Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402

CNL-17-134

October 13, 2017

10 CFR 50.4
10 CFR 50 Appendix H

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Watts Bar Nuclear Plant, Unit 2
Facility Operating License No. NPF-96
NRC Docket No. 50-391

Subject: **Transmittal of WCAP-18191-NP, "Watts Bar Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and Supplemental Reactor Vessel Integrity Evaluations"**

Reference: TVA letter to NRC, CNL-17-112, "Revision to Watts Bar Nuclear Plant, Unit 2, Reactor Vessel Surveillance Capsule Withdrawal Schedule," dated September 5, 2017 (ML17248A420)

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," Paragraph III.B.3, in the referenced letter, the Tennessee Valley Authority (TVA) requested Nuclear Regulatory Commission (NRC) approval of a revision to the reactor vessel surveillance capsule removal schedule for Watts Bar Nuclear Plant (WBN) Unit 2. Specifically, TVA requested NRC approval to revise the withdrawal schedule for capsule U from the first WBN Unit 2 refueling outage (U2R1) to the second WBN Unit 2 refueling outage (U2R2, Spring 2019).

Enclosure 1 to the referenced letter contained a reference to the Westinghouse non-proprietary report, WCAP-18191-NP, Revision 0, "Watts Bar Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and Supplemental Reactor Vessel Integrity Evaluations," dated May 2017. In order to assist with the NRC review of the referenced letter, TVA is submitting WCAP-18191-NP, Revision 0.

There are no new regulatory commitments associated with this submittal. Please address any questions regarding this letter to Ed Schrull at 423-751-3850.

Respectfully,

for

J. W. Shea
Vice President, Nuclear Regulatory Affairs and Support Services

Enclosure

cc: See Page 2

Enclosure: WCAP-18191-NP, Revision 0

cc (Enclosure):
NRC Regional Administrator - Region II
NRC Senior Resident Inspector - Watts Bar Nuclear Plant
NRC Project Manager – Watts Bar Nuclear Plant

Enclosure

WCAP-18191-NP, Revision 0

WCAP-18191-NP
Revision 0

May 2017

Watts Bar Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and Supplemental Reactor Vessel Integrity Evaluations

Westinghouse

WCAP-18191-NP
Revision 0

Watts Bar Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and Supplemental Reactor Vessel Integrity Evaluations

Benjamin E. Mays*
Materials Center of Excellence

Louis W. Turicik*
Reactor Internals Aging Management

Craig A. Chelednik*
Nuclear Operations & Radiation Analysis

May 2017

Reviewers: J. Brian Hall*
Elaine M. Ruminski*
Materials Center of Excellence

Anees Udyawar*
Alexandria M. Carolan*
Piping Analysis and Fracture Mechanics

Jianwei Chen*
Nuclear Operations & Radiation Analysis

Approved: David B. Love*, Manager
Materials Center of Excellence

Benjamin A. Leber*, Manager
Piping Analysis and Fracture Mechanics

Laurent P. Houssay*, Manager
Nuclear Operations & Radiation Analysis

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

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

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RECORD OF REVISION

Revision 0: Original Issue

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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 Watts Bar Unit 2 reactor vessel. The analyses herein consider implementation of Tritium-Producing Burnable Absorber Rods (TPBARs) at the beginning of Cycle 4. The heatup and cooldown P-T limit curves were generated using the limiting Adjusted Reference Temperature (ART) values for Watts Bar Unit 2. The limiting ART values were those of Intermediate Shell Forging 05 (Position 1.1) at both 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations.

The P-T limit curves were generated for 32 effective full-power years (EFPY) using the K_{Ic} methodology detailed in the 1998 through the 2000 Addenda Edition 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 60 and 100°F/hr, and cooldown rates of 0 (steady-state), 20, 40, 60, and 100°F/hr were used to generate the P-T limit curves, with the flange requirements and without margins for instrumentation errors. The Watts Bar Unit 2 End of License (EOL) corresponding to 40 years of operation is 32 EFPY. The EOL P-T limit curves can be found in Figures 8-1 and 8-2.

Appendix A contains the thermal stress intensity factors for the maximum heatup and cooldown rates at 32 EFPY.

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 material (Intermediate Shell Forging 05) bound the P-T limit curves for the reactor vessel inlet and outlet nozzles for Watts Bar Unit 2 at 32 EFPY.

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

Appendix D contains an upper-shelf energy (USE) evaluation for all Watts Bar Unit 2 reactor vessel beltline and extended beltline materials. Per Appendix D, all beltline and extended beltline materials are projected to maintain USE values above the 50 ft-lb screening criterion per 10 CFR 50 Appendix G at 32 EFPY.

Appendix E contains a pressurized thermal shock (PTS) evaluation for all Watts Bar Unit 2 reactor vessel beltline and extended beltline materials. Per Appendix E, all beltline and extended beltline materials have projected RT_{PTS} values below the screening criteria set forth in 10 CFR 50.61. Additionally, Watts Bar Unit 2 will remain in Category I of the Emergency Response Guidelines through 32 EFPY.

Appendix F contains an updated surveillance capsule withdrawal schedule. Per Appendix F, three surveillance capsules are recommended to be withdrawn from the Watts Bar Unit 2 reactor before end of license.

