	688	39
170	827	778	49	809	759	50	744	701	43
175	848	794	54	832	777	55	763	716	47
180	871	811	60	858	797	61	784	732	52
185	896	830	66	886	819	67	808	751	57
190	924	851	73	916	843	73	834	771	63
195	954	875	79	951	870	81	863	793	70
200	988	901	87	988	900	88	894	818	76
205	1026	929	97	1026	929	97	929	846	83
210	1067	961	106	1067	961	106	968	876	92
215	1113	995	118	1113	995	118	1011	910	101
220	1164	1034	130	1164	1034	130	1058	947	111
225	1220	1077	143	1220	1077	143	1110	988	122
230	1277	1124	153	1276	1124	152	1168	1033	135
235	1340	1176	164	1334	1176	158	1232	1083	149
240	1409	1233	176	1398	1233	165	1302	1138	164
245	1486	1291	195	1469	1289	180	1379	1199	180
250	1571	1356	215	1547	1349	198	1465	1266	199
255	1665	1427	238	1634	1415	219	1560	1340	220
260	1769	1506	263	1729	1487	242	1664	1422	242
265	1883	1593	290	1835	1567	268	1779	Note (d)	-
270	2010	1690	320	1951	1656	295	1901	1613	288
275	2150	1796	354	2080	1754	326	2020	1719	301
280	2304	1913	391	2222	1862	360	2150	1818	332
285	2474	2043	431	2378	1981	397	2295	1928	367
290	-	2187	-	-	2113	-	2454	2049	405
295	-	2345	-	-	2258	-	-	2183	-
300	-	-	-	-	-	-	-	2330	-

Notes contained on following page. Dashes in the table indicate that a value is not applicable.

Notes:

(a) Data points for the newly developed P-T limit curves are generated in Section 6 of this report.

(b) Data points for the current P-T limit curves were generated in WCAP-15112 (Reference 17), which forms the basis of the current Technical Specifications (TS) P-T limit curves without uncertainties or adjustments.

(c) Margin equals New P-T limit curve data point minus WCAP-15112 P-T limit curve data point for each temperature and rate.

(d) WCAP-15112 (Reference 17) does not provide a 60°F/hr heatup data point at 265°F.

Table 7-3 North Anna Units 1 and 2 Cooldown P-T Limit Curve Pressure Margin Summary Between the Current P-T Limit Curves and the New P-T Limit Curves to 72 EFPY

Temp. (°F)	Steady-State			-20°F/hr.			-40°F/hr.			-60°F/hr.			-100°F/hr.		
	New[a] (psig)	TS[b] (psig)	Margin[c] (psid)	New[a] (psig)	TS[b] (psig)	Margin[c] (psid)	New[a] (psig)	TS[b] (psig)	Margin[c] (psid)	New[a] (psig)	TS[b] (psig)	Margin[c] (psid)	New[a] (psig)	TS[b] (psig)	Margin[c] (psid)
60	621	621	0	614	606	8	575	566	9	535	525	10	452	441	11
65	621	621	0	617	608	9	577	568	10	537	527	10	455	442	12
70	621	621	0	619	610	9	580	570	10	540	529	11	458	444	13
75	621	621	0	621	612	9	583	572	11	543	531	12	461	447	14
80	621	621	0	621	614	7	586	574	12	546	533	13	464	449	15
85	621	621	0	621	617	4	590	577	13	550	536	14	469	452	16
90	621	621	0	621	620	1	594	580	14	554	539	15	473	455	18
95	621	621	0	621	621	0	598	583	15	559	542	16	478	459	19
98	621	621	0	621	621	0	-	-	-	-	-	-	-	-	-
98	678	664	13	640	625	15	-	-	-	-	-	-	-	-	-
100	680	666	14	642	627	15	603	587	16	564	546	18	484	464	21
105	685	670	15	647	631	16	609	591	18	570	551	19	491	468	23
110	690	674	16	653	635	18	615	596	19	577	556	21	499	474	25
115	697	679	18	659	640	19	622	601	21	584	561	23	507	480	27
120	703	684	20	667	645	21	630	607	23	592	567	25	516	487	29
125	711	690	21	675	652	23	638	613	25	601	574	27	527	495	32
130	719	696	23	684	658	25	648	620	27	611	582	30	539	503	35
135	729	703	26	693	666	28	658	628	30	623	590	33	552	513	38
140	739	711	28	704	674	30	670	637	33	635	599	36	566	524	42
145	750	719	31	716	683	33	683	647	36	649	610	39	582	536	46
150	763	729	34	730	693	37	697	657	40	664	622	43	601	550	51
155	776	739	37	745	704	40	713	669	43	682	634	47	621	565	56
160	792	751	41	761	717	44	731	683	48	701	649	52	643	582	61
165	809	764	45	779	731	48	750	698	53	722	665	57	668	600	67
170	827	778	49	799	746	53	772	714	58	745	682	63	695	621	74
175	848	794	54	821	763	59	796	732	64	771	702	69	726	644	82
180	871	811	60	846	781	65	822	752	70	800	724	76	760	670	90
185	896	830	66	873	802	71	852	775	77	832	748	84	798	699	99
190	924	851	72	903	825	78	884	799	85	867	775	92	840	730	109
195	954	875	80	936	850	86	920	827	94	906	804	102	886	766	120
200	988	901	88	973	878	95	960	857	103	949	837	112	938	805	133
205	1026	929	97	1014	909	105	1004	890	114	997	874	124	995	848	147
210	1067	961	107	1059	943	116	1053	927	126	1051	914	136	1051	896	154

Table 7-3 North Anna Units 1 and 2 Cooldown P-T Limit Curve Pressure Margin Summary Between the Current P-T Limit Curves and the New P-T Limit Curves to 72 EFPY

Temp. (°F)	Steady-State			-20°F/hr.			-40°F/hr.			-60°F/hr.			-100°F/hr.		
	New[(a)] (psig)	TS[(b)] (psig)	Margin[(c)] (psid)	New[(a)] (psig)	TS[(b)] (psig)	Margin[(c)] (psid)	New[(a)] (psig)	TS[(b)] (psig)	Margin[(c)] (psid)	New[(a)] (psig)	TS[(b)] (psig)	Margin[(c)] (psid)	New[(a)] (psig)	TS[(b)] (psig)	Margin[(c)] (psid)
215	1113	995	118	1108	981	128	1107	968	139	1107	959	148	1107	949	158
220	1164	1034	130	1163	1023	141	1163	1014	149	1163	1008	155	1163	1009	155
225	1220	1077	143	1220	1069	151	1220	1064	155	1220	1063	157	1220	1074	146
230	1282	1124	158	1282	1120	162	1282	1120	162	1282	1124	158	1282	1124	158
235	1350	1176	174	1350	1176	174	1350	1176	174	1350	1176	174	1350	1176	174
240	1426	1233	192	1426	1233	192	1426	1233	192	1426	1233	192	1426	1233	192
245	1509	1297	212	1509	1297	212	1509	1297	212	1509	1297	212	1509	1297	212
250	1601	1367	234	1601	1367	234	1601	1367	234	1601	1367	234	1601	1367	234
255	1703	1444	259	1703	1444	259	1703	1444	259	1703	1444	259	1703	1444	259
260	1816	1530	286	1816	1530	286	1816	1530	286	1816	1530	286	1816	1530	286
265	1941	1625	316	1941	1625	316	1941	1625	316	1941	1625	316	1941	1625	316
270	2078	1730	349	2078	1730	349	2078	1730	349	2078	1730	349	2078	1730	349
275	2231	1846	385	2231	1846	385	2231	1846	385	2231	1846	385	2231	1846	385
280	2399	1973	425	2399	1973	425	2399	1973	425	2399	1973	425	2399	1973	425
285	-	2115	-	-	2115	-	-	2115	-	-	2115	-	-	2115	-
290	-	2271	-	-	2271	-	-	2271	-	-	2271	-	-	2271	-
295	-	2444	-	-	2444	-	-	2444	-	-	2444	-	-	2444	-

Notes:

(a) Data points for the newly developed P-T limit curves are generated in Section 6 of this report. Dashes in the table indicate that a value is not applicable.

(b) Data points for the current P-T limit curves were generated in WCAP-15112 (Reference 17), which forms the basis of the current Technical Specifications (TS) P-T limit curves without uncertainties or adjustments.

(c) Margin equals New P-T limit curve data point minus WCAP-15112 P-T limit curve data point for each temperature and rate. Dashes in the table indicate that a value is not applicable.

Table 7-4 North Anna Units 1 and 2 Heatup Margin Summary Between the Current P-T Limit Curves and the New P-T Limit Curves to 72 EFPY

Temperature (°F)	20°F/hr. Pressure Margin (psig)	40°F/hr. Pressure Margin (psig)	60°F/hr. Pressure Margin (psig)
60	0	0	1
65	0	0	1
70	0	0	1
75	0	0	1
80	0	0	1
85	0	0	1
90	0	0	1
95	0	0	1
98	14	7	-
100	14	9	2
105	15	11	3
110	16	14	6
115	18	16	9
120	19	19	12
125	21	21	15
130	23	24	17
135	26	26	20
140	28	28	23
145	31	32	26
150	34	35	29
155	37	38	32
160	41	42	35
165	45	46	39
170	49	50	43
175	54	55	47
180	60	61	52
185	66	67	57
190	73	73	63
195	79	81	70
200	87	88	76
205	97	97	83
210	106	106	92
215	118	118	101
220	130	130	111
225	143	143	122
230	153	152	135
235	164	158	149
240	176	165	164
245	195	180	180
250	215	198	199
255	238	219	220
260	263	242	242
265	290	268	-
270	320	295	288
275	354	326	301
280	391	360	332
285	431	397	367
290	-	-	405

Table 7-5 North Anna Units 1 and 2 Cooldown Margin Summary Between the Current P-T Limit Curves and the New P-T Limit Curves to 72 EFPY

Temperature (°F)	Steady State Pressure Margin (psig)	-20°F/hr. Pressure Margin (psig)	-40°F/hr. Pressure Margin (psig)	-60°F/hr. Pressure Margin (psig)	-100°F/hr. Pressure Margin (psig)
60	0	8	9	10	11
65	0	9	10	10	12
70	0	9	10	11	13
75	0	9	11	12	14
80	0	7	12	13	15
85	0	4	13	14	16
90	0	1	14	15	18
95	0	0	15	16	19
98	13	15	-	-	-
100	14	15	16	18	21
105	15	16	18	19	23
110	16	18	19	21	25
115	18	19	21	23	27
120	20	21	23	25	29
125	21	23	25	27	32
130	23	25	27	30	35
135	26	28	30	33	38
140	28	30	33	36	42
145	31	33	36	39	46
150	34	37	40	43	51
155	37	40	43	47	56
160	41	44	48	52	61
165	45	48	53	57	67
170	49	53	58	63	74
175	54	59	64	69	82
180	60	65	70	76	90
185	66	71	77	84	99
190	72	78	85	92	109
195	80	86	94	102	120
200	88	95	103	112	133
205	97	105	114	124	147
210	107	116	126	136	154
215	118	128	139	148	158
220	130	141	149	155	155
225	143	151	155	157	146
230	158	162	162	158	158
235	174	174	174	174	174
240	192	192	192	192	192
245	212	212	212	212	212
250	234	234	234	234	234
255	259	259	259	259	259
260	286	286	286	286	286
265	316	316	316	316	316
270	349	349	349	349	349
275	385	385	385	385	385
280	425	425	425	425	425
285	470	-	-	-	-

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

Tables A-1 and A-2 contain the thermal stress intensity factors (K_{It}) for the maximum heatup and cooldown rates at 72 EFPY for North Anna Units 1 and 2 based on the Section 6 P-T limit curves. The reactor vessel cylindrical shell radii to the 1/4T and 3/4T locations are as follows:

- 1/4T Radius = 80.575 inches
- 3/4T Radius = 84.414 inches

Table A-1 K_{It} Values for North Anna Units 1 and 2 60°F/hr Heatup Curve at 72 EFPY (w/o Margins for Instrument Errors)

Water Temp. (°F)	Vessel Temperature at 1/4T Location for 60°F/hr Heatup (°F)	1/4T Thermal Stress Intensity Factor (ksi √in.)	Vessel Temperature at 3/4T Location for 60°F/hr Heatup (°F)	3/4T Thermal Stress Intensity Factor (ksi √in.)
60	56.623	-1.062	55.200	0.608
65	60.067	-2.336	56.104	1.594
70	63.664	-3.196	57.938	2.314
75	67.622	-3.941	60.474	2.888
80	71.803	-4.472	63.543	3.327
85	76.142	-4.922	67.044	3.676
90	80.649	-5.251	70.866	3.945
95	85.231	-5.532	74.941	4.160
100	89.927	-5.739	79.204	4.330
105	94.656	-5.921	83.615	4.469
110	99.462	-6.056	88.140	4.581
115	104.281	-6.178	92.753	4.673
120	109.150	-6.270	97.434	4.749
125	114.024	-6.356	102.168	4.813
130	118.931	-6.422	106.944	4.867
135	123.838	-6.486	111.751	4.915
140	128.767	-6.536	116.583	4.956
145	133.695	-6.587	121.434	4.993
150	138.638	-6.628	126.301	5.027
155	143.579	-6.670	131.179	5.058
160	148.529	-6.706	136.066	5.087
165	153.477	-6.744	140.960	5.114
170	158.432	-6.776	145.859	5.140
175	163.386	-6.811	150.763	5.166
180	168.343	-6.841	155.670	5.190
185	173.299	-6.874	160.580	5.214
190	178.258	-6.903	165.492	5.237
195	183.216	-6.935	170.405	5.260
200	188.176	-6.964	175.319	5.283
205	193.136	-6.995	180.235	5.306
210	198.095	-7.023	185.150	5.328

Table A-2 K_{It} Values for North Anna Units 1 and 2 -100°F/hr Cooldown Curve at 72 EFPY (w/o Margins for Instrument Errors)

Water Temp. (°F)	Vessel Temperature at 1/4T Location for -100°F/hr Cooldown (°F)	-100°F/hr Cooldown 1/4T Thermal Stress Intensity Factor (ksi √in.)
210	231.258	12.630
205	226.188	12.577
200	221.117	12.525
195	216.047	12.473
190	210.977	12.420
185	205.907	12.368
180	200.837	12.316
175	195.766	12.263
170	190.697	12.212
165	185.627	12.159
160	180.557	12.108
155	175.487	12.056
150	170.418	12.004
145	165.348	11.952
140	160.279	11.901
135	155.209	11.849
130	150.140	11.798
125	145.071	11.747
120	140.002	11.696
115	134.933	11.645
110	129.865	11.594
105	124.796	11.543
100	119.728	11.493
95	114.659	11.442
90	109.591	11.392
85	104.523	11.342
80	99.455	11.292
75	94.387	11.241
70	89.319	11.192
65	84.252	11.142
60	79.186	11.091

APPENDIX B REACTOR VESSEL INLET AND OUTLET NOZZLES COOLDOWN PRESSURE-TEMPERATURE LIMITS

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

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

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

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

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

$$\sigma = A_0 + A_1 x + A_2 x^2 + A_3 x^3$$

where,

σ = through-wall stress distribution

x = through-wall distance from inside surface

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

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

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

where,

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

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

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

Figures B-1 and B-2 show the most limiting 72 EFPY inlet and outlet nozzle P-T limit curves for North Anna Units 1 and 2 based on limiting ART values of 34.8°F and 8.0°F for the inlet and outlet nozzles, respectively, as determined from Tables 5-6 and 5-9 using surface fluence in the main body of this report. The nozzle P-T limits are provided for a cooldown rate of 100°F/hr, along with a steady-state curve. Also shown in Figures B-1 and B-2 are the North Anna Units 1 and 2 beltline EOLE cooldown P-T limit curves from WCAP-15112 (Reference B-4). These beltline cooldown P-T limit curves are the basis for the cooldown P-T limit curves currently implemented in the North Anna Power Station Technical Specifications and were shown to be valid through SLR, i.e. 72 EFPY, in Section 7 of this report.

Conclusion

Based on the results shown in Figures B-1 and B-2, it is concluded that the nozzle P-T limits are bounded by the traditional cylindrical beltline curves. Therefore, the P-T limits provided in WCAP-15112 remain limiting for the beltline and non-beltline reactor vessel components.