1 INTRODUCTION

The purpose of this report is to present the calculations and the development of the Watts Bar Unit 2 heatup and cooldown P-T limit curves for 32 EFPY. This report documents the calculated Adjusted Reference Temperature (ART) values and the development of the P-T limit curves for normal operation. The analyses herein consider implementation of Tritium-Producing Burnable Absorber Rods (TPBARs) at the beginning of Cycle 4.

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

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 steels. The U.S. Nuclear Regulatory Commission (NRC) has published a method for predicting radiation embrittlement in Regulatory Guide 1.99, Revision 2 [Ref. 1]. Regulatory Guide 1.99, Revision 2 is used for the calculation of ART values ($RT_{NDT(U)} + \Delta RT_{NDT}$ + 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 32 EFPY are documented in Section 7 of this report. The fluence projections used in calculation of the ART values are provided in Section 2 of this report

The heatup and cooldown P-T limit curves documented in this report were generated using the most limiting ART values (plus an additional margin to account for future perturbations such as an uprate or surveillance capsule results) and the NRC-approved methodology documented in WCAP-14040-A, Revision 4 [Ref. 2], which allows use of ASME Code Cases N-641 and N-640. Specifically, the K_{Ic} methodology of the 1998 through the 2000 Addenda Edition of ASME Code, Section XI, Appendix G [Ref. 3] was used. 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 herein were generated without instrumentation errors. The reactor vessel flange requirements of 10 CFR 50, Appendix G [Ref. 4] have been incorporated in the P-T limit curves. The P-T limit curves generated in Section 8 bound the P-T limit curves for the reactor vessel inlet and outlet nozzles generated in Appendix B for Watts Bar Unit 2 at 32 EFPY. Discussion of the other ferritic RCPB components relative to P-T limits is contained in Appendix C.

Appendices D, E, and F provide supplemental reactor vessel integrity analyses including a USE evaluation (Appendix D), a PTS and Emergency Response Guideline (ERG) analysis (Appendix E), and a revised surveillance capsule withdrawal schedule (Appendix F).

2 CALCULATED NEUTRON FLUENCE

2.1 INTRODUCTION

A discrete ordinates (S_n) transport analysis was performed for the Watts Bar Unit 2 reactor to determine the neutron radiation environment within the beltline and extended beltline regions 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 section were based on nuclear cross-section data derived from the Evaluated Nuclear Data File (ENDF) database (specifically, ENDF/B-VI). Furthermore, these neutron transport methodologies adhere to the guidelines and meet the requirements of the U.S. Nuclear Regulatory Commission (USNRC) Regulatory Guide 1.190 [Ref. 5]. Additionally, the methods used to determine the pressure vessel neutron exposure are consistent with the USNRC approved methodology described in WCAP-14040-A [Ref. 2].

2.2 DISCRETE ORDINATES ANALYSIS

In performing the fast neutron exposure evaluations for the Watts Bar Unit 2 reactor vessel, plant-specific forward transport calculations were carried out using the following three-dimensional flux synthesis technique:

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

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.

For the transport calculations, the [r,θ] models depicted in Figure 2-1 through Figure 2-3 were utilized since, with the exception of the neutron pads, the reactor is octant symmetric. These [r,θ] models included representations of the core, the reactor internals, the neutron pads, the pressure vessel cladding and vessel wall, the insulation external to the pressure vessel, and the primary biological shield wall. The difference among the models is the azimuthal extent of the neutron pad and the configuration of surveillance capsules on those pads. The reactor model including a neutron pad segment without surveillance capsules and associated structures is shown in Figure 2-1. The geometric model of the neutron pad segment with a single surveillance capsule holder attached is shown in Figure 2-2, while a neutron pad segment with a dual surveillance capsule holder attached is shown in Figure 2-3. In developing these analytical models, nominal design dimensions were employed for the various structural components. Likewise, water temperatures, and hence, coolant densities in the reactor core 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. 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 170 radial by 98 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 Watts Bar Unit 2 calculations is shown in Figure 2-4 and extends radially from the centerline of the reactor core out to a location interior to the primary biological shield. Axially, the model extends from an elevation of about 1.6 feet below the bottom head ring to bottom peel circumferential weld (lowest analyzed point) to the centerline of the outlet and inlet nozzles. Therefore, the core, core baffle, former plates, core barrel, neutron pad, downcomer water, pressure vessel, cladding, inlet plenum, lower core support plates, lower support columns, lower core plates, nozzle legs, plates, gaps, upper core plates, and outlet plenums are explicitly included in the model. The analyzed locations as measured from the midplane of the active fuel can be seen in Table 2-1. 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 [r,z] geometric mesh description of these reactor models consisted of 153 radial by 158 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 technique consisted of the same 153 radial mesh intervals included in the [r,z] model. Thus, radial synthesis factors could be determined on a meshwise basis throughout the entire geometry.