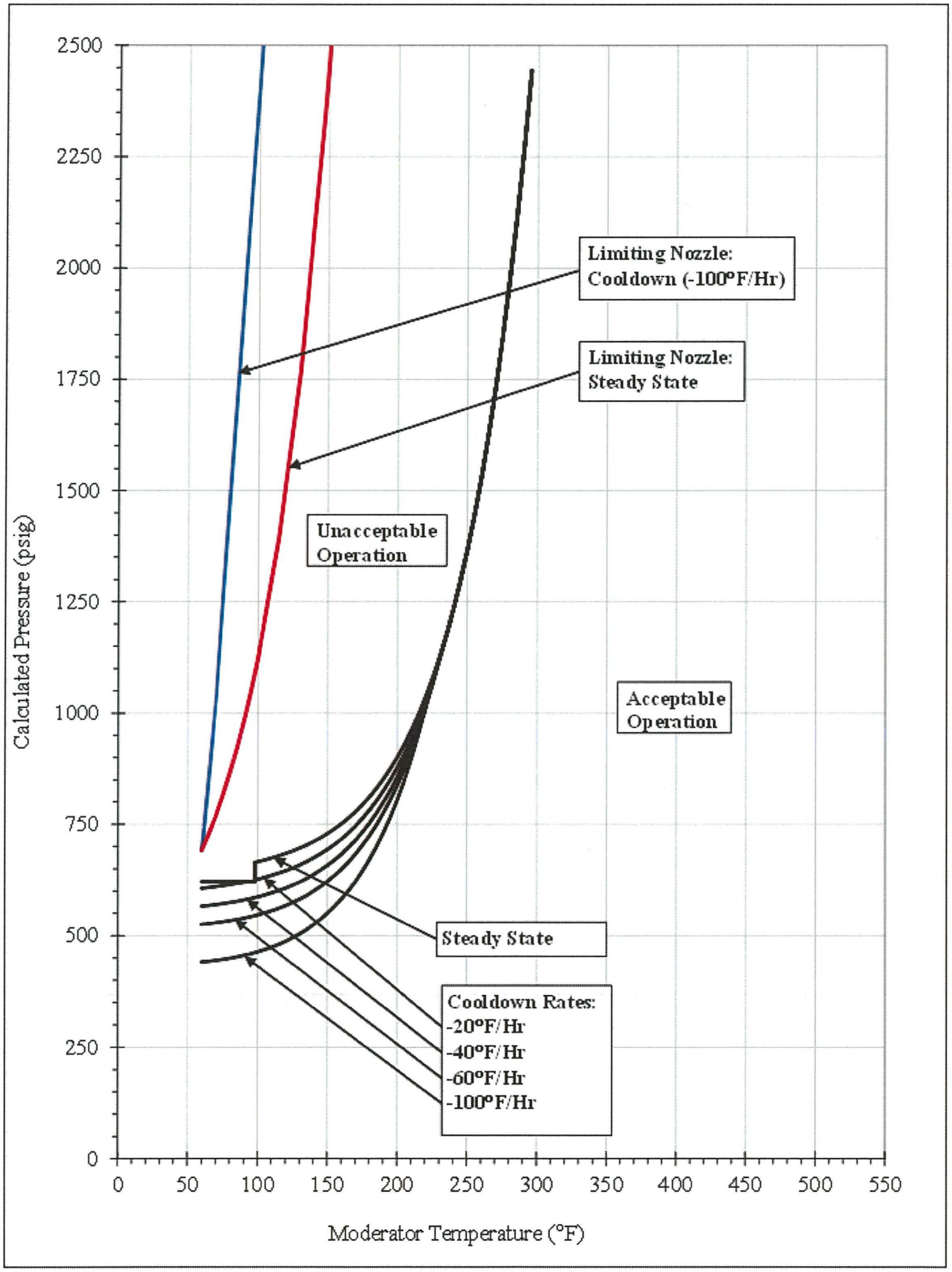

Figure B-1 Comparison of North Anna (72 EFPY for Units 1 and 2) Inlet Nozzle Cooldown P-T Limits (K_{Ic}) with the Beltline Cylindrical Shell P-T limits (K_{Ic}) Without Margins for Instrument Error or Pressure Correction and With Flange Requirements ($ART_{inlet\ nozzle}$ = 34.8°F)

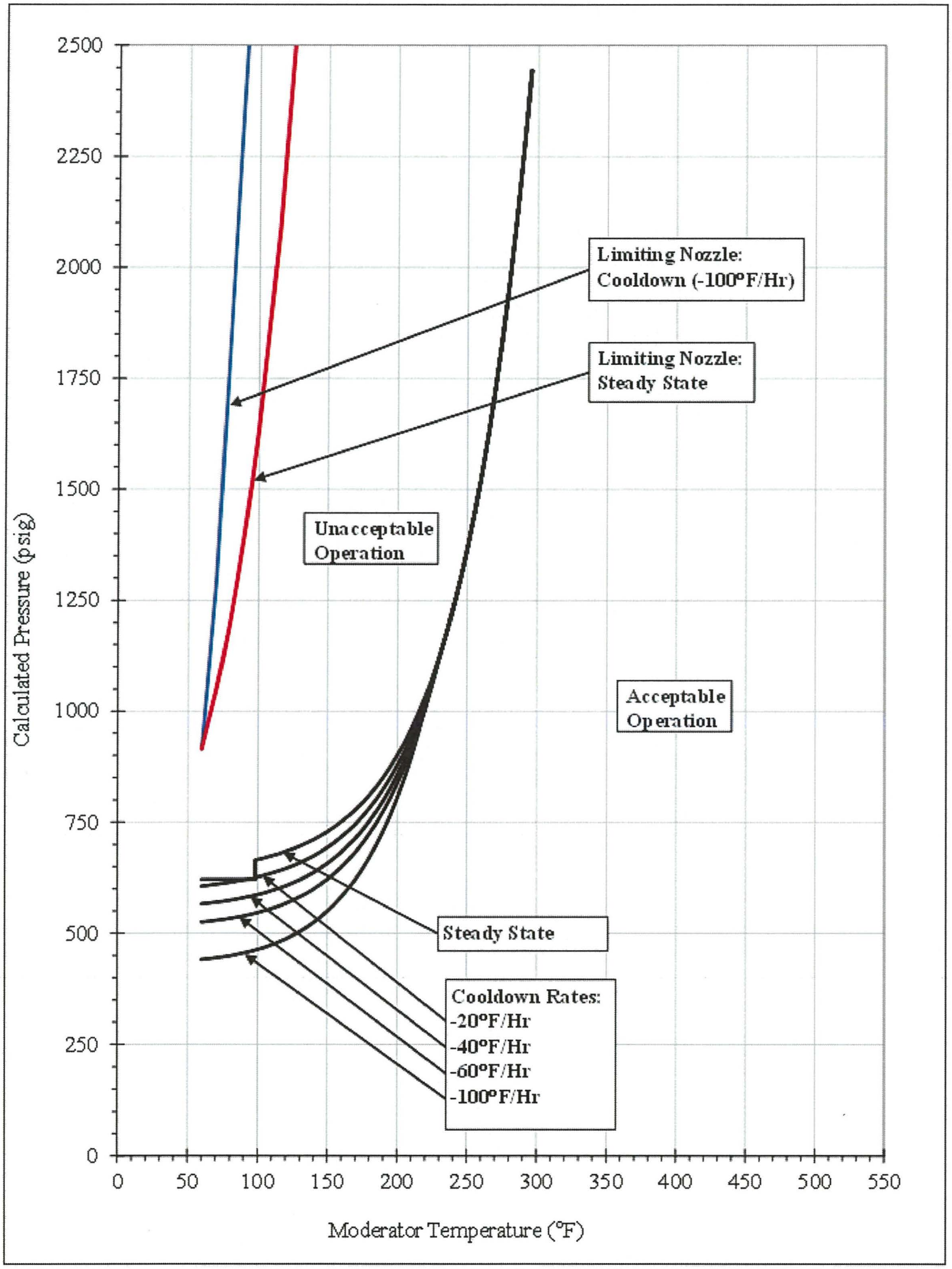

Figure B-2 Comparison of North Anna (72 EFPY for Units 1 and 2) Outlet Nozzle Cooldown P-T Limits (K_{Ic}) with Beltline Cylindrical Shell P-T limits (K_{Ic}) Without Margins for Instrument Error or Pressure Correction and With Flange Requirements ($ART_{outlet\ nozzle}$ = 8.0°F)

B.1 REFERENCES

B-1. Westinghouse Report WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," May 2004.

B-2. Oak Ridge National Laboratory Report, ORNL/TM-2010/246, Revision 1, "Stress and Fracture Mechanics Analyses of Boiling Water Reactor and Pressurized Water Reactor Pressure Vessel Nozzles –Revision 1," June 2012. *[ADAMS Accession Number ML110060164]*

B-3. ASME PVP2011-57015, "Additional Improvements to Appendix G of ASME Section XI Code for Nozzles," G. Stevens, H. Mehta, T. Griesbach, D. Sommerville, July 2011.

B-4. Westinghouse Report WCAP-15112, Revision 2, "North Anna Units 1 and 2 WOG Reactor Vessel 60-Year Evaluation Minigroup Heatup and Cooldown Limit Curves for Normal Operation," March 2001.

B-5. U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988. *[ADAMS Accession Number ML003740284]*

APPENDIX C OTHER RCPB FERRITIC COMPONENTS

10 CFR Part 50, Appendix G (Reference C-1), requires that all Reactor Coolant Pressure Boundary (RCPB) components meet the requirements of Section III of the ASME Code. The lowest service temperature requirement (LST) for all RCPB components, which is specified in NB-2332(b) and NB-3211 of Section III of the ASME Code, is the relevant requirement that would affect the pressure-temperature (P-T) limits. This requirement is applicable to ferritic materials outside of the reactor vessel with a nominal wall thickness greater than 2 ½ inches, such as piping, pumps and valves (Reference C-2).

The North Anna Units 1 and 2 reactor coolant systems do not contain ferritic materials in the Class 1 piping, pumps and valves per Section 4.3 of this report. Therefore, the LST requirements of NB-2332(b) and NB-3211 are not applicable to the North Anna Units 1 and 2 P-T limits.

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

The replacement reactor vessel closure head materials do not affect the flange requirements considered in the development of the North Anna Power Station Technical specifications P-T limits. Additionally, the replacement reactor vessel closure heads for Units 1 and 2 were constructed to the French Construction Code (RCC-M) 1993 Edition with 1st Addenda June 1994, 2nd Addenda June 1995, 3rd Addenda June 1996 and Modification Sheets FM 797, 798, 801, 802, 803, 804, 805, 806, and 807. The sizing calculations and stress and fatigue analyses were performed to Section III of the ASME Code, 1995 Edition through 1996 Addenda. The Design Report and Report of Reconciliation certify that the closure head meets the design requirements for the ASME Code Section III 1968 Edition through Winter 1968 Addenda (Reference C-3).

The steam generators were designed and analyzed to the 1968 Edition through Winter 1968 Addenda of Section III of the ASME Code, and met all applicable requirements at the time of construction. Portions of the steam generators were repaired, and these portions were fabricated and manufactured in accordance with the 1986 Edition of Section III of the ASME Code (Reference C-3). Therefore, no further consideration is necessary for these components with regards to P-T limits.

The pressurizers were designed and analyzed to the 1968 Edition through Winter 1968 Addenda of Section III of the ASME Code, and met all applicable requirements at the time of construction (Reference C-3). No further consideration is necessary for these components with regards to P-T limits.

C.1 REFERENCES

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

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

C-3. North Anna Power Station (NAPS) Updated Final Safety Analysis Report (UFSAR), Amendment No. 54, September 2018.

APPENDIX D LTOP SYSTEM ENABLE TEMPERATURE

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

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

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

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

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

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

Where,

F = 1.1, accumulation factor for safety relief valves

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

p = vessel design pressure, ksig

R_i = vessel inner radius, in.

t = vessel wall thickness, in.

Notes:

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

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

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

RT_{NDT} = 205°F for 72 EFPY (at 1/4T per Table 5-1)

F = 1.1

M_m = 2.566 $\sqrt{in.}$ (See Section 4 for equations used to calculate M_m)

p = 2.485 ksig

R_i = 78.656 in.

t = 7.677 in.

The LTOP system shall be effective below the higher temperature determined in accordance with (1) and (2) in the preceding list, which results in a T_e = 245°F for 72 EFPY. Alternatively, LTOP systems shall be effective below the higher temperature determined in accordance with (1) and (3) above, which results in a T_e = 236.2°F for 72 EFPY. Since Item (3) is less than Item (2), the minimum enable temperature will be based on T_e = 236.2°F.

The enable temperature determined for the fastest heatup rate will result in the highest enable temperature and will bound the enable temperature for all other heatup, cooldown, and isothermal conditions. During a 60°F/hr heatup, the 1/4T metal temperature will reach 236.2°F when the coolant temperature is equal to 249°F. Since this temperature is also greater than 200°F [Item (1) above], the minimum required enable temperature (without margins for instrument uncertainty) is a coolant temperature equal to **249°F** for 72 EFPY.

D.1 REFERENCES

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

D-2. U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988. *[ADAMS Accession Number ML003740284]*

APPENDIX E CREDIBILITY EVALUATION OF THE NORTH ANNA UNITS 1 AND 2 SURVEILLANCE PROGRAMS

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

To date there have been three surveillance capsules removed and tested from each of the North Anna Units 1 and 2 reactor vessels. The Unit 1 forging and weld surveillance data are judged to be non-credible based on the five criteria in Regulatory Guide 1.99, Revision 2. The Unit 2 weld surveillance data are judged to be credible based on the five criteria in Regulatory Guide 1.99, Revision 2; however, the Unit 2 forging surveillance data are judged to be non-credible. Appendix I contains an explanation of the North Anna licensing basis for the use of credible / non-credible surveillance data.

Table E-1 reviews the five criteria in Regulatory Guide 1.99, Revision 2. The following subsections evaluate each of these five criteria for North Anna Units 1 and 2 in order to determine the credibility of the surveillance data for use in neutron radiation embrittlement calculations.

It should be noted that this report also uses surveillance data from Sequoyah Units 1 and 2 surveillance programs. The credibility conclusions for the surveillance data from these programs are contained in WCAP-17539-NP (Reference E-2), Appendix A. The conclusions in WCAP-17539-NP will not be readdressed here as the use of surveillance data in this report does not affect the credibility conclusions.

Table E-1
Regulatory Guide 1.99, Revision 2, Credibility Criteria

Criterion No.	Description
1	Materials in the capsules should be those judged most likely to be controlling with regard to radiation embrittlement.
2	Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions should be small enough to permit the determination of the 30 ft-lbs temperature and upper-shelf energy unambiguously.
3	When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Position 2.1 normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail this criterion for use in shift calculations, they may be credible for determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E185-82.
4	The irradiation temperature of the Charpy specimens in the capsule should match the vessel wall temperature at the cladding/base metal interface within +/- 25°F.
5	The surveillance data for the correlation monitor material in the capsule should fall within the scatter band of the database for that material.

E.1 NORTH ANNA UNIT 1 CREDIBILITY EVALUATION

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

The North Anna Unit 1 reactor vessel consists of the following beltline region materials, which likely would have been considered at the time the surveillance program was designed and licensed:

a) Upper Shell Forging
b) Intermediate Shell Forging
c) Lower Shell Forging
d) Intermediate to Lower Shell Circumferential Weld
e) Upper to Intermediate Shell Circumferential Weld

At the time that the North Anna Unit 1 surveillance program was designed and licensed, the materials selected for use in the North Anna Unit 1 surveillance program (Lower Shell Forging 03 and the Intermediate to Lower Shell Circumferential Weld) were those judged to be most likely controlling with regard to radiation embrittlement according to the accepted methodology. These materials remain limiting with respect to fluence and ART. Thus, the North Anna Unit 1 surveillance program meets the intent of this criterion.

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

The surveillance capsule analysis report, BAW-2356 (Reference E-3), which supports the Position 2.1 chemistry factor calculations, was reviewed and it was determined that this criterion is met.

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

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

Following is the calculation of the best-fit line as described in Regulatory Position 2.1 of Regulatory Guide 1.99, Revision 2. In addition, the recommended NRC methods for determining credibility will be followed. The NRC methods were presented to the industry at a meeting held by the NRC on February 12 and 13, 1998 (Reference E-4). At this meeting the NRC presented five cases. Of the five cases, Case 1 ("Surveillance data available from plant but no other source") most closely represents the situation for the North Anna Unit 1 surveillance forging and weld material.

Case 1: Lower Shell Forging 03 and Weld Heat # 25531

Following the NRC Case 1 guidelines, the North Anna Unit 1 surveillance forging and weld metal (Heat # 25531) will be evaluated using the North Anna Unit 1 data. Only North Anna Unit 1 data is being considered; therefore, no temperature adjustment is required.

The scatter of ΔRT_{NDT} values about the functional form of a best-fit line drawn as described in Regulatory Position 2.1 is presented in Table E.1-1.

Table E.1-1
North Anna Unit 1 Surveillance Capsule Data Scatter about the Best-Fit Line Using All Available Surveillance Data

Material	Capsule	CF[(a)] (Slope$_{best-fit}$) (°F)	Capsule Fluence (x 10^{19} n/cm^2)	FF	Measured ΔRT_{NDT}[(b)] (°F)	Predicted ΔRT_{NDT} (°F)	Scatter ΔRT_{NDT}[(c)] (°F)	<17°F (Base Metal) <28°F (Weld)
Lower Shell Forging 03 (Tangential)	V	81.68	0.306	0.675	51	55.2	4.2	Yes
	U	81.68	0.914	0.975	116	79.6	36.4	No
	W	81.68	2.05	1.196	93	97.7	4.7	Yes
Lower Shell Forging 03 (Axial)	V	81.68	0.306	0.675	29	55.2	26.2	No
	U	81.68	0.914	0.975	72	79.6	7.6	Yes
	W	81.68	2.05	1.196	96	97.7	1.7	Yes
Surveillance Weld Material (Heat # 25531)	V	67.53	0.306	0.675	88	45.6	42.4	No
	U	67.53	0.914	0.975	30	65.8	35.8	No
	W	67.53	2.05	1.196	86	80.7	5.3	Yes

Notes:

(a) Since the Position 2.1 CFs in Table 3-4 did not consider chemistry or temperature adjustments the interim CFs are equal to the Position 2.1 CFs calculated in Table 3-4 of this report.