The core power distributions used in the plant-specific transport analysis for each of the first 7 projected fuel cycles at Watts Bar Unit 2 included cycle-dependent fuel assembly initial enrichments, burnups, and axial power distributions. Note that Watts Bar Unit 2 is currently operating in Cycle 1; therefore, no fuel cycles have been completed. Cycles 1-6 are based on the expected core design for these cycles and Cycle 7 is a representative equilibrium fuel cycle that is then extended to 32 and 40 Effective Full Power Years (EFPY). These calculations take into account implementation of TPBARs at the beginning of Cycle 4. The TPBAR core design was reflected in the radial power distributions that are used as critical inputs into the reactor vessel fluence calculation. In general, the TPBAR core designs show a higher relative power in the peripheral assemblies than that of a low leakage core design, which causes an increase in the reactor pressure vessel fast neutron fluence exposure in those TPBAR cycles. The increases can be seen in Table 2-4. The reactor vessel integrity evaluations herein consider the TPBAR core design and justify safe operation through 32 Effective Full Power Years with respect to reactor vessel integrity. This information was used to develop spatial- and energy-dependent core source distributions averaged over each individual fuel cycle. Therefore, the results from the neutron transport calculations provided data in terms of fuel cycle-averaged neutron fluence rate, which, when multiplied by the appropriate fuel cycle length, generated the incremental fast neutron exposure for each fuel cycle. In constructing these core source distributions, the energy distribution of the source was based on an appropriate fission split for uranium and plutonium isotopes based on the initial enrichment and burnup history of individual fuel assemblies. From these assembly-dependent fission splits, composite values of energy release per fission, neutron yield per fission, and fission spectrum were determined.

All of the transport calculations were carried out using the DORT discrete ordinates code [Ref. 7] coupled with the BUGLE-96 cross-section library [Ref. 8]. The BUGLE-96 library provides a 67-group coupled

neutron-gamma ray 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 data tabulations include both plant- and fuel-cycle-specific calculated neutron exposures at the end of Cycles (EOC) 1 through 7 and at further projections to 32 and 40 EFPY. The calculations account for the expected core power of 3411 MWt in Cycle 1 and 3459 MWt in Cycles 2-7. The projections after Cycle 7 are based on the assumption that the core power distributions and associated plant operating characteristics from Cycle 7 are representative of future plant operations.

Selected results from the neutron transport analyses are provided in Table 2-2 through Table 2-10. The calculated fast neutron exposure rate and total exposure values at the geometric center of the surveillance capsule test specimen are provided in Table 2-2 and Table 2-3, respectively. Likewise, Table 2-4 through 2-6 show the maximum calculated exposure and exposure rate at the clad/base metal interface. Azimuthally, angles 0.00, 20.00 through 22.00 at 0.25 degrees intervals, 23.00, 30.00, and 45.00 were analyzed. For Cycles 1-6, Table 2-5 and Table 2-6 show neutron exposure at the azimuthal angle of 45 degrees and a radial distance of 220.11 cm from the pressure vessel centerline, which is at the clad/base metal interface, because this resulted in the maximum neutron fluence and dpa. For Cycle 7 and on, Table 2-5 and Table 2-6 show neutron exposure at the azimuthal angle of 22 degrees and a radial distance of 220.11 cm from the pressure vessel centerline, which is at the clad/base metal interface, because this resulted in the maximum neutron fluence and dpa. Table 2-7 through Table 2-10 show the maximum neutron exposure at the pressure vessel clad/base metal interface at various axial points of interest. The data tabulation includes the projected plant-specific calculated fluence at the end of Cycles 1 through 7 and projections for an operating time extending to 32 and 40 EFPY.

2.3 CALCULATIONAL UNCERTAINTIES

An uncertainty analysis that includes comparisons of calculations with test and power reactor benchmarks and an analytical uncertainty study has been completed and documented in an NRC-approved topical report [Ref. 2 and 6]. The overall uncertainty in the transport calculations was demonstrated to be 13% (1σ). This level of uncertainty meets the 20% (1σ) requirement of Regulatory Guide 1.190.

Table 2-1 Pressure Vessel Material Locations

Location	Distance from Midplane of Active Fuel (cm)
Flange Mating Surface	556.823
Centerline of Inlet and Outlet Nozzles (Upper extent of r,z model)	343.463
Lowest Extent of Outlet Nozzle to Shell Weld	276.263
Lowest Extent of Inlet Nozzle to Shell Weld	265.263
Center of Intermediate Shell to Nozzle Shell Circumferential Weld	226.423
Bottom of Upper Core Plate	216.463
Top of Active Fuel	182.880
Center of Lower Shell A to Intermediate Shell Circumferential Weld	12.023
Midplane of Active Fuel	0.000
Bottom of Active Fuel	-182.880
Center of Lower Shell A to Lower Shell B Circumferential Weld	-201.677
Top of Lower Core Plate	-191.206
Center of Lower Ring to Lower Shell B Circumferential Weld	-314.777
Lowest extent of r,z model	-363.296
Lower Head to Lower Ring Circumferential Weld	-401.977

Table 2-2 Calculated Fast Neutron Exposure Rate at the Geometric Center of the Surveillance Capsules at Core Midplane

Cycle	Cumulative Operating Time (EFPY)	Neutron Flux (E > 1.0 MeV) [n/cm^2-s]			Iron Atom Displacement Rate [dpa/s]		
		31.5 Degrees Dual	34.0 Degrees Dual	34.0 Degrees Single	31.5 Degrees Dual	34.0 Degrees Dual	34.0 Degrees Single
1	**1.27**	9.077E+10	1.075E+11	1.091E+11	1.787E-10	2.150E-10	2.226E-10
2	**2.61**	6.987E+10	8.059E+10	8.174E+10	1.360E-10	1.593E-10	1.648E-10
3	**4.14**	5.622E+10	6.520E+10	6.614E+10	1.093E-10	1.289E-10	1.333E-10
4	**5.53**	7.263E+10	8.412E+10	8.534E+10	1.417E-10	1.667E-10	1.725E-10
5	**6.91**	8.054E+10	9.590E+10	9.734E+10	1.578E-10	1.909E-10	1.976E-10
6	**8.30**	8.107E+10	9.563E+10	9.705E+10	1.588E-10	1.903E-10	1.970E-10
7	**9.69**	7.870E+10	9.059E+10	9.191E+10	1.541E-10	1.802E-10	1.864E-10
Future	**32.00**	7.870E+10	9.059E+10	9.191E+10	1.541E-10	1.802E-10	1.864E-10
Future	**40.00**	7.870E+10	9.059E+10	9.191E+10	1.541E-10	1.802E-10	1.864E-10
Surveillance Capsule Center = 207.32 cm							

Table 2-3 Calculated Fast Neutron Exposure at the Geometric Center of the Surveillance Capsules at Core Midplane

Cycle	Cumulative Operating Time (EFPY)	Neutron Fluence (E > 1.0 MeV) [n/cm^2]			Iron Atom Displacement [dpa]		
		31.5 Degrees Dual	34.0 Degrees Dual	34.0 Degrees Single	31.5 Degrees Dual	34.0 Degrees Dual	34.0 Degrees Single
1	**1.27**	3.636E+18	4.307E+18	4.370E+18	7.158E-03	8.615E-03	8.917E-03
2	**2.61**	6.590E+18	7.714E+18	7.826E+18	1.291E-02	1.535E-02	1.589E-02
3	**4.14**	9.304E+18	1.086E+19	1.102E+19	1.819E-02	2.157E-02	2.232E-02
4	**5.53**	1.248E+19	1.454E+19	1.475E+19	2.438E-02	2.887E-02	2.987E-02
5	**6.91**	1.600E+19	1.873E+19	1.901E+19	3.128E-02	3.721E-02	3.851E-02
6	**8.30**	1.955E+19	2.291E+19	2.325E+19	3.822E-02	4.553E-02	4.712E-02
7	**9.69**	2.301E+19	2.690E+19	2.729E+19	4.500E-02	5.345E-02	5.532E-02
Future	**32.00**	7.842E+19	9.069E+19	9.200E+19	1.535E-01	1.803E-01	1.866E-01
Future	**40.00**	9.829E+19	1.136E+20	1.152E+20	1.924E-01	2.258E-01	2.336E-01
Surveillance Capsule Center = 207.32 cm							

Table 2-4 Calculated Maximum Fast Neutron Exposure Rate at the Pressure Vessel Clad/Base Metal Interface

Cycle	Cumulative Operating Time (EFPY)	Neutron Flux (E > 1.0 MeV) [n/cm^2-s]	Iron Atom Displacement Rate [dpa/s]
1	**1.27**	2.193E+10	3.466E-11
2	**2.61**	1.614E+10	2.547E-11
3	**4.14**	1.351E+10	2.132E-11
4	**5.53**	1.687E+10	2.663E-11
5	**6.91**	2.004E+10	3.162E-11
6	**8.30**	1.962E+10	3.098E-11
7	**9.69**	1.946E+10	2.974E-11
Future	**32.00**	1.946E+10	2.974E-11
Future	**40.00**	1.946E+10	2.974E-11

Table 2-5 Calculated Maximum Fast Neutron Exposure at the Pressure Vessel Clad/Base Metal Interface (Neutron Fluence)

Cycle	Cumulative Operating Time (EFPY)	Neutron Fluence (E > 1.0 MeV) [n/cm^2]					
		0.0 Degrees	22.0 Degrees	23.0 Degrees	30.0 Degrees	45.0 Degrees	Max.
1	**1.27**	4.513E+17	8.262E+17	8.250E+17	7.019E+17	8.787E+17	8.787E+17
2	**2.61**	8.036E+17	1.467E+18	1.470E+18	1.277E+18	1.560E+18	1.560E+18
3	**4.14**	1.191E+18	2.074E+18	2.077E+18	1.811E+18	2.212E+18	2.212E+18
4	**5.53**	1.583E+18	2.797E+18	2.800E+18	2.433E+18	2.950E+18	2.950E+18
5	**6.91**	1.968E+18	3.560E+18	3.565E+18	3.108E+18	3.826E+18	3.826E+18
6	**8.30**	2.411E+18	4.354E+18	4.359E+18	3.793E+18	4.684E+18	4.684E+18
7	**9.69**	2.894E+18	5.210E+18	5.210E+18	4.484E+18	5.478E+18	5.478E+18
Future	**32.00**	1.064E+19	1.891E+19	1.882E+19	1.555E+19	1.821E+19	1.891E+19
Future	**40.00**	1.341E+19	2.382E+19	2.370E+19	1.952E+19	2.277E+19	2.382E+19
Base Metal Inner Radius = 220.11 cm							