(b) ΔRT_{NDT} values are the measured 30 ft-lbs shift values taken from BAW-2356 (Reference E-3).

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

For North Anna (see Appendix I), if one or more of the surveillance data fall outside +/- 1σ scatter band of the Position 2.1 CF trend line then the data is considered non-credible. Table E.1-1 indicates that only four of the six surveillance data points fall inside the +/- 1σ of 17°F scatter band for surveillance base metals. Therefore, the forging data is deemed "non-credible" per the third criterion.

Table E.1-1 indicates that two of the three surveillance data points fall outside the +/- 1σ of 28°F scatter band for surveillance weld materials. Therefore, the surveillance weld data is deemed "non-credible" per the third criterion.

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

The capsule specimens are located in the reactor between the thermal shield and the vessel wall and are positioned opposite the center of the core. The test capsules are in guide tubes attached to the thermal shield. The location of the specimens with respect to the reactor vessel beltline provides assurance that the reactor vessel wall and the specimens experience equivalent operating conditions such that the temperatures will not differ by more than 25°F. Hence, this criterion is met.

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

The North Anna Unit 1 surveillance program does not contain correlation monitor material. Hence, this criterion is not applicable to the North Anna Unit 1 surveillance program.

CONCLUSION:

Based on the preceding responses to the five criteria of Regulatory Guide 1.99, Revision 2, Section B, the Lower Shell Forging 03 and weld Heat # 25531 surveillance data are deemed non-credible.

E.2 NORTH ANNA UNIT 2 CREDIBILITY EVALUATION

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

The North Anna Unit 2 reactor vessel consists of the following beltline region materials, which likely would have been considered at the time the surveillance program was designed and licensed:

a) Upper Shell Forging
b) Intermediate Shell Forging
c) Lower Shell Forging
d) Intermediate to Lower Shell Circumferential Weld
e) Upper to Intermediate Shell Circumferential Weld

At the time that the North Anna Unit 2 surveillance program was designed and licensed, the materials selected for use in the North Anna Unit 2 surveillance program (Intermediate Shell Forging 04 and the Intermediate to Lower Shell Circumferential Weld) were those judged to be most likely controlling with regard to radiation embrittlement according to the accepted methodology. These materials remain limiting with respect to ART. Thus, the North Anna Unit 2 surveillance program meets the intent of this criterion.

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

The surveillance capsule analysis report, BAW-2376 (Reference E-5), which supports the Position 2.1 chemistry factor calculations, was reviewed and it was determined that this criterion is met.

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

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

Following is the calculation of the best-fit line as described in Regulatory Position 2.1 of Regulatory Guide 1.99, Revision 2. In addition, the recommended NRC methods for determining credibility will be followed. The NRC methods were presented to the industry at a meeting held by the NRC on February 12 and 13, 1998 (Reference E-4). At this meeting the NRC presented five cases. Of the five cases, Case 1 ("Surveillance data available from plant but no other source") most closely represents the situation for the North Anna Unit 2 surveillance forging and weld material.

Case 1: Intermediate Shell Forging 04 and Weld Heat # 716126

Following the NRC Case 1 guidelines, the North Anna Unit 2 surveillance forging and weld metal (Heat # 716126) will be evaluated using the North Anna Unit 2 data. Only North Anna Unit 2 data is being considered; therefore, no temperature adjustment is required.

The scatter of ΔRT_{NDT} values about the functional form of a best-fit line drawn as described in Regulatory Position 2.1 is presented in Table E.2-1.

Table E.2-1
North Anna Unit 2 Surveillance Capsule Data Scatter about the Best-Fit Line Using All Available Surveillance Data

Material	Capsule	CF[(a)] (Slope$_{best\text{-}fit}$) (°F)	Capsule Fluence (x 10^{19} n/cm^2)	FF	Measured ΔRT_{NDT}[(b)] (°F)	Predicted ΔRT_{NDT} (°F)	Scatter ΔRT_{NDT}[(c)] (°F)	<17°F (Base Metal) <28°F (Weld)
Intermediate Shell Forging 04 (Tangential)	V	53.44	0.286	0.658	19	35.2	16.2	Yes
	U	53.44	0.985	0.996	33	53.2	20.2	No
	W	53.44	2.08	1.199	86	64.1	21.9	No
Intermediate Shell Forging 04 (Axial)	V	53.44	0.286	0.658	21	35.2	14.2	Yes
	U	53.44	0.985	0.996	66	53.2	12.8	Yes
	W	53.44	2.08	1.199	65	64.1	0.9	Yes
Surveillance Weld Material (Heat # 716126)	V	26.61	0.286	0.658	18	17.5	0.5	Yes
	U	26.61	0.985	0.996	8	26.5	18.5	Yes
	W	26.61	2.08	1.199	47	31.9	15.1	Yes

Notes:

(a) Since the Position 2.1 CFs in Table 3-6 did not consider chemistry or temperature adjustments the interim CFs are equal to the Position 2.1 CFs calculated in Table 3-6.

(b) ΔRT_{NDT} values are the measured 30 ft-lbs shift values taken from BAW-2376 (Reference E-5).

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

For North Anna (see Appendix I), if one or more of the surveillance data fall outside +/- 1σ scatter band of the Position 2.1 CF trend line then the data is considered non-credible. Table E.2-1 indicates that only four of the six surveillance data points fall inside the +/- 1σ of 17°F scatter band for surveillance base metals. Therefore, the forging data is deemed "non-credible" per the third criterion.

Table E.2-1 indicates that all three of the three surveillance data points for the surveillance weld materials fall inside the +/- 1σ of 28°F scatter band. Therefore, the surveillance weld data is deemed "credible" per the third criterion.

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

The capsule specimens are located in the reactor between the thermal shield and the vessel wall and are positioned opposite the center of the core. The test capsules are in guide tubes attached to the thermal shield. The location of the specimens with respect to the reactor vessel beltline provides assurance that the reactor vessel wall and the specimens experience equivalent operating conditions such that the temperatures will not differ by more than 25°F. Hence, this criterion is met.

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

The North Anna Unit 2 surveillance program does not contain correlation monitor material. Hence, this criterion is not applicable to the North Anna Unit 2 surveillance program.

CONCLUSION:

Based on the preceding responses to the criteria of Regulatory Guide 1.99, Revision 2, Section B, the Intermediate Shell Forging 04 surveillance data are deemed non-credible; however, the weld Heat # 716126 surveillance data are deemed credible.

E.3 REFERENCES

E-1. U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988. *[ADAMS Accession Number ML003740284]*

E-2. Westinghouse Report WCAP-17539-NP, Revision 0, "Sequoyah Units 1 and 2 Time-Limited Aging Analysis on Reactor Vessel Integrity," March 2012. *[ADAMS Accession Number ML13032A253]*

E-3. BAW-2356, "Analysis of Capsule W Virginia Power North Anna Unit No. 1 Nuclear Power Plant, Reactor Vessel Material Surveillance Program," September 1999.

E-4. K. Wichman, M. Mitchell, and A. Hiser, US NRC, Generic Letter 92-01 and RPV Integrity Workshop Handouts, "NRC/Industry Workshop on RPV Integrity Issues," February 12, 1998. *[ADAMS Accession Number ML110070570]*

E-5. BAW-2376, "Analysis of Capsule W Virginia Power North Anna Unit No. 2 Nuclear Power Plant, Reactor Vessel Material Surveillance Program," August 2000.

APPENDIX F NORTH ANNA UNITS 1 AND 2 UPPER-SHELF ENERGY EVALUATION AT 72 EFPY

F.1 INTRODUCTION

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

The requirements on USE are included in 10 CFR 50, Appendix G (Reference F-1). 10 CFR 50, Appendix G requires utilities to submit an analysis at least 3 years prior to the time that the USE of any RPV material is predicted to drop below 50 ft-lb, as measured by Charpy V-notch specimen testing.

There are two methods that can be used to predict the decrease in USE with irradiation, depending on the availability of credible surveillance capsule data as defined in Regulatory Guide 1.99, Revision 2 (Reference F-2). For vessel beltline materials that are not in the surveillance program or have non-credible data, the Charpy USE (Position 1.2) is assumed to decrease as a function of fluence and copper content, as indicated in Regulatory Guide 1.99, Revision 2. When two or more credible surveillance sets become available from the reactor, they may be used to determine the Charpy USE of the surveillance material. The surveillance data are then used in conjunction with the Regulatory Guide to predict the change in USE (Position 2.2) of the RPV material due to irradiation. Per Regulatory Guide 1.99, Revision 2, when credible data exist, the Position 2.2 projected USE value should be used in preference to the Position 1.2 projected USE value. Note, if data from the surveillance materials is determined to be non-credible for determination of ΔRT_{NDT} by Credibility Criterion 3 of Regulatory Guide 1.99, Revision 2, then "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."

The 72 EFPY Position 1.2 USE values of the vessel materials can be predicted using the corresponding 1/4T fluence projections, the copper content of the materials, and Figure 2 in Regulatory Guide 1.99, Revision 2 (see Figures F-1 and F-3 of this report).

The predicted Position 2.2 USE values are determined for the reactor vessel materials that are contained in the surveillance program by using the reduced plant surveillance data along with the corresponding 1/4T fluence projection. The reduced plant surveillance data was obtained from Table 7-6 of BAW-2356 (Reference F-3) for North Anna Unit 1. The reduced plant surveillance data was obtained from Table 7-6 of BAW-2376 (Reference F-4) for North Anna Unit 2. The surveillance data was plotted in Regulatory Guide 1.99, Revision 2, Figure 2 (see Figures F-2 and F-4 of this report) using the surveillance capsule fluence values documented in Table 2-1 of this report for North Anna Unit 1 and Table 2-2 of this report for North Anna Unit 2. This data was fitted by drawing a line parallel to the existing lines as the upper bound of all the surveillance data. These reduced lines were used instead of the existing lines to determine the Position 2.2 SLR USE values.

The projected USE values were calculated to determine if the North Anna Units 1 and 2 beltline and extended beltline materials remain above the 50 ft-lb criterion at 72 EFPY (SLR). These calculations are summarized in Tables F-1 and F-2. Fluence values corresponding to the lowest extent of the nozzle welds at the surface were used to conservatively calculate the projected USE values for the nozzle forgings.

F.2 CONCLUSION

For North Anna Unit 1, the limiting USE value at 72 EFPY is 50.0 ft-lb (see Table F-1); this value corresponds to Inlet Nozzle Forging 11 using Position 1.2. The Unit 1 Inlet Nozzle 11 USE value set equal to 50 ft-lbs results in a projected drop of 10.7%. A review of Regulatory Guide 1.99, Revision 2, Figure 2 resulted in a conservative estimate of approximately 11%, but the figure has limited precision. A decrease of 10.7% is considered appropriate based on the following conservativism in the calculations. The estimated % decrease is based on a fluence of 2 x 10^{17} n/cm^2 (E > 1.0 MeV), which is the lowest fluence line displayed in Regulatory Guide 1.99, Revision 2, Figure 2. The actual fluence is projected to be roughly half this, i.e. 1.20 x 10^{17} n/cm^2 (E > 1.0 MeV), at the lowest extent of the nozzle weld and would be even lower at higher axial elevations. In addition, the fluence would be further decreased if attenuation to the 1/4T location were considered. These additional decreases in fluence would raise the projected USE of Unit 1 Inlet Nozzle 11 above 50 ft-lbs. As shown in Table F-1, all North Anna Unit 1 reactor vessel materials are projected to remain at or above the USE screening criterion value of 50 ft-lbs at 72 EFPY.

For North Anna Unit 2, the limiting USE value at 72 EFPY is 48.2 ft-lb (see Table F-2); this value corresponds to the Intermediate Shell Forging 04 using Position 2.2. Position 2.2 was used to determine the Unit 2 Intermediate Shell Forging 04 USE value even though its surveillance data was deemed non-credible per Appendix E. Per Regulatory Guide 1.99, Revision 2, this is appropriate since the upper shelf can be clearly determined from the surveillance test results. As shown in Table F-2, all other North Anna Unit 2 reactor vessel materials are projected to remain above the USE screening criterion value of 50 ft-lbs at 72 EFPY.

The North Anna Unit 2 Intermediate Shell Forging 04 reactor vessel material, which is projected to drop below 50 ft-lbs USE at SLR, is addressed in the equivalent margins analysis (EMA) performed under PWROG PA-MSC-1481. to qualify the material at 72 EFPY. The material-specific EMA in PA-MSC-1481 is underway, and must be submitted at least 3 years prior to the USE dropping below 50 ft-lbs. The Unit 2 Intermediate Shell Forging 04 is projected to drop below 50 ft-lbs at 52.3 EFPY (EOLE), which is projected to occur in 2040.

In addition to the material discussed above, PA-MSC-1481 includes EMAs for all of the following materials at each Unit for conservativism.

- Upper Shell Forging
- Intermediate Shell Forging
- Inlet Nozzle Forgings
- Outlet Nozzle Forgings
- Inlet Nozzle Welds
- Outlet Nozzle Welds

Table F-1 Predicted USE Values at 72 EFPY (SLR) for North Anna Unit 1

Reactor Vessel Material	Heat #	Wt. % Cu[(a)]	1/4T SLR Fluence[(b)] (x 10^{19} n/cm^2, E > 1.0 MeV)	Unirradiated USE[(a)] (ft-lbs)	Projected USE Decrease[(c)] (%)	Projected SLR USE (ft-lbs)
Position 1.2						
Upper Shell Forging 05	990286 / 295213	0.16	0.192	72	17.0	59.8
Upper to Intermediate Shell Circumferential Weld (OD 94%)	25295	0.352	0.221	112	34.0	73.9
Upper to Intermediate Shell Circumferential Weld (ID 6%)[(f)]	4278	0.12	0.221	105	18.5	85.6
Intermediate Shell Forging 04	990311 / 298244	0.12	4.46	91	30.0	63.7
Intermediate to Lower Shell Circumferential Weld	25531	0.098	4.44	95	34.5	62.2
Lower Shell Forging 03	990400 / 292332	0.156	4.54	85	36.0	54.4
Inlet Nozzle Forging 09 to Upper Shell Weld	Rotterdam	0.35	0.00898	72	0.0[(e)]	72.0
Inlet Nozzle Forging 10 to Upper Shell Weld			0.0313	72	26.0	53.3
Inlet Nozzle Forging 11 to Upper Shell Weld			0.0120	72	24.0	54.7
Outlet Nozzle Forging 12 to Upper Shell Weld			0.0182	72	24.0	54.7
Outlet Nozzle Forging 13 to Upper Shell Weld			0.00522	72	0.0[(e)]	72.0
Outlet Nozzle Forging 14 to Upper Shell Weld			0.00697	72	0.0[(e)]	72.0
Inlet Nozzle Forging 09	990290-11	0.13	0.00898	71	0.0[(e)]	71.0
Inlet Nozzle Forging 10	990290-12	0.13	0.0313	58	10.0	52.2
Inlet Nozzle Forging 11	990268-21	0.18	0.0120	56	10.7[(g)]	**50.0**[(g)]
Outlet Nozzle Forging 12	990290-31	0.13	0.0182	66	9.0	60.1
Outlet Nozzle Forging 13	990290-22	0.13	0.00522	59	0.0[(e)]	59.0
Outlet Nozzle Forging 14	990290-21	0.13	0.00697	59	0.0[(e)]	59.0
Position 2.2[(d)]						
Intermediate to Lower Shell Circumferential Weld	25531	0.098	4.44	95	27.0	69.4[(h)]
Lower Shell Forging 03	990400 / 292332	0.156	4.54	85	36.0	54.4[(h)]

Notes on the following page.

Notes:

(a) Copper weight percent values and unirradiated USE values were taken from Table 3-1 of this report.

(b) Values taken from Table 5-2. The surface fluence at the lowest extent of the nozzle to upper shell weld centerline was used to represent the inlet and outlet nozzle forgings and associated welds. Fluence values above 1 x 10^{17} n/cm^2 but below 2 x 10^{17} n/cm^2 (E > 1.0 MeV) were rounded to 2 x 10^{17} n/cm^2 (E > 1.0 MeV) when determining the % decrease because 2 x 10^{17} n/cm^2 is the lowest fluence displayed in Figure 2 of the Guide.

(c) The Position 1.2 USE decrease values were calculated by plotting the 1/4T fluence values onto Figure 2 of Regulatory Guide 1.99, Revision 2 and using the material-specific Cu wt. % values. Base metal and weld Cu wt. % lines were extended into the low fluence area of Regulatory Guide 1.99, Revision 2, Figure 2, i.e., below 10^{18} n/cm^2, in order to determine the USE % decrease as needed.

(d) Calculated using surveillance capsule measured percent decrease in USE from BAW-2356 (Reference F-3) and Regulatory Guide 1.99, Revision 2, Position 2.2; see Figure F-2.

(e) Embrittlement effects only need to be considered if the fluence is greater than 10^{17} n/cm^2 (E > 1.0 MeV).

(f) Since this inner diameter (ID) weld is only 6% of the vessel thickness, the weld is not present at the 1/4T location; hence, it is not applicable to this calculation. It is presented for information only.