Table 2-6 Calculated Maximum Fast Neutron Exposure at the Pressure Vessel Clad/Base Metal Interface (Iron Displacement)

Cycle	Cumulative Operating Time (EFPY)	Iron Atom Displacement [dpa]					
		0.0 Degrees	22.0 Degrees	23.0 Degrees	30.0 Degrees	45.0 Degrees	Max.
1	**1.27**	6.990E-04	1.262E-03	1.261E-03	1.092E-03	1.388E-03	1.388E-03
2	**2.61**	1.246E-03	2.244E-03	2.249E-03	1.987E-03	2.464E-03	2.464E-03
3	**4.14**	1.846E-03	3.174E-03	3.180E-03	2.818E-03	3.493E-03	3.493E-03
4	**5.53**	2.455E-03	4.281E-03	4.287E-03	3.785E-03	4.659E-03	4.659E-03
5	**6.91**	3.052E-03	5.448E-03	5.458E-03	4.835E-03	6.041E-03	6.041E-03
6	**8.30**	3.739E-03	6.665E-03	6.674E-03	5.901E-03	7.395E-03	7.395E-03
7	**9.69**	4.488E-03	7.973E-03	7.974E-03	6.976E-03	8.651E-03	8.651E-03
Future	**32.00**	1.649E-02	2.891E-02	2.879E-02	2.418E-02	2.876E-02	2.891E-02
Future	**40.00**	2.080E-02	3.642E-02	3.625E-02	3.036E-02	3.597E-02	3.642E-02
Base Metal Inner Radius = 220.11 cm							

Table 2-7 Calculated Maximum Fast Neutron Exposure at the Pressure Vessel Circumferential Welds (Neutron Fluence)

Cycle	Cumulative Operating Time (EFPY)	Neutron Fluence (E > 1.0 MeV) [n/cm^2]			
		Bottom Head Ring to Bottom Peel Circ. Weld	Lower Shell to Bottom Head Ring Circ. Weld	Intermediate to Lower Shell Circ. Weld	Upper to Intermediate Shell Circ. Weld
1	**1.27**	4.197E+13	1.195E+17	8.708E+17	5.241E+16
2	**2.61**	8.044E+13	2.168E+17	1.545E+18	9.055E+16
3	**4.14**	1.211E+14	3.145E+17	2.191E+18	1.328E+17
4	**5.53**	1.637E+14	4.250E+17	2.921E+18	1.766E+17
5	**6.91**	2.107E+14	5.507E+17	3.785E+18	2.253E+17
6	**8.30**	2.559E+14	6.726E+17	4.632E+18	2.719E+17
7	**9.69**	2.919E+14	7.724E+17	5.413E+18	2.952E+17
Future	**32.00**	8.985E+14	2.454E+18	1.861E+19	1.097E+18
Future	**40.00**	1.121E+15	3.071E+18	2.344E+19	1.385E+18
Bottom Head Ring to Bottom Peel Circ. Weld Elevation in r,z model: z = -314.777 cm, r = 220.11 cm Lower Shell to Bottom Head Ring Circ. Weld Elevation in r,z model: z = -201.677 cm, r = 220.11 cm Intermediate to Lower Shell Circ. Weld Elevation in r,z model: z = 12.023 cm, r = 220.11 cm Upper to Intermediate Shell Circ. Weld Elevation in r,z model: z = 226.423 cm, r = 217.56 cm					

Table 2-8 Calculated Maximum Fast Neutron Exposure at the Pressure Vessel Circumferential Welds (Iron Displacement)

Cycle	Cumulative Operating Time (EFPY)	Iron Atom Displacement [dpa]			
		Bottom Head Ring to Bottom Peel Circ. Weld	Lower Shell to Bottom Head Ring Circ. Weld	Intermediate to Lower Shell Circ. Weld	Upper to Intermediate Shell Circ. Weld
1	**1.27**	1.892E-07	1.900E-04	1.376E-03	8.335E-05
2	**2.61**	3.496E-07	3.448E-04	2.440E-03	1.440E-04
3	**4.14**	5.121E-07	4.999E-04	3.461E-03	2.110E-04
4	**5.53**	6.899E-07	6.754E-04	4.612E-03	2.805E-04
5	**6.91**	8.916E-07	8.752E-04	5.976E-03	3.580E-04
6	**8.30**	1.087E-06	1.069E-03	7.313E-03	4.321E-04
7	**9.69**	1.250E-06	1.228E-03	8.547E-03	5.082E-04
Future	**32.00**	3.874E-06	3.782E-03	2.846E-02	1.728E-03
Future	**40.00**	4.851E-06	4.734E-03	3.584E-02	2.165E-03
Bottom Head Ring to Bottom Peel Circ. Weld Elevation in r,z model: z = -314.777 cm, r = 220.11 cm Lower Shell to Bottom Head Ring Circ. Weld Elevation in r,z model: z = -201.677 cm, r = 220.11 cm Intermediate to Lower Shell Circ. Weld Elevation in r,z model: z = 12.023 cm, r = 220.11 cm Upper to Intermediate Shell Circ. Weld Elevation in r,z model: z = 226.423 cm, r = 217.56 cm					