(g) The Unit 1 Inlet Nozzle 11 USE value is set equal to 50 ft-lbs which results in a projected drop of 10.7%. A review of Regulatory Guide 1.99, Revision 2, Figure 2 resulted in a conservative estimate of approximately 11%, but the figure has limited precision. A decrease of 10.7% is considered appropriate based on the following conservativism in the calculations. The estimated % decrease is based on a fluence of 2 x 10^{17} n/cm^2 (E > 1.0 MeV), which is the lowest fluence line displayed in Regulatory Guide 1.99, Revision 2, Figure 2. The actual fluence is projected to be roughly half this, i.e. 1.20 x 10^{17} n/cm^2 (E > 1.0 MeV), at the lowest extent of the nozzle weld and would be even lower at higher axial elevations. In addition, the fluence would be further decreased if attenuation to the 1/4T location were considered. These additional decreases in fluence would raise the projected USE of Unit 1 Inlet Nozzle 11 above 50 ft-lbs.

(h) Position 2.2 was used to determine the Unit 1 Lower Shell Forging 03 and the Intermediate to Lower Shell Weld USE value even though the surveillance data were deemed non-credible per Appendix E. Per Regulatory Guide 1.99, Revision 2, this is appropriate since the upper shelf can be clearly determined from the surveillance test results.

Table F-2 Predicted USE Values at 72 EFPY (SLR) for North Anna Unit 2

Reactor Vessel Material	Heat #	Wt. % Cu[(a)]	1/4T SLR Fluence[(b)] (x 10^{19} n/cm^2, E > 1.0 MeV)	Unirradiated USE[(a)] (ft-lbs)	Projected USE Decrease[(c)] (%)	Projected SLR USE (ft-lbs)
Position 1.2						
Upper Shell Forging 05	990598 / 291396	0.08	0.194	72	13.0	62.6
Upper to Intermediate Shell Circumferential Weld (OD 94%)	4278	0.12	0.224	105	18.5	85.6
Upper to Intermediate Shell Circumferential Weld (ID 6%)[(f)]	801	0.18	0.224	75	23.0	57.8
Intermediate Shell Forging 04	990496 / 292424	0.107	4.54	72	28.0	51.8
Intermediate to Lower Shell Circumferential Weld	716126	0.066	4.53	109	29.0	77.4
Lower Shell Forging 03	990533 / 297355	0.13	4.63	80	32.0	54.4
Inlet Nozzle Forging 09 to Upper Shell Weld	8816 20459 27622	0.23	0.00826	75	0.0[(e)]	75.0
Inlet Nozzle Forging 10 to Upper Shell Weld			0.0314	75	17.0	62.3
Inlet Nozzle Forging 11 to Upper Shell Weld			0.0118	75	15.0	63.8
Outlet Nozzle Forging 12 to Upper Shell Weld			0.0182	75	15.0	63.8
Outlet Nozzle Forging 13 to Upper Shell Weld			0.00479	75	0.0[(e)]	75.0
Outlet Nozzle Forging 14 to Upper Shell Weld			0.00687	75	0.0[(e)]	75.0
Inlet Nozzle Forging 09	990426	0.19	0.00826	56	0.0[(e)]	56.0
Inlet Nozzle Forging 10	54567-2	0.14	0.0314	77	10.5	68.9
Inlet Nozzle Forging 11	54590-2	0.155	0.0118	75	10.0	67.5
Outlet Nozzle Forging 12	990426-22	0.19	0.0182	60	11.5	53.1
Outlet Nozzle Forging 13	990426-31	0.19	0.00479	56	0.0[(e)]	56.0
Outlet Nozzle Forging 14	791291	0.12	0.00687	74	0.0[(e)]	74.0
Position 2.2[(d)]						
Intermediate Shell Forging 04	990496 / 292424	0.107	4.54	72	33.0	**48.2**[(g)]
Intermediate to Lower Shell Circumferential Weld	716126	0.066	4.53	109	28.0	78.5

Notes on the following page.

Notes:

(a) Copper weight percent values and unirradiated USE values were taken from Table 3-2 of this report.

(b) Values taken from Table 5-3. The surface fluence at the lowest extent of the nozzle to upper shell weld centerline was used to represent the inlet and outlet nozzle forgings and associated welds. Fluence values above 1 x 10^{17} n/cm^2 but below 2 x 10^{17} n/cm^2 (E > 1.0 MeV) were rounded to 2 x 10^{17} n/cm^2 (E > 1.0 MeV) when determining the % decrease because 2 x 10^{17} n/cm^2 is the lowest fluence displayed in Figure 2 of the Guide.

(c) The Position 1.2 USE decrease values were calculated by plotting the 1/4T fluence values onto Figure 2 of Regulatory Guide 1.99, Revision 2 and using the material-specific Cu wt. % values. Base metal and weld Cu wt. % lines were extended into the low fluence area of Regulatory Guide 1.99, Revision 2, Figure 2, i.e., below 10^{18} n/cm^2, in order to determine the USE % decrease as needed.

(d) Calculated using surveillance capsule measured percent decrease in USE from BAW-2376 (Reference F-4) and Regulatory Guide 1.99, Revision 2, Position 2.2; see Figure F-4.

(e) Embrittlement effects only need to be considered if the fluence is greater than 10^{17} n/cm^2 (E > 1.0 MeV).

(f) Since this inner diameter (ID) weld is only 6% of the vessel thickness, the weld is not present at the 1/4T location; hence, it is not applicable to this calculation. It is presented for information only.

(g) Position 2.2 was used to determine the Unit 2 Intermediate Shell Forging 04 USE value even though its surveillance data was deemed non-credible per Appendix E. Per Regulatory Guide 1.99, Revision 2, this is appropriate since the upper shelf can be clearly determined from the surveillance test results.

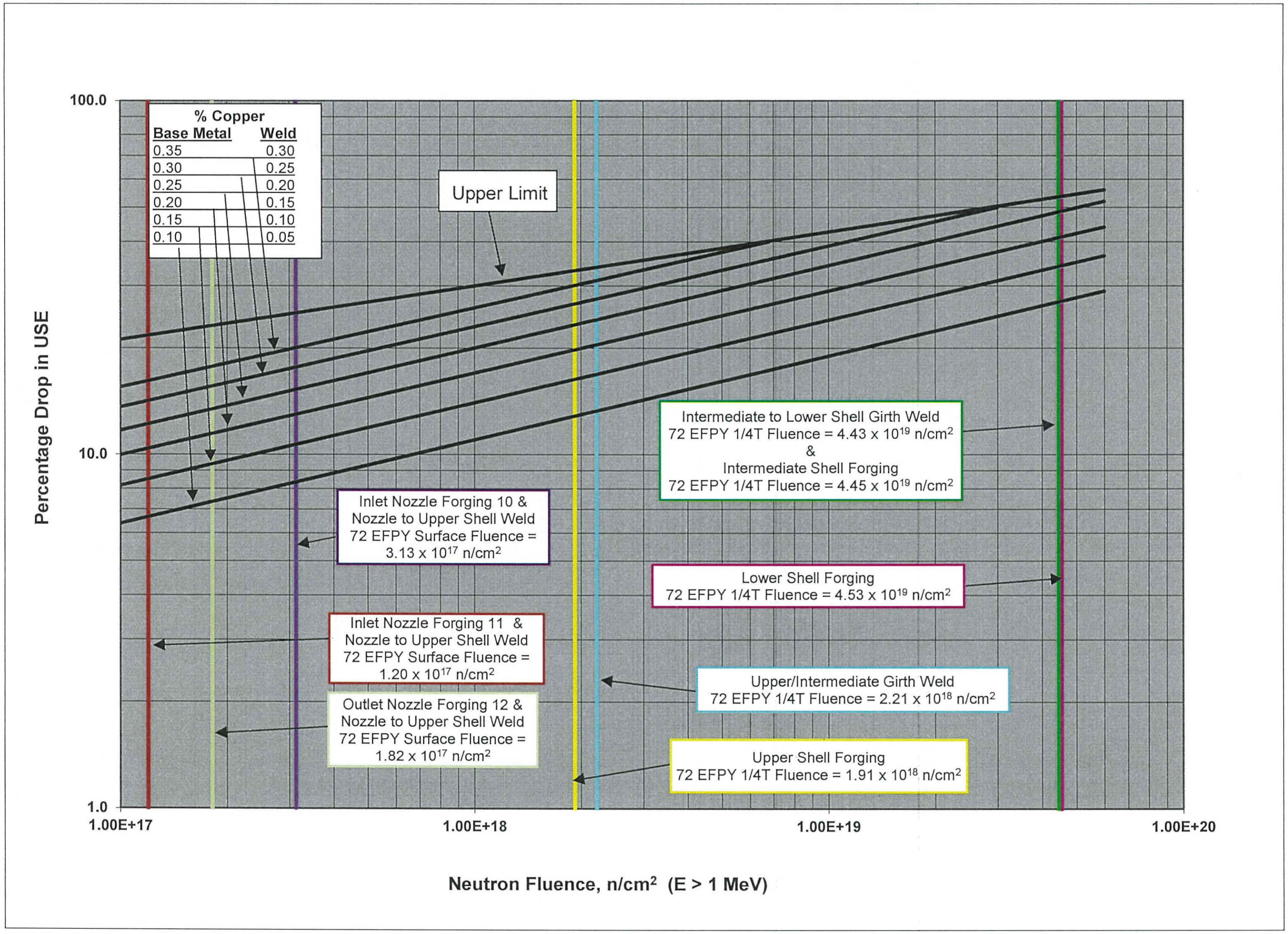

Figure F-1 Regulatory Guide 1.99, Revision 2, Position 1.2 Predicted Decrease in Upper-Shelf Energy as a Function of Copper and Fluence for North Anna Unit 1 at SLR (72 EFPY)

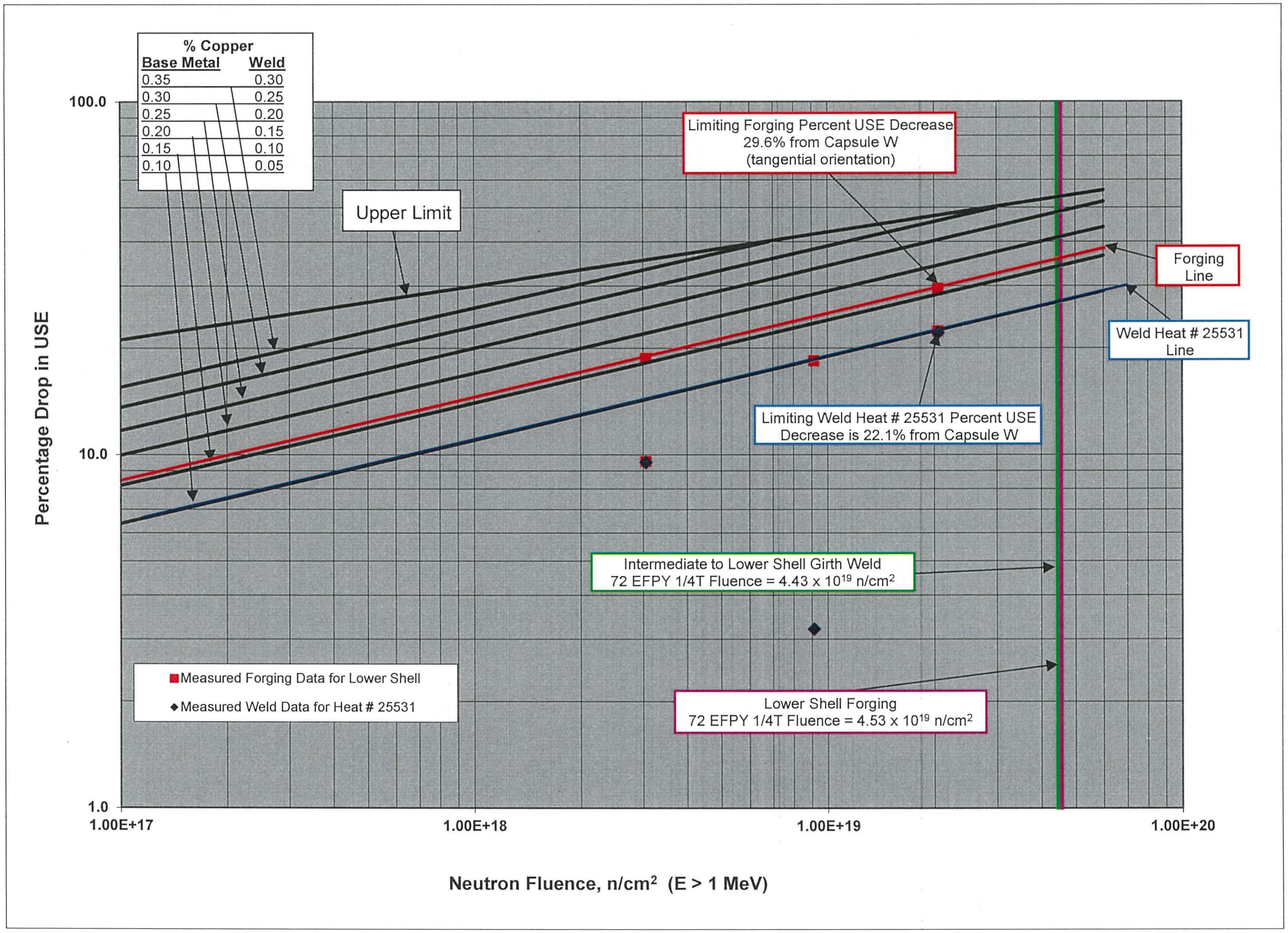

Figure F-2 Regulatory Guide 1.99, Revision 2, Position 2.2 Predicted Decrease in Upper-Shelf Energy as a Function of Copper and Fluence for North Anna Unit 1 at SLR (72 EFPY)

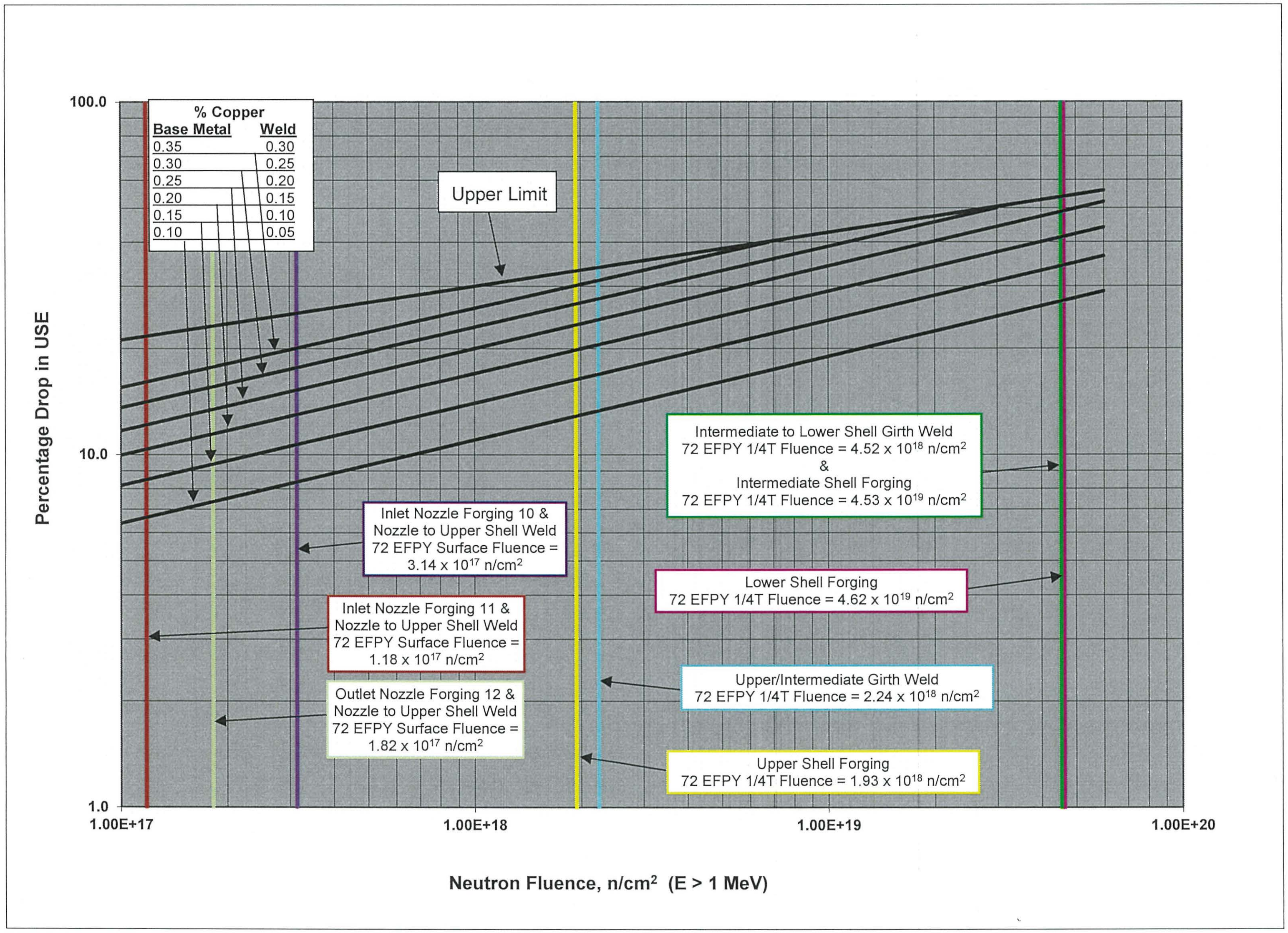

Figure F-3 Regulatory Guide 1.99, Revision 2, Position 1.2 Predicted Decrease in Upper-Shelf Energy as a Function of Copper and Fluence for North Anna Unit 2 at SLR (72 EFPY)

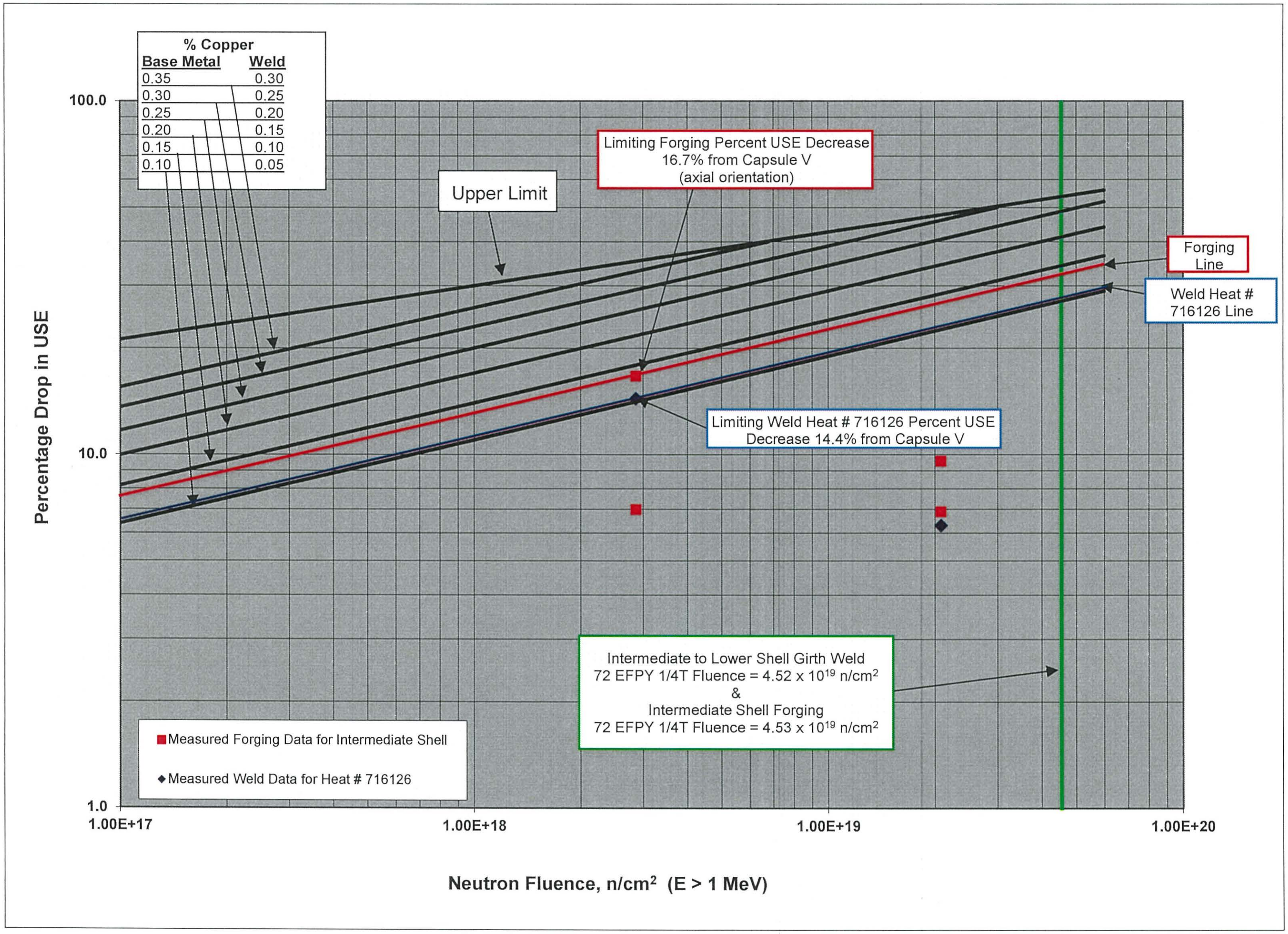

Figure F-4 Regulatory Guide 1.99, Revision 2, Position 2.2 Predicted Decrease in Upper-Shelf Energy as a Function of Copper and Fluence for North Anna Unit 2 at SLR (72 EFPY)

F.3 REFERENCES

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

F-2. U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988. *[ADAMS Accession Number ML003740284]*

F-3. BAW-2356, "Analysis of Capsule W Virginia Power North Anna Unit No. 1 Nuclear Power Plant, Reactor Vessel Material Surveillance Program," September 1999.

F-4. BAW-2376, "Analysis of Capsule W Virginia Power North Anna Unit No. 2 Nuclear Power Plant, Reactor Vessel Material Surveillance Program," August 2000.

APPENDIX G MATERIAL PROPERTY INPUT COMPARISON

This appendix provides tables which compare the material property input values utilized in this report, taken from PWROG-18005-NP (Reference G-1), with those utilized in Dominion calculation SM-1008, Addendum 00M (Reference G-2) and the North Anna Power Station Updated Final Safety Analysis Report (UFSAR) (Reference G-3), as applicable.

As shown in Tables G-1 and G-2, several materials in North Anna Units 1 and 2 had initial RT_{NDT} values that increased as a result of using the material properties defined in PWROG-18005-NP (Reference G-1). However, these increases do not result in changes to the North Anna Power Station EOLE P-T limit curves or in violations of the 10 CFR 50.61 Pressurized Thermal Shock (PTS) limits. The materials with initial RT_{NDT} values that increased are not the limiting materials in the previous evaluations of the P-T limits from WCAP-15112 and PTS analyses of record (AOR), nor are the initial RT_{NDT} increases significant enough to make the associated materials limiting. Therefore, these increases do not adversely affect the P-T limit curves or PTS analyses of record (AOR) for EOLE.

As shown in Tables G-5 and G-6, several materials in North Anna Units 1 and 2 have initial USE values that decreased as a result of using the material properties defined in PWROG-18005-NP (Reference G-1). However, these decreases are not significant enough to cause the USE results in the AOR, SM-1008 (Reference G-2), to violate the USE screening criterion of 10 CFR 50, Appendix G (Reference G-6) for EOLE.

Table G-1 Comparison of Previous and Current Initial RT_{NDT} Values for North Anna Unit 1

Material Identification	Previous Initial RT_{NDT}[a] (°F)	Current Initial RT_{NDT}[b] (°F)
Replacement Reactor Vessel Closure Head Flange (Heat # E4483/E4484)	-76	-76
Reactor Vessel Flange 06 (Heat # 522582)	-22	-22
Inlet Nozzle 09 (Heat # 990290-11)	-26	-14
Inlet Nozzle 10 (Heat # 990290-12)	-22	-10
Inlet Nozzle 11 (Heat # 990268-21)	3	8
Outlet Nozzle 12 (Heat # 990290-31)	-3	-6
Outlet Nozzle 13 (Heat # 990290-22)	-22	-7
Outlet Nozzle 14 (Heat # 990290-21)	-4	8
Inlet Nozzle to Upper Shell Welds	---	30
Outlet Nozzle to Upper Shell Welds	---	30
Upper Shell Forging 05 (Heat # 990286 / 295213)	6	1
Upper to Intermediate Shell Circumferential Weld (OD 94%) (Heat # 25295)	0	-40
Upper to Intermediate Shell Circumferential Weld (ID 6%) (Heat # 4278)	0	-4
Intermediate Shell Forging 04 (Heat # 990311 / 298244)	17	-6
Intermediate to Lower Shell Circumferential Weld (Heat # 25531)	19	-2
Lower Shell Forging 03 (Heat # 990400 / 292332)	38	33

Notes:

(a) The original initial RT_{NDT} values were taken from the North Anna Power Station UFSAR, Table 5.2-26 (Reference G-3), as available. These values are consistent with those documented in Dominion Calculation SM-1008 (Reference G-2); however, some initial RT_{NDT} values are only listed in the UFSAR or only listed in SM-1008. The UFSAR values are identified as historic.

(b) Current initial RT_{NDT} values correspond to the values utilized herein. In some cases, these values have been updated or defined based on evaluation in PWROG-18005-NP (Reference G-1).

Table G-2 Comparison of Previous and Current Initial RT_{NDT} Values for North Anna Unit 2

Material Identification	Previous Initial RT_{NDT}[(a)] (°F)	Current Initial RT_{NDT}[(b)] (°F)
Replacement Reactor Vessel Closure Head Flange (Heat # H1681/H1682)	-49	-49
Reactor Vessel Flange 06 (Heat # 523000)	-22	-22
Inlet Nozzle Forging 9 (Heat # 990426)	20	11
Inlet Nozzle Forging 10 (Heat # 54567-2)	13	5
Inlet Nozzle Forging 11 (Heat # 54590-2)	-21	-31
Outlet Nozzle Forging 12 (Heat # 990426-22)	1	8
Outlet Nozzle Forging 13 (Heat # 990426-31)	3	1
Outlet Nozzle Forging 14 (Heat # 791291)	-19	-22
Inlet Nozzle to Upper Shell Welds (Multiple Heats)	---	30
Outlet Nozzle to Upper Shell Welds (Multiple Heats)	---	30
Upper Shell Forging 05 (Heat # 990598 / 291396)	9	8
Upper to Intermediate Shell Circumferential Weld (OD 94%) (Heat # 4278)	0	-4
Upper to Intermediate Shell Circumferential Weld (ID 6%) (Heat # 801)	0	10
Intermediate Shell Forging 04 (Heat # 990496 / 292424)	75	69
Intermediate to Lower Shell Circumferential Weld (Heat # 716126)	-48	-67
Lower Shell Forging 03 (Heat # 990533 / 297355)	56	37

Notes:

(a) The original initial RT_{NDT} values were taken from the North Anna Power Station UFSAR, Table 5.2-27 (Reference G-3), as available. These values are consistent with those documented in Dominion Calculation SM-1008 (Reference G-2); however, some initial RT_{NDT} values are only listed in the UFSAR or only listed in SM-1008. The UFSAR values are identified as historic.

(b) Current initial RT_{NDT} values correspond to the values utilized herein. In some cases, these values have been updated or defined based on evaluation in PWROG-18005-NP (Reference G-1).

Table G-3 Comparison of Previous and Current σ_I Values for North Anna Unit 1

Material Identification	Previous σ_I[(a)] (°F)	Current σ_I[(b)] (°F)
Replacement Reactor Vessel Closure Head Flange (Heat # E4483/E4484)	---	0
Reactor Vessel Flange 06 (Heat # 522582)	---	0
Inlet Nozzle 09 (Heat # 990290-11)	---	0
Inlet Nozzle 10 (Heat # 990290-12)	---	0
Inlet Nozzle 11 (Heat # 990268-21)	---	0
Outlet Nozzle 12 (Heat # 990290-31)	---	0
Outlet Nozzle 13 (Heat # 990290-22)	---	0
Outlet Nozzle 14 (Heat # 990290-21)	---	0
Inlet Nozzle to Upper Shell Welds	---	0
Outlet Nozzle to Upper Shell Welds	---	0
Upper Shell Forging 05 (Heat # 990286 / 295213)	30	0
Upper to Intermediate Shell Circumferential Weld (OD 94%) (Heat # 25295)	20	0
Upper to Intermediate Shell Circumferential Weld (ID 6%) (Heat # 4278)	20	0
Intermediate Shell Forging 04 (Heat # 990311 / 298244)	0	0
Intermediate to Lower Shell Circumferential Weld (Heat # 25531)	0	0
Lower Shell Forging 03 (Heat # 990400 / 292332)	0	0

Notes:

(a) The previous σ_I values were taken from Dominion Calculation SM-1008 (Reference G-2), as applicable.

(b) Current σ_I values correspond to the values utilized herein. In some cases, these values have been updated or defined based on evaluation in PWROG-18005-NP (Reference G-1).

Table G-4 Comparison of Previous and Current σ_I Values for North Anna Unit 2

Material Identification	Previous σ_I[(a)] (°F)	Current σ_I[(b)] (°F)
Replacement Reactor Vessel Closure Head Flange (Heat # H1681/H1682)	---	0
Reactor Vessel Flange 06 (Heat # 523000)	---	0
Inlet Nozzle Forging 9 (Heat # 990426)	---	0
Inlet Nozzle Forging 10 (Heat # 54567-2)	---	0
Inlet Nozzle Forging 11 (Heat # 54590-2)	---	0
Outlet Nozzle Forging 12 (Heat # 990426-22)	---	0
Outlet Nozzle Forging 13 (Heat # 990426-31)	---	0
Outlet Nozzle Forging 14 (Heat # 791291)	---	0
Inlet Nozzle to Upper Shell Welds (Multiple Heats)	---	0
Outlet Nozzle to Upper Shell Welds (Multiple Heats)	---	0
Upper Shell Forging 05 (Heat # 990598 / 291396)	30	0
Upper to Intermediate Shell Circumferential Weld (OD 94%) (Heat # 4278)	20	0
Upper to Intermediate Shell Circumferential Weld (ID 6%) (Heat # 801)	20	0
Intermediate Shell Forging 04 (Heat # 990496 / 292424)	0	0
Intermediate to Lower Shell Circumferential Weld (Heat # 716126)	0	0
Lower Shell Forging 03 (Heat # 990533 / 297355)	0	0

Notes:

(a) The previous σ_I values were taken from Dominion Calculation SM-1008 (Reference G-2), as applicable.

(b) Current σ_I values correspond to the values utilized herein. In some cases, these values have been updated or defined based on evaluation in PWROG-18005-NP (Reference G-1).

Table G-5 Comparison of Previous and Current Unirradiated USE Values for North Anna Unit 1

Material Identification	Previous Unirradiated USE[a] (ft-lbs)	Current Unirradiated USE[b] (ft-lbs)
Replacement Reactor Vessel Closure Head Flange (Heat # E4483/E4484)	---	187
Reactor Vessel Flange 06 (Heat # 522582)	105 (161)	≥ 108
Inlet Nozzle 09 (Heat # 990290-11)	69 (106)	≥ 71
Inlet Nozzle 10 (Heat # 990290-12)	57 (88)	≥ 58
Inlet Nozzle 11 (Heat # 990268-21)	52 (80)	56
Outlet Nozzle 12 (Heat # 990290-31)	65 (100)	≥ 66
Outlet Nozzle 13 (Heat # 990290-22)	59 (90)	≥ 59
Outlet Nozzle 14 (Heat # 990290-21)	59 (90)	≥ 59
Inlet Nozzle to Upper Shell Welds	---	72
Outlet Nozzle to Upper Shell Welds	---	72
Upper Shell Forging 05 (Heat # 990286 / 295213)	39[c] (60)	72
Upper to Intermediate Shell Circumferential Weld (OD 94%) (Heat # 25295)	111	112
Upper to Intermediate Shell Circumferential Weld (ID 6%) (Heat # 4278)	105	105
Intermediate Shell Forging 04 (Heat # 990311 / 298244)	92	91
Intermediate to Lower Shell Circumferential Weld (Heat # 25531)	102	95
Lower Shell Forging 03 (Heat # 990400 / 292332)	85	85

Notes contained on the following page.

Notes:

(a) The original unirradiated USE values were taken from the North Anna Power Station UFSAR, Table 5.2-26 (Reference G-3). These values are consistent with those documented in Dominion Calculation SM-1008 (Reference G-2), unless otherwise noted. Some unirradiated USE values are only listed in the UFSAR and reported in the strong direction, i.e., parallel to the major working direction. In this case, the strong direction USE is shown in parenthesis and the weak direction USE is determined by multiplying the strong direction USE by 65% per BTP 5-3, Position 1.2 (Reference G-4). The UFSAR values are identified as historic.

(b) Current Unirradiated USE values correspond to the values utilized herein. In some cases, these values have been updated or defined based on evaluation in PWROG-18005-NP (Reference G-1). A greater than or equal to symbol, "≥", identifies a material with no available upper shelf data; thus, the initial USE values for these materials were conservatively estimated based on the highest recorded absorbed energy points.

(c) SM-1008 reports a USE value in the transverse direction of 74 ft-lbs. Per BAW-2224 (Reference G-5), this value is equal to the limiting USE of the North Anna Units 1 and 2 beltline forgings.

Table G-6 Comparison of Previous and Current Unirradiated USE Values for North Anna Unit 2

Material Identification	Previous Unirradiated USE[(a)] (ft-lbs)	Current Unirradiated USE[(b)] (ft-lbs)
Replacement Reactor Vessel Closure Head Flange (Heat # H1681/H1682)	---	192
Reactor Vessel Flange 06 (Heat # 523000)	95 (146)	≥ 95
Inlet Nozzle Forging 9 (Heat # 990426)	47 (72)	56
Inlet Nozzle Forging 10 (Heat # 54567-2)	77 (118)	≥ 77
Inlet Nozzle Forging 11 (Heat # 54590-2)	60 (92)	≥ 75
Outlet Nozzle Forging 12 (Heat # 990426-22)	52 (80)	≥ 60
Outlet Nozzle Forging 13 (Heat # 990426-31)	40 (62)	56
Outlet Nozzle Forging 14 (Heat # 791291)	75 (115)	≥ 74
Inlet Nozzle to Upper Shell Welds (Multiple Heats)	---	75
Outlet Nozzle to Upper Shell Welds (Multiple Heats)	---	75
Upper Shell Forging 05 (Heat # 990598 / 291396)	56[(c)] (86)	72
Upper to Intermediate Shell Circumferential Weld (OD 94%) (Heat # 4278)	105	105
Upper to Intermediate Shell Circumferential Weld (ID 6%) (Heat # 801)	90	75
Intermediate Shell Forging 04 (Heat # 990496 / 292424)	74	72
Intermediate to Lower Shell Circumferential Weld (Heat # 716126)	107	109
Lower Shell Forging 03 (Heat # 990533 / 297355)	80	80

Notes contained on the following page.