Table 2-9 Calculated Maximum Fast Neutron Exposure at Pressure Vessel Nozzle Welds

Cycle	Cumulative Operating Time (EFPY)	Neutron Fluence (E > 1.0 MeV) [n/cm^2]		Iron Atom Displacement [dpa]	
		Inlet Nozzle Weld	Outlet Nozzle Weld	Inlet Nozzle Weld	Outlet Nozzle Weld
1	**1.27**	3.764E+15	1.947E+15	6.63E-06	3.571E-06
2	**2.61**	6.668E+15	3.459E+15	1.17E-05	6.322E-06
3	**4.14**	9.924E+15	5.176E+15	1.74E-05	9.409E-06
4	**5.53**	1.334E+16	6.959E+15	2.34E-05	1.265E-05
5	**6.91**	1.667E+16	8.690E+15	2.92E-05	1.581E-05
6	**8.30**	2.004E+16	1.044E+16	3.52E-05	1.902E-05
7	**9.69**	2.407E+16	1.255E+16	4.22E-05	2.286E-05
Future	**32.00**	8.861E+16	4.630E+16	1.55E-04	8.423E-05
Future	**40.00**	1.118E+17	5.841E+16	1.96E-04	1.062E-04
Outlet Nozzle Weld Elevation in r,z model: z = 276.263 cm, θ = 22°, r = 217.56 cm Inlet Nozzle Weld Elevation in r,z model: z = 265.263 cm, θ = 23°, r = 217.56 cm					

Table 2-10 Calculated Maximum Fast Neutron Exposure at Pressure Vessel Shells

Cycle	Cumulative Operating Time (EFPY)	Neutron Fluence (E > 1.0 MeV) [n/cm^2]			Iron Atom Displacement [dpa]		
		Intermediate Shell	Lower Shell	Bottom Head Ring	Intermediate Shell	Lower Shell	Bottom Head Ring
1	**1.27**	8.708E+17	8.787E+17	1.195E+17	1.376E-03	1.388E-03	1.900E-04
2	**2.61**	1.545E+18	1.560E+18	2.168E+17	2.440E-03	2.464E-03	3.448E-04
3	**4.14**	2.191E+18	2.212E+18	3.145E+17	3.461E-03	3.493E-03	4.999E-04
4	**5.53**	2.921E+18	2.950E+18	4.250E+17	4.612E-03	4.659E-03	6.754E-04
5	**6.91**	3.785E+18	3.826E+18	5.507E+17	5.976E-03	6.041E-03	8.752E-04
6	**8.30**	4.632E+18	4.684E+18	6.726E+17	7.313E-03	7.395E-03	1.069E-03
7	**9.69**	5.413E+18	5.478E+18	7.724E+17	8.547E-03	8.651E-03	1.228E-03
Future	**32.00**	1.861E+19	1.891E+19	2.454E+18	2.846E-02	2.891E-02	3.782E-03
Future	**40.00**	2.344E+19	2.382E+19	3.071E+18	3.584E-02	3.642E-02	4.734E-03
Bottom Head Ring Spans: -314.777 cm to -201.677 cm Lower Shell Spans: -201.677 cm to 12.023 cm Intermediate Shell Spans: 12.023 cm to 226.423 cm							

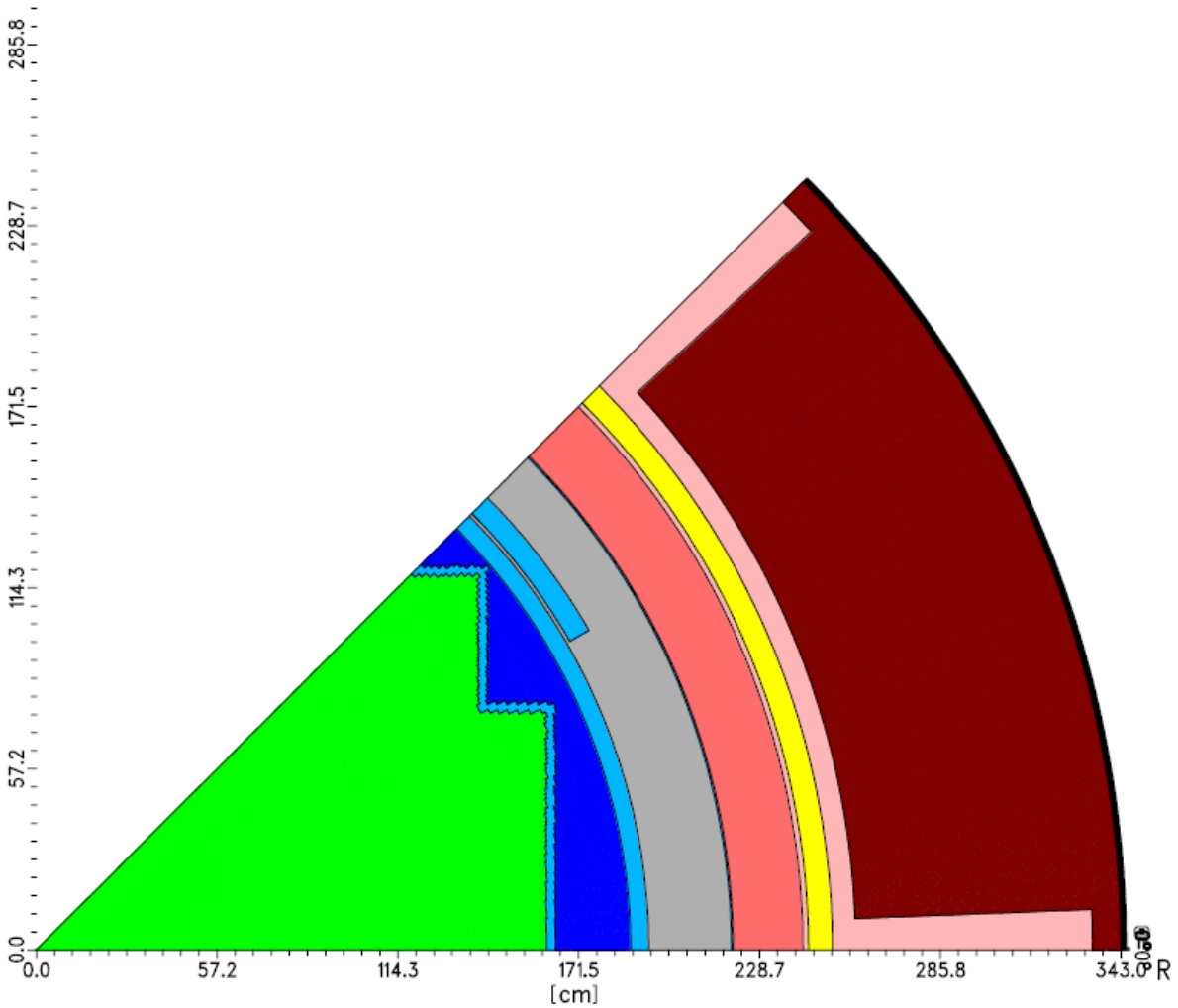

Figure 2-1 Watts Bar Unit 2 Reactor Geometry; [r,θ] Plan View, 15° Neutron Pad Configuration without Surveillance Capsules