Notes:

(a) The original unirradiated USE values were taken from the North Anna Power Station UFSAR, Table 5.2-27 (Reference G-3). These values are consistent with those documented in Dominion Calculation SM-1008 (Reference G-2), unless otherwise noted. Some unirradiated USE values are only listed in the UFSAR and reported in the strong direction, i.e., parallel to the major working direction. In this case, the strong direction USE is shown in parenthesis and the weak direction USE is determined by multiplying the strong direction USE by 65% per BTP 5-3, Position 1.2 (Reference G-4). The UFSAR values are identified as historic.

(b) Current Unirradiated USE values correspond to the values utilized herein. In some cases, these values have been updated or defined based on evaluation in PWROG-18005-NP (Reference G-1). A greater than or equal to symbol, "≥", identifies a material with no available upper shelf data; thus, the initial USE values for these materials were conservatively estimated based on the highest recorded absorbed energy points.

(c) SM-1008 reports a USE value in the transverse direction of 74 ft-lbs. Per BAW-2224 (Reference G-5), this value is equal to the limiting USE of the North Anna Units 1 and 2 beltline forgings.

G.1 REFERENCES

G-1. Pressurized Water Reactor (PWR) Owners Group (PWROG) Report PWROG-18005-NP, Revision 2, "Determination of Unirradiated RT_{NDT} and Upper-Shelf Energy Values of the North Anna Units 1 and 2 Reactor Vessel Materials," September 2019.

G-2. Dominion Calculation SM-1008, Revision 0, Addendum M.

G-3. North Anna Power Station (NAPS) Updated Final Safety Analysis Report (UFSAR), Amendment No. 54, September 2018.

G-4. NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Chapter 5 of LWR Edition, Branch Technical Position 5-3, "Fracture Toughness Requirements," Revision 2, U.S. Nuclear Regulatory Commission, March 2007. *[ADAMS Accession Number ML070850035]*

G-5. B&W Nuclear Technologies Report BAW-2224, "North Anna Units 1 and 2 Response to Closure Letter for NRC Generic Letter 92-01, Revision 1," July 1994.

G-6. Code of Federal Regulations, 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Federal Register, Volume 60, No. 243, dated December 19, 1995.

APPENDIX H USE OF MASTER CURVE DATA FOR NORTH ANNA UNIT 1 LOWER SHELL FORGING

In order to support future asset management considerations, Master Curve testing was performed to determine the fracture toughness transition temperature (T_0) of the North Anna Unit 1 Lower Shell Forging 03. The Master Curve method provides a more realistic and appropriate reference temperature than the American Society of Mechanical Engineers (ASME) Code Section III NB-2331 method, which is used to determine initial RT_{NDT}. The Master Curve method is based on fracture toughness measurements per ASTM E1921, as opposed to the Charpy impact and drop-weight measurements associated with ASME Section III NB-2331. WCAP-18463-NP (Reference H-1) contains the methodology for performing embrittlement calculation with Master Curve data. Note that the Master Curve data provided in this Appendix does not represent the current design or licensing basis values, nor is it proposed to use it for SLR, but is only to support future asset management considerations.

H.1 METHODOLOGY

WCAP-18463-NP (Reference H-1) describes the methodologies for the use of T_0 results to determine a reference temperature based on Master Curve testing data (RT_{T0}) and the use of RT_{T0} in reactor vessel integrity analyses, for North Anna Unit 1 Lower Shell Forging 03 (e.g., ART calculations). The methodology is similar to Regulatory Guide 1.99, Revision 2 with minor modifications as shown below to account for differences between RT_{T0} and RT_{NDT}. The methodology also accounts for the initial RT_{T0} of North Anna Unit 1 Lower Shell Forging 03 being based on data from specimens irradiated in Capsule V from the North Anna Unit 1 Reactor Vessel Surveillance Program.

Per Regulatory Guide 1.99, Revision 2, the ART is calculated as shown below:

$$ART = RT_{NDT} + \Delta RT_{NDT} + \text{Margin}$$

However, there is a potential 10% bias when comparing ΔRT_{NDT} and ΔT_0 values for plate and forging materials. Therefore, a 10% bias should be applied to Regulatory Guide 1.99, Revision 2 methodology calculations of ΔRT_{NDT} for North Anna Unit 1 Lower Shell Forging 03 when the initial RT_{T0} is used. This bias is included as a part of an Adjustment term.

It is important to note that the T_0 results for North Anna Unit 1 Lower Shell Forging 03 are based on materials from North Anna Unit 1 Surveillance Capsule V, which were irradiated for 1 cycle of operation. The Charpy V-notch tests, performed at the time of removal, exhibited a measured 30 ft-lb shift of 29°F in the axial ("weak") direction. The irradiation induced shift can be credited in the determination of ART values. Thus, it is appropriate to remove 29°F of calculated margin when analyzing the North Anna Unit 1 Lower Shell Forging 03. This 29°F is considered as a reduction in an Adjustment term.

Therefore, the method to be used in ART calculations which use an initial RT_{T0} for North Anna Unit 1 Lower Shell Forging 03 is as follows per WCAP-18463-NP:

$$ART = RT_{T0} + \Delta RT_{NDT} + \text{Margin} + \text{Adjustment}$$

$$\text{Margin} = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2}$$

$$\text{Adjustment} = 0.1 * \Delta RT_{NDT} - \Delta T_{ME}$$

Where:

RT_{T0} = 10.1°F per WCAP-18463-NP

σ_I = 0°F per WCAP-18463-NP[(1)]

$\sigma_\Delta = \sigma_\Delta$ determined per Regulatory Guide 1.99, Revision 2

$\Delta RT_{NDT} = \Delta RT_{NDT}$ determined per Regulatory Guide 1.99, Revision 2

ΔT_{ME} = Measured Material Embrittlement = 29°F

The ART values calculated in this manner are appropriate for input to an ASME Section XI, Appendix G analysis.

H.2 CALCULATION

This section utilizes the RT_{T0} in ART calculations, similar to those calculations performed in Section 5 of the body of this report, in order to demonstrate the margin gained for the improved material condition. Table H-1 presents the unirradiated RT_{T0} ($RT_{T0(U)}$) value, which is taken from WCAP-18463-NP (Reference H-1) and was determined with Master Curve testing results and ASTM E1921-19. A σ_I value of 0°F is associated with this $RT_{NDT(U)}$ ($RT_{T0(U)}$) value. This value is then used in Table H-2 to calculate the ART values at the 1/4T and 3/4T locations at 72 EFPY. The results show significant reductions in the ART values at the 1/4T and 3/4T locations at 72 EFPY.

Table H-1 North Anna Unit 1 Unirradiated RT_{NDT} Values with and without Master Curve Data

Material Description	Heat Number	Wt. % Cu	Wt. % Ni	$RT_{NDT(U)}$ or $RT_{T0(U)}$ (°F)	Method
Lower Shell Forging 03	990400 / 292332	0.156	0.817	33	ASME Code Section III, NB-2331
				10.1	Master Curve Method

(1) WCAP-18463-NP recommends a σ_I = 14.2°F, but notes σ_I = 0°F is appropriate since RT_{T0} is based on material-specific North Anna Unit 1 Lower Shell Forging 03 measured data.

Table H-2 Calculation of ART Values Using Master Curve Data for North Anna Unit 1 Lower Shell Forging 03

Material	Heat Number	R.G. 1.99, Rev. 2 Position	CF(a)	1/4T Fluence(b) (x 10^{19} n/cm², E > 1.0 MeV)	1/4T FF(b)	Method	$RT_{NDT(U)}$ or $RT_{T0(U)}$(c) (°F)	Predicted ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ(d) (°F)	M (°F)	ART (°F)
Calculation of the North Anna Unit 1 Lower Shell Forging 03 ART Values at the 1/4T Location at 72 EFPY												
Lower Shell Forging 03	990400 / 292332	1.1	119.97	4.54	1.383	NB-2331	33	165.9	0.0	17.0	34.0	232.9
						Master Curve	10.1	165.9	0.0	17.0	21.6(f)	197.6
Using non-credible surveillance data(h)		2.1	81.68	4.54	1.383	NB-2331	33	113.0	0.0	17.0	34.0	180.0
						Master Curve	10.1	113.0	0.0	17.0	16.3(f)	139.4
Calculation of the North Anna Unit 1 Lower Shell Forging 03 ART Values at the 3/4T Location at 72 EFPY												
Lower Shell Forging 03	990400 / 292332	1.1	119.97	1.81	1.383	NB-2331	33	139.5	0.0	17.0	34.0	206.5
						Master Curve	10.1	139.5	0.0	17.0	18.9(f)	168.5
Using non-credible surveillance data(e)		2.1	81.68	1.81	1.163	NB-2331	33	95.0	0.0	17.0	34.0	162.0
						Master Curve	10.1	95.0	0.0	17.0	14.5(f)	119.6

Notes:

(a) Chemistry factors are taken from Table 3-7 of this report.

(b) Fluence and Fluence Factors are taken from Tables 2-1 and 5-2 of this report.

(c) $RT_{NDT(U)}$ and $RT_{T0(U)}$ values are taken from Table H-1 of this calculation note.

(d) Per the guidance of Regulatory Guide 1.99, Revision 2 (Reference 1), the base metal σ_Δ = 17°F for Position 1.1 and Position 2.1 with non-credible surveillance data. However, σ_Δ need not exceed 0.5*ΔRT_{NDT} for either forgings or welds with or without surveillance data.

(e) The credibility evaluation for the North Anna Unit 1 surveillance data in Appendix E of this report determined that the Lower Shell Forging 03 surveillance data are deemed non-credible.

(f) For this calculation, the value shown is "Margin + Adjustment" = $2\sqrt{\sigma_I^2 + \sigma_\Delta^2} + 0.1*\Delta RT_{NDT} - \Delta T_{ME}$, where ΔT_{ME} = 29°F.

H.3 REFERENCES

H-1. Westinghouse Report WCAP-18463-NP, Revision 0, "Determination and Use of RT_{T0} for North Anna Unit 1 Lower Shell Forging 03," August 2019.

H-2. U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988. *[ADAMS Accession Number ML003740284]*

APPENDIX I NORTH ANNA UNITS 1 AND 2 LICENSING BASIS FOR DETERMINING CHEMISTRY FACTOR WHEN SURVEILANCE DATA IS AVAILABLE

Regulatory Guide (RG) 1.99, Revision 2 (Reference I-1), indicates the Position 2.1 CFs may be used with a reduced margin term whenever the surveillance data has been deemed credible using the methodology described in Appendix E. Regulatory Guide 1.99, Revision 2 is less prescriptive if the surveillance data is deemed non-credible. However, additional guidance can be drawn from 10 CFR 50.61 (Reference I-2) and the NRC Generic Letter (GL) 92-01 Guidance (Reference I-3).

10 CFR 50.61(c)(2) states that "licensees shall consider plant specific information that could affect the level of embrittlement". The plant specific information referred to in 10 CFR 50.61(c)(2) is data derived from reactor vessel materials surveillance programs that must be considered in the determination of ΔRT_{NDT} for the beltline material. GL 92-01 Guidance, published by NRC on February 12, 1998, describes the treatment of surveillance data for application to the corresponding reactor vessel beltline material, including: how to apply data from different data sources; how to correct surveillance data (ΔRT_{NDT}) values for differences in irradiation temperature and chemical composition relative to the reactor vessel beltline material being evaluated; how to evaluate the scatter of ΔRT_{NDT} data around a best fit Position 2.1 CF ΔRT_{NDT} trend line to determine surveillance data credibility; and how to compare surveillance data against a Position 1.1 CF trend line based on mean surveillance material chemical compositions to determine conservatism of the use of a Position 1.1 CF. As described in the NRC GL 92-01 Guidance, individual surveillance data points are not to be discarded on the basis of their deviation from a best fit Position 2.1 CF trend line alone; there must also be a recorded deficiency or a physical basis for classifying the data point as atypical. The same logic would apply to not discarding a data set.

When surveillance data is deemed non-credible per RG 1.99, NRC GL 92-01 Guidance considers the surveillance data to still be applicable for characterizing the beltline material through the direction of its use with a full margin term (σ_Δ) to establish the Position 2.1 RT_{NDT} when the Position 1.1 CF is concluded to be non-conservative based on that same data (See case 3 in the NRC GL 92-01 Guidance). Dominion Energy proposed and licensed logically consistent application of surveillance data concluded to be non-credible due only to data scatter for both non-conservative and conservative RG 1.99, Position 1.1 CFs. The Dominion licensing position is as follows:

a. The greater of the RG 1.99 Revision 2 Position 1.1 CF and 2.1 CF is used with a full margin term (σ_Δ = 17°F for base metal and σ_Δ = 28°F for welds) for evaluation of the reactor vessel beltline material when one or more of the surveillance data fall outside of the Position 2.1 CF trend line by more than one times σ_Δ (data is non-credible), and one or more of the surveillance data fall more than two times σ_Δ above the Position 1.1 CF trend line (Position 1.1 CF is <u>non-conservative</u>)

b. The lesser of the RG 1.99 Revision 2 Position 1.1. and 2.1 CFs is used with a full margin term (σ_Δ = 17°F for base metal and σ_Δ = 28°F for welds) for evaluation of the reactor vessel beltline material when one or more of the surveillance data fall outside of the Position 2.1 CF trend line by more than one times σ_Δ (data is non-credible), and none of the surveillance data fall more than two times σ_Δ above the Position 1.1 CF trend line (Position 1.1 CF is <u>conservative</u>).

Table I-1 contains the list of correspondences applicable to the submittal and approval of this licensing basis for North Anna Units 1 and 2. Appendixes I.1 and I.2 evaluate whether the RG 1.99, Position 1.1 CFs are conservative based on the above licensing basis and the North Anna Units 1 and 2 surveillance data.

Table I-1 North Anna Units 1 and 2 Chemistry Factor Licensing Basis History

Subject Document(s)	Content Relevant to North Anna Units 1 and 2 Chemistry Factor Licensing Basis	Date	Reference Number(s)
NRC Letter (Incoming Correspondence No. 99-361)	Closure of the NRC review of the North Anna Units 1 and 2 GL 92-01 response. This letter notes that for material with non-credible surveillance but with all measured ΔRT_{NDT} data points below the RG 1.99, Position 1.1 ΔRT_{NDT} predictions + $2\sigma_\Delta$, the RG 1.99, Position 2.1 CFs were used with a full margin value to calculate RT_{PTS}.	June 23, 1999	I-4
VEPCO Letter Serial No. 99-361	Response to the NRC on discrepancies between the Reactor Vessel Integrity Database (RVID) and previously provided data. This letter reiterates the use of non-credible surveillance data with a full margin value when all measured ΔRT_{NDT} data points are below the RG 1.99, Position 1.1 ΔRT_{NDT} predictions + $2\sigma_\Delta$.	Sept. 1, 1999	I-5
VEPCO Letter Serial No. 99-452A	Evaluation of the material properties based on the results from North Anna Unit 1, Capsule W. This letter reiterates the use of non-credible surveillance data with a full margin value when all measured ΔRT_{NDT} data points are below the RG 1.99, Position 1.1 ΔRT_{NDT} predictions + $2\sigma_\Delta$. It also includes portions of SM-1008.	Nov. 19, 1999	I-6
VEPCO Letter Serial No. 00-306	Submittal of the 32.3 EFPY (Unit 1) and 34.3 EFPY (Unit 2) heatup and cooldown curves and Low Temperature Overpressure Protection System (LTOPS) setpoints. This letter includes a detailed evaluation based on the latest reactor vessel materials surveillance data, i.e. North Anna Unit 1, Capsule W. The North Anna Unit 1, Capsule W report, BAW-2356, is also attached.	June 22, 2000	I-7
VEPCO Letters Nos. 01-020, 01-020A, 01-020B, 01-168, 01-168A	Corrections and supplements to the above cited letters. No changes were made to the material property basis in these letters.	Various	I-8, I-9, I-10, I-11, & I-12
VEPCO Letter No. 01-262	Revises the RVID for North Anna Unit 2 based on the Sequoyah Unit 2 surveillance data. This letter demonstrates that conservatism or non-conservatism of the RG 1.99 ΔRT_{NDT} prediction is determined by whether all measured ΔRT_{NDT} data points are below the RG 1.99, Position 1.1 ΔRT_{NDT} predictions + $2\sigma_\Delta$.	April 27, 2001	I-13

Table I-1 North Anna Units 1 and 2 Chemistry Factor Licensing Basis History

Subject Document(s)	Content Relevant to North Anna Units 1 and 2 Chemistry Factor Licensing Basis	Date	Reference Number(s)
NRC Letter (Incoming Correspondence No. 01-293)	Safety Evaluation (SE) for the 32.3 EFPY (Unit 1) and 34.3 EFPY (Unit 2) heatup and cooldown curves and LTOPS setpoints. Reiterates that for material with non-credible surveillance data but all measured ΔRT_{NDT} data points below the RG 1.99, Position 1.1 ΔRT_{NDT} predictions + $2\sigma_{\Delta}$, the RG 1.99, Position 2.1 CFs were used with a full margin value.	May 2, 2001	I-14
VEPCO Letter No. 01-282	License Renewal Submittal	May 29, 2001	I-15
VEPCO Letter No. 02-601	License Renewal RAI Response on RV embrittlement issues. Reiterates that conservatism is defined as below RG Position 1.1 ΔRT_{NDT} predictions $+2\sigma_{\Delta}$.	Oct. 15, 2002	I-16
VEPCO Letter No. 04-380	Submittal of the 50.3 EFPY (Unit 1) and 52.3 EFPY (Unit 2) heatup and cooldown curves and LTOPS setpoints.	July 1, 2004	I-17
VEPCO Letter No. 04-380A	Provides RAI responses related to the 50.3 EFPY / 52.3 EFPY P-T curves submittal.	Oct. 28, 2004	I-18
VEPCO Letter No. 04-380B	Editorial correction for the 50.3 EFPY/52.3 EFPY P-T curves submittal.	Nov. 16, 2004	I-19
NRC Letter (Incoming Correspondence No. 05-460)	SE for the 50.3 EFPY (Unit 1) and 52.3 EFPY (Unit 2) heatup and cooldown curves and LTOPS setpoints.	July 8, 2005	I-20

I.1 Regulatory Guide 1.99, Revision 2, Position 1.1 Conservativism Evaluation for North Anna Unit 1 Surveillance Data

The purpose of this evaluation is to determine if the North Anna Unit 1 non-credible surveillance data is less than $2\sigma_\Delta$ above of the RG 1.99, Position 1.1 prediction, thereby, determining whether the RG 1.99, Position 1.1 CF is conservative. This evaluation is performed in Table I-2.

Table I-2 Conservatism Check for Position 1.1 for Non-Credible North Anna Unit 1 Surveillance Data

Material	Capsule	CF[(a)] (Pos. 1.1) (°F)	Capsule Fluence[(b)] (x 10^{19} n/cm^2)	FF[(c)]	Measured ΔRT_{NDT}[(b)] (°F)	Predicted ΔRT_{NDT}[(d)] (°F)	Scatter ΔRT_{NDT}[(e)] (°F)	< 2σ[(f)]
Lower Shell Forging 03 (Tangential)	V	121.63	0.306	0.675	51	82.1	-31.1	Yes
	U	121.63	0.914	0.975	116	118.6	-2.6	Yes
	W	121.63	2.05	1.196	93	145.4	-52.4	Yes
Lower Shell Forging 03 (Axial)	V	121.63	0.306	0.675	29	82.1	-53.1	Yes
	U	121.63	0.914	0.975	72	118.6	-46.6	Yes
	W	121.63	2.05	1.196	96	145.4	-49.4	Yes
Surveillance Weld Material (Heat # 25531)	V	56.22	0.306	0.675	88	38.0	50.0	Yes
	U	56.22	0.914	0.975	30	54.8	-24.8	Yes
	W	56.22	2.05	1.196	86	67.2	18.8	Yes

Notes:

(a) CF values are taken from Table 3-7.

(b) Fluence and Measured ΔRT_{NDT} values are taken from Table 3-4.

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

(d) Predicted ΔRT_{NDT} = CF * FF

(e) Scatter ΔRT_{NDT} = Measured ΔRT_{NDT} - Predicted ΔRT_{NDT}

(f) σ_Δ = 17°F for base metal and σ_Δ = 28°F for welds

All data points are no more than $2\sigma_\Delta$ above of the RG 1.99, Position 1.1 prediction. Therefore, the RG 1.99, Position 1.1 CF is conservative, and the RG 1.99, Position 2.1 CF may be used with a full margin term for North Anna Unit 1 Lower Shell Forging 03 and Heat # 25531 weld material.

I.2 Regulatory Guide 1.99, Revision 2, Position 1.1 Conservativism Evaluation for North Anna Unit 2 Surveillance Data

The purpose of this evaluation is to determine if the North Anna Unit 2 non-credible surveillance data is less than $2\sigma_\Delta$ above of the RG 1.99, Position 1.1 prediction, thereby, determining whether the RG 1.99, Position 1.1 CF is conservative. This evaluation is performed in Table I-3.

Note the North Anna Unit 2 surveillance weld Heat # 716126 was determined to be credible in Appendix E. Therefore, the conservatism of the Position 1.1 CF does not need to be determined because the Position 2.1 CF with a reduced margin term will be used regardless.

Table I-3 Conservatism Check for Position 1.1 for Non-Credible North Anna Unit 2 Surveillance Data

Material	Capsule	CF(a) (Pos. 1.1) (°F)	Capsule Fluence(b) (x 10^{19} n/cm^2)	FF(c)	Measured ΔRT_{NDT}(b) (°F)	Predicted ΔRT_{NDT}(d) (°F)	Scatter ΔRT_{NDT}(e) (°F)	< 2σ(f)
Intermediate Shell Forging 04 (Tangential)	V	82.40	0.286	0.658	19	54.2	-35.2	Yes
	U	82.40	0.985	0.996	33	82.1	-49.1	Yes
	W	82.40	2.08	1.199	86	98.8	-12.8	Yes
Intermediate Shell Forging 04 (Axial)	V	82.40	0.286	0.658	21	54.2	-33.2	Yes
	U	82.40	0.985	0.996	66	82.1	-16.1	Yes
	W	82.40	2.08	1.199	65	98.8	-33.8	Yes

Notes:

(a) CF values are taken from Table 3-8.

(b) Fluence and Measured ΔRT_{NDT} values are taken from Table 3-6.

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

(d) Predicted ΔRT_{NDT} = CF * FF

(e) Scatter ΔRT_{NDT} = Measured ΔRT_{NDT} - Predicted ΔRT_{NDT}

(f) σ_Δ = 17°F for base metal and σ_Δ = 28°F for welds

All data points are no more than $2\sigma_\Delta$ above of the RG 1.99, Position 1.1 prediction. Therefore, the RG 1.99, Position 1.1 CF is conservative, and the RG 1.99, Position 2.1 CF may be used with a full margin term for North Anna Unit 2 Intermediate Shell Forging 04.

I.3 REFERENCES

I-1. U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988. *[Agencywide Documents Access and Management System (ADAMS) Accession Number ML003740284]*

I-2. Code of Federal Regulations 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," U.S. Nuclear Regulatory Commission, Federal Register, Volume 60, No. 243, dated December 19, 1995, effective January 18, 1996.

I-3. K. Wichman, M. Mitchell, and A. Hiser, US NRC, Generic Letter 92-01 and RPV Integrity Workshop Handouts, "NRC/Industry Workshop on RPV Integrity Issues," February 12, 1998. *[ADAMS Accession Number ML110070570]*

I-4. NRC Letter "Closure of the Review of the Response to Generic Letter 92-01, Revision 1, Supplement 1, 'Reactor Vessel Structural Integrity,' the North Anna Nuclear Power Plant, Units 1 and 2 (TAC Nos. MA0555 and MA0556)," June 23, 1999. [Dominion Serial No. 99-361, Incoming NRC Letter]

I-5. VEPCO Letter 99-361, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Response to NRC Request for Comments Generic Letter 92-01, Revision 1, Supplement 1," September 1, 1999.

I-6. VEPCO Letter 99-452A, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Surry Power Station Units 1 and 2, Evaluation of Reactor Vessel Materials Surveillance Data," November 19, 1999.

I-7. VEPCO Letter 00-306, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Proposed Technical Specifications Changes Requests for Exemption per 10 CFR 50.60(b) Reactor Coolant System Pressure/Temperature Limits, LTOPS Setpoints, and LTOPS Enable Temperatures," June 22, 2000.

I-8. VEPCO Letter 01-020, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Response to Request for Additional Information, Proposed Technical Specifications Changes, Reactor Coolant System Pressure/Temperature Limits, LTOPS Setpoints, and LTOPS Enable Temperatures," January 4, 2001.

I-9. VEPCO Letter 01-020A, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Response to Request for Additional Information, Proposed Technical Specifications Changes, Reactor Coolant System Pressure/Temperature Limits, LTOPS Setpoints, and LTOPS Enable Temperatures," February 14, 2001.

I-10. VEPCO Letter 01-020B, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Response to Request for Additional Information and Clarification of Exemption Request Regarding Proposed Technical Specifications Changes for Reactor Coolant System Pressure/Temperature Limits, LTOPS Setpoints, and LTOPS Enable Temperatures," March 13, 2001.

I-11. VEPCO Letter 01-168, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Response to Request for Additional Information, Proposed Technical Specifications Changes, Reactor Coolant System Pressure/Temperature Limits, LTOPS Setpoints, and LTOPS Enable Temperatures," March 22, 2001.

I-12. VEPCO Letter 01-168A, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Proposed Technical Specifications Changes, Editorial Correction to Proposed Reactor Coolant System Pressure/Temperature Limit Curves Applicable to Cooldown," April 11, 2001.

I-13. VEPCO Letter 01-262, "Virginia Electric and Power Company, North Anna Power Station Unit 2, Application of Sequoyah 2 Surveillance Data to North Anna Unit 2 Reactor Vessel Weld Material Fabricated from Weld Wire Heat 4278," April 27, 2001.

I-14. NRC Letter "North Anna Power Station, Units 1 and 2 - Issuance of Amendments and Exemption from the Requirements of 10 CFR Part 50, Section 50.60(a) Re: Amended Pressure-Temperature Limits (TAC Nos. MA9343, MA9344, MA9347, and MA9348)," May 2, 2001. [Dominion Serial No. 01-293, Incoming NRC Letter]

I-15. VEPCO Letter 01-282, "Virginia Electric and Power Company, Surry and North Anna Power Stations Units 1 and 2 License Renewal Applications - Submittal," May 29, 2001. *[ADAMS Accession Number ML011500496]*

I-16. VEPCO Letter 02-601, "Virginia Electric and Power Company (Dominion), Surry and North Anna Power Stations Units 1 and 2, Response to Request for Supplemental Information, License Renewal Applications," October 15, 2002. *[ADAMS Accession Number ML022960411]*

I-17. VEPCO Letter 04-380, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Proposed Technical Specifications Change Request, Reactor Coolant System Pressure/Temperature Limits, LTOPS Setpoints and LTOPS Enable Temperatures," July 1, 2004.

I-18. VEPCO Letter 04-380A, "Virginia Electric and Power Company (Dominion), North Anna Power Station Units 1 and 2, Request for Additional Information, Proposed Technical Specifications Change Request, Reactor Coolant System Pressure/Temperature Limits, LTOPS Setpoints and LTOPS Enable Temperatures," October 28, 2004.

I-19. VEPCO Letter 04-380B, "Virginia Electric and Power Company (Dominion), North Anna Power Station Units 1 and 2, Editorial Correction for Proposed Technical Specifications Change Request, Reactor Coolant System Pressure/Temperature Limits, LTOPS Setpoints and LTOPS Enable Temperatures," November 16, 2004.

I-20. NRC Letter "North Anna Power Station, Units 1 and 2 - Issuance of Amendments on Reactor Coolant System Pressure and Temperature Limits (TAC Nos. MC3705 and MC3706)," July 8, 2005. [Dominion Serial No. 05-460, Incoming NRC Letter]

APPENDIX J CURRENT TECHNICAL SPECIFICATIONS HEATUP AND COOLDOWN CURVES FOR NORTH ANNA UNITS 1 AND 2

Figures J-1 and J-2 show the North Anna Units 1 and 2 heatup and cooldown curves, respectively, as currently depicted in the North Anna Power Station Technical Specifications (Reference J-1). Tables J-2 and J-3 provide the data points corresponding to the heatup and cooldown curves, respectively, as currently depicted in the North Anna Power Station Technical Specifications. The data points were calculated by modifying the End of License Renewal data points in WCAP-15112 (Reference J-2) with the correction factors in Table J-1 associated with the current Technical Specifications.

Table J-1 North Anna Units 1 and 2 Pressure and Temperature Correction Factors

Type	Current TS Value[(a)]	Units
Pressure adjustment for head loss.	57[(b)]	psid
Pressure correction for instrument uncertainty.	70.1	psid
Temperature correction for instrument uncertainty.	13.5	°F

Notes:

(a) Values were taken from Reference J-3.

(b) This value considers one reactor coolant pump (RCP), two RCP, and three RCP operation.

The corrections are made to the unadjusted P-T limits as follows:

(1) The pressure difference between the point of measurement (Narrow Range or Wide Range RCS pressure measured in the RCS hot leg) and the point of interest (reactor vessel beltline) due to head loss is subtracted from the pressure as calculated for the P-T limit curves;

(2) The uncertainty associated with the RCS pressure instrumentation is subtracted from the pressure as calculated for the P-T limit curves; and

(3) The uncertainty associated with the RCS temperature instrumentation is added to the temperature as calculated for the P-T limit curves.

These data points are consistent with those in VEPCO letter 04-380 (Reference J-3).

J.1 REFERENCES

J-1. North Anna Power Station Technical Specifications, Revised August 8, 2018.

J-2. Westinghouse Report WCAP-15112, Revision 2, "North Anna Units 1 and 2 WOG Reactor Vessel 60-Year Evaluation Minigroup Heatup and Cooldown Limit Curves for Normal Operation," March 2001.

J-3. VEPCO Letter 04-380, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Proposed Technical Specifications Change Request, Reactor Coolant System Pressure/Temperature Limits, LTOPS Setpoints and LTOPS Enable Temperatures," July 1, 2004.

RCS P/T Limits
3.4.3

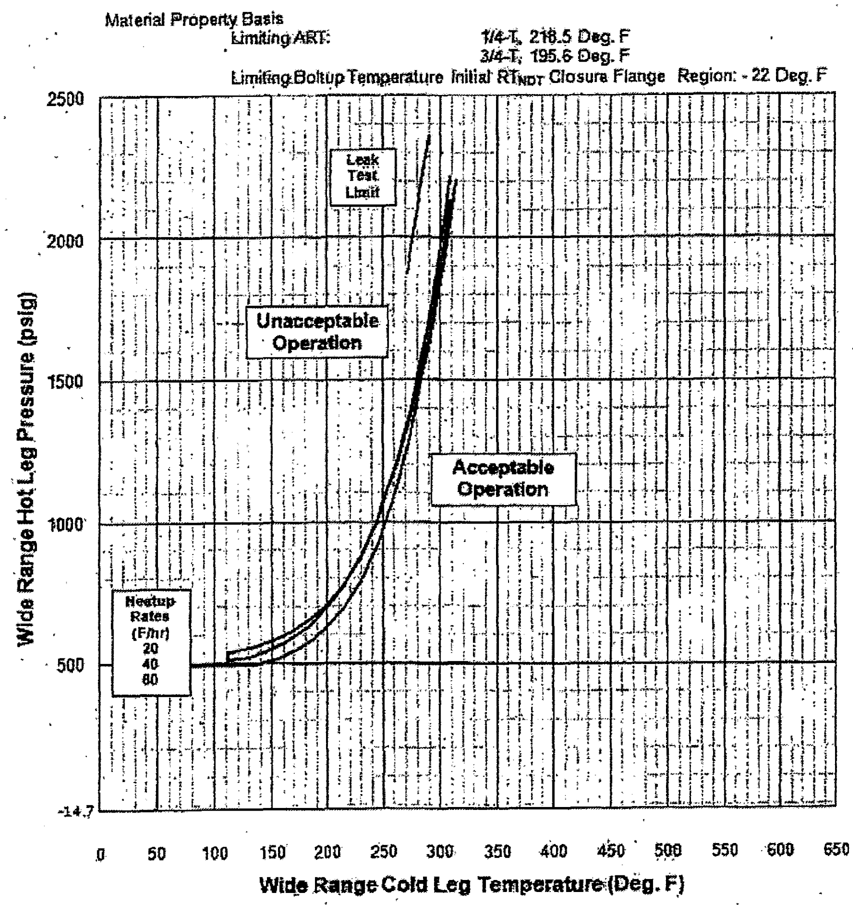

Figure 3.4.3-1 (page 1 of 1)
North Anna Units 1 and 2 Reactor Coolant System Heatup Limitations
(Heatup Rates up to 60°F/hr),
Applicable for the first 50.3 EFPY for Unit 1, and 52.3 EFPY for Unit 2
(Including Margins for Instrumentation Errors)

North Anna Units 1 and 2 3.4.3-3 Amendments 275, 257

Figure J-1 Current North Anna Units 1 and 2 Technical Specifications Heatup P-T Limit Curves

Table J-2 Current North Anna Units 1 and 2 Technical Specifications EOLE Heatup Curves Data Points (with K_{Ic}, with Flange Requirements, and with Margins for Instrumentation Errors and Pressure Correction)

20°F/hr Heatup		40°F/hr Heatup		60°F/hr Heatup	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
73.5	493.9	73.5	493.9	73.5	490.9
78.5	493.9	78.5	493.9	78.5	490.9
83.5	493.9	83.5	493.9	83.5	490.9
88.5	493.9	88.5	493.9	88.5	490.9
93.5	493.9	93.5	493.9	93.5	490.9
98.5	493.9	98.5	493.9	98.5	490.9
103.5	493.9	103.5	493.9	103.5	490.9
108.5	493.9	108.5	493.9	108.5	490.9
111.5	493.9	111.5	493.9	111.5	490.9
111.5	536.9	111.5	512.9	111.5	490.9
113.5	538.9	113.5	512.9	113.5	490.9
118.5	542.9	118.5	514.9	118.5	490.9
123.5	546.9	123.5	517.9	123.5	490.9
128.5	551.9	128.5	521.9	128.5	491.9
133.5	556.9	133.5	526.9	133.5	493.9
138.5	562.9	138.5	532.9	138.5	496.9
143.5	568.9	143.5	538.9	143.5	501.9
148.5	575.9	148.5	546.9	148.5	506.9
153.5	583.9	153.5	555.9	153.5	512.9
158.5	591.9	158.5	564.9	158.5	519.9
163.5	601.9	163.5	575.9	163.5	528.9
168.5	611.9	168.5	587.9	168.5	537.9
173.5	623.9	173.5	600.9	173.5	548.9
178.5	636.9	178.5	615.9	178.5	560.9
183.5	650.9	183.5	631.9	183.5	573.9
188.5	666.9	188.5	649.9	188.5	588.9
193.5	683.9	193.5	669.9	193.5	604.9
198.5	702.9	198.5	691.9	198.5	623.9
203.5	723.9	203.5	715.9	203.5	643.9
208.5	747.9	208.5	742.9	208.5	665.9
213.5	773.9	213.5	772.9	213.5	690.9
218.5	801.9	218.5	801.9	218.5	718.9
223.5	833.9	223.5	833.9	223.5	748.9
228.5	867.9	228.5	867.9	228.5	782.9
233.5	906.9	233.5	906.9	233.5	819.9
238.5	949.9	238.5	949.9	238.5	860.9
243.5	996.9	243.5	996.9	243.5	905.9
248.5	1048.9	248.5	1048.9	248.5	955.9
253.5	1105.9	253.5	1105.9	253.5	1010.9
258.5	1163.9	258.5	1161.9	258.5	1071.9
263.5	1228.9	263.5	1221.9	263.5	1138.9
268.5	1299.9	268.5	1287.9	268.5	1212.9
273.5	1378.9	273.5	1359.9	273.5	1294.9

Table J-2 Current North Anna Units 1 and 2 Technical Specifications EOLE Heatup Curves Data Points (with K_{Ic}, with Flange Requirements, and with Margins for Instrumentation Errors and Pressure Correction)

20°F/hr Heatup		40°F/hr Heatup		60°F/hr Heatup	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
278.5	1465.9	278.5	1439.9	-	-
283.5	1562.9	283.5	1528.9	283.5	1485.9
288.5	1668.9	288.5	1626.9	288.5	1591.9
293.5	1785.9	293.5	1734.9	293.5	1690.9
298.5	1915.9	298.5	1853.9	298.5	1800.9
303.5	2059.9	303.5	1985.9	303.5	1921.9
308.5	2217.9	308.5	2130.9	308.5	2055.9
-	-	-	-	313.5	2202.9

RCS P/T Limits
3.4.3

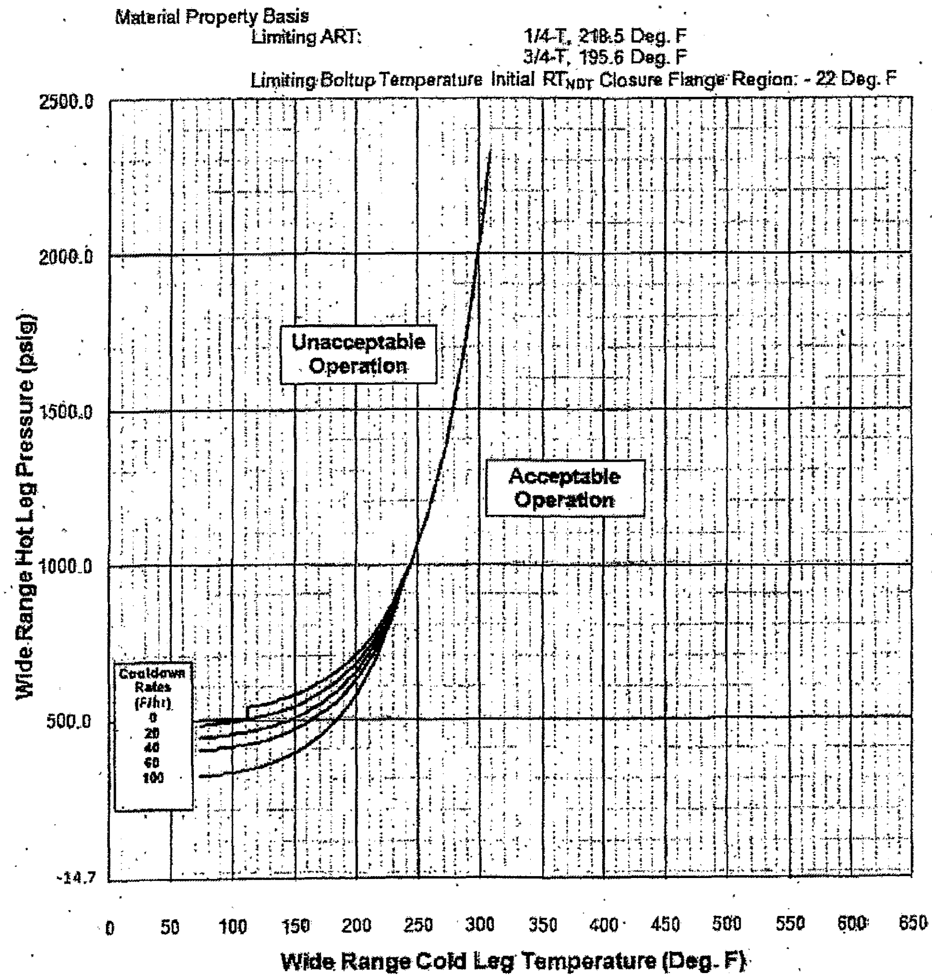

Figure 3.4.3-2 (page 1 of 1)
North Anna Units 1 and 2 Reactor Coolant System Cooldown Limitations
(Cooldown Rates up to 100°F/hr),
Applicable for the first 50.3 EFPY for Unit 1, and 52.3 EFPY for Unit 2
(Including Margins for Instrumentation Errors)

North Anna Units 1 and 2 3.4.3-4 Amendments 275, 257

Figure J-2 Current North Anna Units 1 and 2 Technical Specifications Cooldown P-T Limit Curves

Table J-3 Current North Anna Units 1 and 2 Technical Specifications EOLE Cooldown Curves Data Points (with K_{Ic}, with Flange Requirements, and with Margins for Instrumentation Errors and Pressure Correction)

Steady-State		-20°F/hr		-40°F/hr		-60°F/hr		-100°F/hr	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
73.5	493.90	73.5	479.07	73.5	438.97	73.5	398.04	73.5	313.74
78.5	493.90	78.5	480.80	78.5	440.62	78.5	399.67	78.5	315.36
83.5	493.90	83.5	482.69	83.5	442.50	83.5	401.54	83.5	317.28
88.5	493.90	88.5	484.81	88.5	444.62	88.5	403.68	88.5	319.50
93.5	493.90	93.5	487.16	93.5	446.99	93.5	406.10	93.5	322.05
98.5	493.90	98.5	489.78	98.5	449.66	98.5	408.83	98.5	324.98
103.5	493.90	103.5	492.69	103.5	452.65	103.5	411.91	103.5	328.32
108.5	493.90	108.5	493.90	108.5	455.99	108.5	415.38	108.5	332.12
111.5	493.90	111.5	493.90	113.5	459.72	113.5	419.27	113.5	336.42
111.5	537.33	111.5	498.10	118.5	463.89	118.5	423.63	118.5	341.29
113.5	538.73	113.5	499.54	123.5	468.53	123.5	428.51	123.5	346.78
118.5	542.59	118.5	503.55	128.5	473.71	128.5	433.98	128.5	352.95
123.5	546.86	123.5	507.98	133.5	479.47	133.5	440.07	133.5	359.88
128.5	551.58	128.5	512.92	138.5	485.89	138.5	446.88	138.5	367.66
133.5	556.79	133.5	518.38	143.5	493.01	143.5	454.46	143.5	376.37
138.5	562.55	138.5	524.45	148.5	500.93	148.5	462.92	148.5	386.11
143.5	568.92	143.5	531.16	153.5	509.72	153.5	472.32	153.5	396.99
148.5	575.96	148.5	538.62	158.5	519.49	158.5	482.79	158.5	409.14
153.5	583.74	153.5	546.87	163.5	530.32	163.5	494.42	163.5	422.68
158.5	592.34	158.5	556.03	168.5	542.35	168.5	507.35	168.5	437.79
163.5	601.84	163.5	566.15	173.5	555.67	173.5	521.70	173.5	454.60
168.5	612.34	168.5	577.37	178.5	570.46	178.5	537.65	178.5	473.31
173.5	623.95	173.5	589.78	183.5	586.84	183.5	555.34	183.5	494.12
178.5	636.78	178.5	603.54	188.5	605.01	188.5	574.98	188.5	517.27
183.5	650.95	183.5	618.75	193.5	625.12	193.5	596.75	193.5	542.98
188.5	666.62	188.5	635.60	198.5	647.42	198.5	620.90	198.5	571.55
193.5	683.93	193.5	654.23	203.5	672.09	203.5	647.67	203.5	603.26
198.5	703.07	198.5	674.87	208.5	699.44	208.5	677.34	208.5	638.48
203.5	724.21	203.5	697.67	213.5	729.70	213.5	710.22	213.5	677.55

Table J-3 Current North Anna Units 1 and 2 Technical Specifications EOLE Cooldown Curves Data Points (with K_{Ic}, with Flange Requirements, and with Margins for Instrumentation Errors and Pressure Correction)

Steady-State		-20°F/hr		-40°F/hr		-60°F/hr		-100°F/hr	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
208.5	747.58	208.5	722.93	218.5	763.22	218.5	746.66	218.5	720.91
213.5	773.41	213.5	750.85	223.5	800.31	223.5	787.01	223.5	769.00
218.5	801.96	218.5	781.75	228.5	841.38	228.5	831.72	228.5	822.34
223.5	833.51	223.5	815.92	233.5	886.81	233.5	881.23	233.5	881.46
228.5	868.37	228.5	853.72	238.5	937.12	238.5	936.06	238.5	947.01
233.5	906.90	233.5	895.51	243.5	992.76	243.5	996.55	243.5	996.55
238.5	949.49	238.5	941.76	248.5	1048.57	248.5	1048.57	248.5	1048.57
243.5	996.55	243.5	992.88	253.5	1106.05	253.5	1106.05	253.5	1106.05
248.5	1048.57	248.5	1048.57	258.5	1169.58	258.5	1169.58	258.5	1169.58
253.5	1106.05	253.5	1106.05	263.5	1239.79	263.5	1239.79	263.5	1239.79
258.5	1169.58	258.5	1169.58	268.5	1317.38	268.5	1317.38	268.5	1317.38
263.5	1239.79	263.5	1239.79	273.5	1403.14	273.5	1403.14	273.5	1403.14
268.5	1317.38	268.5	1317.38	278.5	1497.91	278.5	1497.91	278.5	1497.91
273.5	1403.14	273.5	1403.14	283.5	1602.66	283.5	1602.66	283.5	1602.66
278.5	1497.91	278.5	1497.91	288.5	1718.41	288.5	1718.41	288.5	1718.41
283.5	1602.66	283.5	1602.66	293.5	1846.35	293.5	1846.35	293.5	1846.35
288.5	1718.41	288.5	1718.41	298.5	1987.73	298.5	1987.73	298.5	1987.73
293.5	1846.35	293.5	1846.35	303.5	2143.99	303.5	2143.99	303.5	2143.99
298.5	1987.73	298.5	1987.73	308.5	2316.68	308.5	2316.68	308.5	2316.68
303.5	2143.99	303.5	2143.99	-	-	-	-	-	-
308.5	2316.68	308.5	2316.68	-	-	-	-	-	-

APPENDIX K JUSTIFICATION FOR THE USE OF PWROG-17090-NP-A

PWROG-17090-NP-A (Reference K-1) was approved to provide generic values of the unirradiated Charpy Upper-Shelf Energy (USE) for American Society of Mechanical Engineers (ASME) SA508, Class 2 (or the corresponding American Society for Testing and Materials [ASTM] A508, Class 2) RV forgings that were fabricated by the Rotterdam Dockyard Company (Rotterdam) as well as generic values of unirradiated Charpy USE, weight percentage copper (Cu) content, and weight percentage nickel (Ni) content for RV Submerged Arc Welds (SAWs) and Shielded Metal Arc Welds (SMAWs).

In the NRC's Safety Evaluation (Reference K-2) it is stipulated that plants citing the report must ensure that their reactor vessel materials meet the criteria set forth below.

The generic properties provided in the TR are for implementation as conservative generic estimates for the material classes identified below only if no measured values of unirradiated Charpy USE, Cu content, and/or Ni content are available for the specific RV material under consideration. PWR plants that implement these generic estimates must identify their RV materials as follows:

- *A PWR plant with a Rotterdam RV proposing to use the generic unirradiated Charpy USE value of 56 ft-lbs. for its RV forging(s) must identify that its forging(s) are of the SA508, Class 2 or A508, Class 2 specification and that the forging(s) were supplied by Rheinstahl Huttenwerke AG.*
- *A PWR plant with a Rotterdam RV proposing to use the generic unirradiated Charpy USE value 52 ft-lbs. for its RV forging(s) must identify that its forging(s) are of the SA508, Class 2 or A508, Class 2 specification. This generic unirradiated Charpy USE value may be used if the Rotterdam RV forging supplier is identified as Fried-Krupp Huttenwerke AG or if the forging supplier is unknown.*
- *A PWR plant with a Rotterdam RV proposing to use the generic unirradiated Charpy USE value of 75 ft-lbs. for its RV weld(s) must identify that the weld(s) are of the SAW type, that the SAWs are <u>not</u> of Linde 80 flux type, and that its SAW(s) were fabricated by Rotterdam.*
- *A PWR plant with a Rotterdam RV proposing to use the generic Cu content of 0.23 percent and generic Ni content of 0.56 percent for its RV weld(s) must identify that the weld(s) are of the SAW type, that the SAWs are <u>not</u> of Linde 80 flux type, and that its SAW(s) were fabricated by Rotterdam.*
- *A PWR plant with a Rotterdam RV proposing to use the generic unirradiated Charpy USE value of 72 ft-lbs. for its RV weld(s) must identify that the weld(s) were fabricated by Rotterdam. This generic unirradiated Charpy USE value may be used if the Rotterdam RV weld is identified as a SMAW or if the Rotterdam RV weld type is unknown.*
- *A PWR plant with a Rotterdam RV proposing to use the RG 1.99, Rev. 2, default Cu content of 0.35 percent and generic Ni content of 1.13 percent for its RV weld(s) must identify that the weld(s) were fabricated by Rotterdam. These values may be used if the Rotterdam RV weld is identified as a SMAW or if the Rotterdam RV weld type is unknown.*

Table K-1 below demonstrates how the relevant North Anna Units 1 and 2 materials meet these stipulations; thus, justifying the use of the listed generic values from PWROG-17090-NP-A for these materials.

Table K-1 North Anna Units 1 and 2 Reactor Vessel Materials which Use PWROG-17090-NP-A Generic Values

Material Description	Heat	Flux Type	Wt. % Cu	Wt. % Ni	Initial USE (ft-lbs)	Justification
Unit 1						
Inlet/Outlet Nozzle to Upper Shell Welds	Unknown		0.35	1.13	72	The inlet/outlet nozzle to upper shell welds were fabricated by Rotterdam with an unknown weld type.
Inlet Nozzle Forging 11	990268-21	-	-	-	56	Forging was supplied by Rheinstahl Huttenwerke AG and is ASTM A508, Class 2 material.
Unit 2						
Upper to Intermediate Shell Circumferential Weld (6% ID)	801	SMIT 89	-	-	75	The weld was fabricated by Rotterdam with a SAW weld type without Linde 80 flux.
Inlet/Outlet Nozzle to Upper Shell Welds	8816 20459 27622	LW320	0.23	0.56	75	The inlet/outlet nozzle to upper shell welds were fabricated by Rotterdam with a SAW weld type without Linde 80 flux.
Inlet Nozzle Forging 09	990426	-	-	-	56	Forgings were supplied by Rheinstahl Huttenwerke AG and are ASTM A508, Class 2 material.
Outlet Nozzle Forging 13	990426-31	-	-	-	56	

K.1 REFERENCES

K-1. PWROG Report PWROG-17090-NP-A, Revision 0, "Generic Rotterdam Forging and Weld Initial Upper Shelf Energy Determination," January 2020. *[ADAMS Accession Number ML20024E238]*

K-2. NRC Safety Evaluation, "Final Safety Evaluation by the Office of Nuclear Reactor Regulation for the Pressurized Water Reactor Owners Group Topical Report PWROG-17090-NP, Revision 0, 'Generic Rotterdam Forging and Weld Initial Upper-Shelf Energy Determination' (EPID L-2018-TOP-0017)," December 12, 2019. *[ADAMS Accession Numbers ML19345F015 and ML19345F137]*