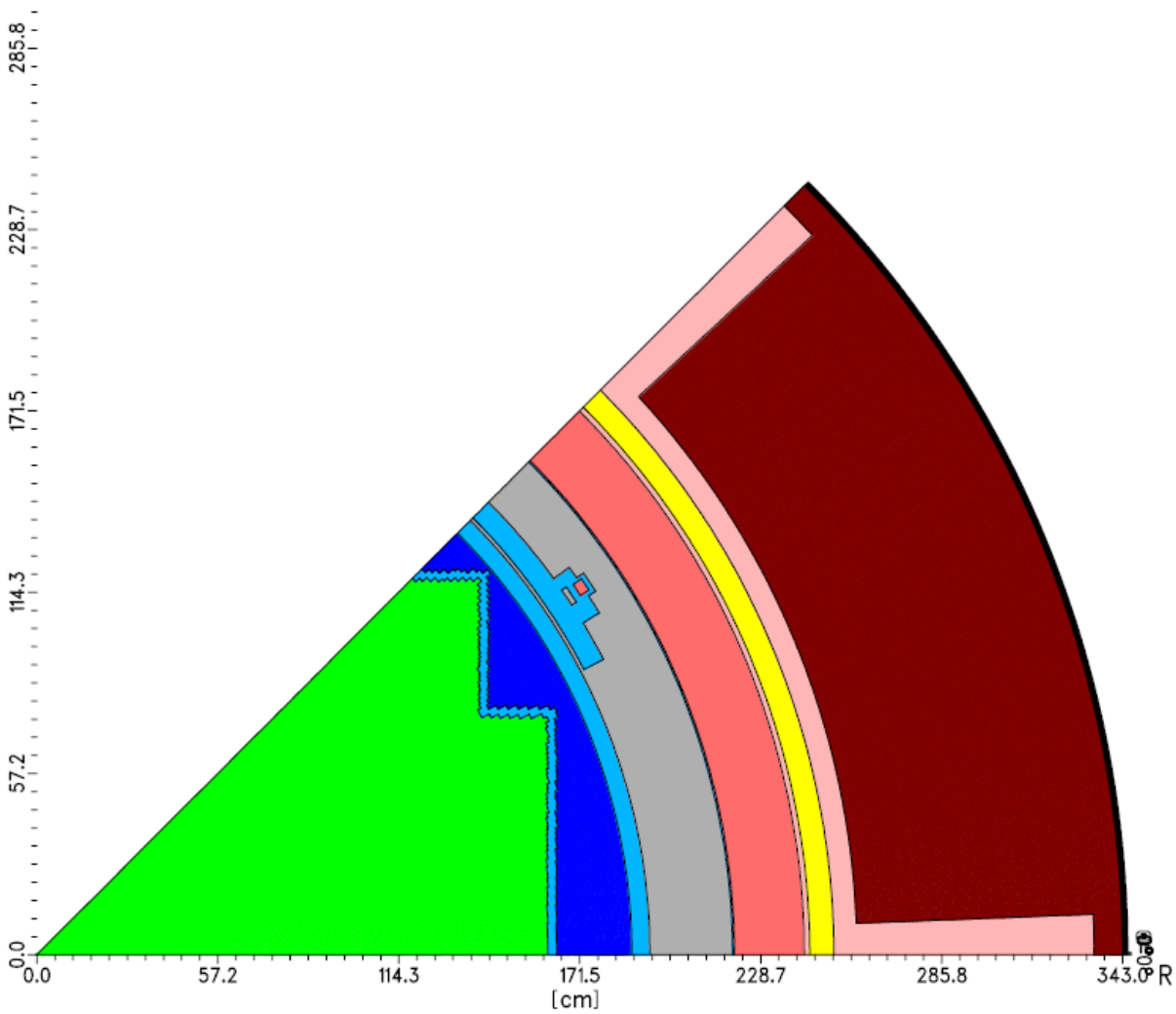

Figure 2-2 Watts Bar Unit 2 Reactor Geometry; [r,θ] Plan View, 17.5° Neutron Pad Configuration with a Single Capsule Holder

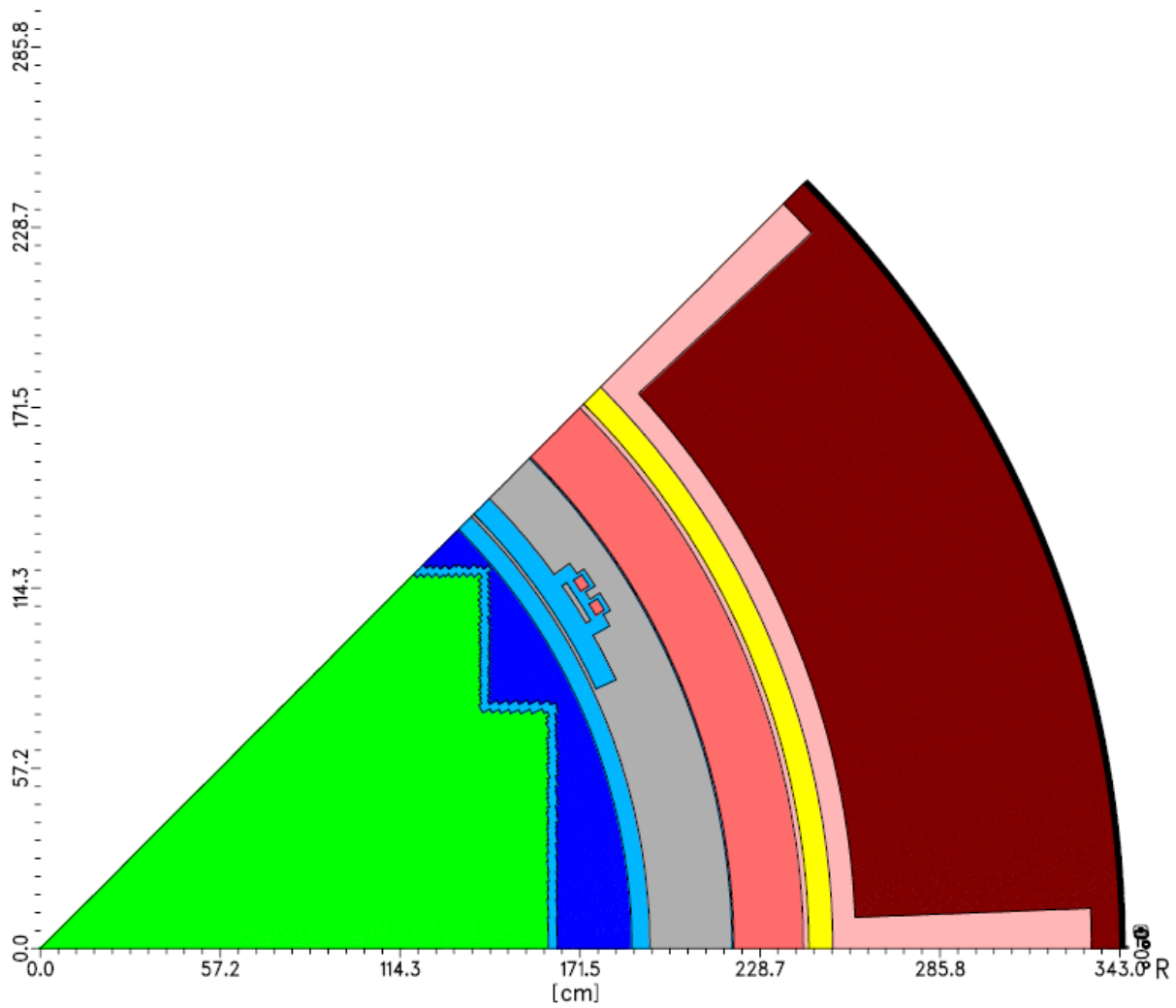

Figure 2-3 Watts Bar Unit 2 Reactor Geometry; [r,θ] Plan View 20° Neutron Pad Configuration with a Dual Capsule Holder

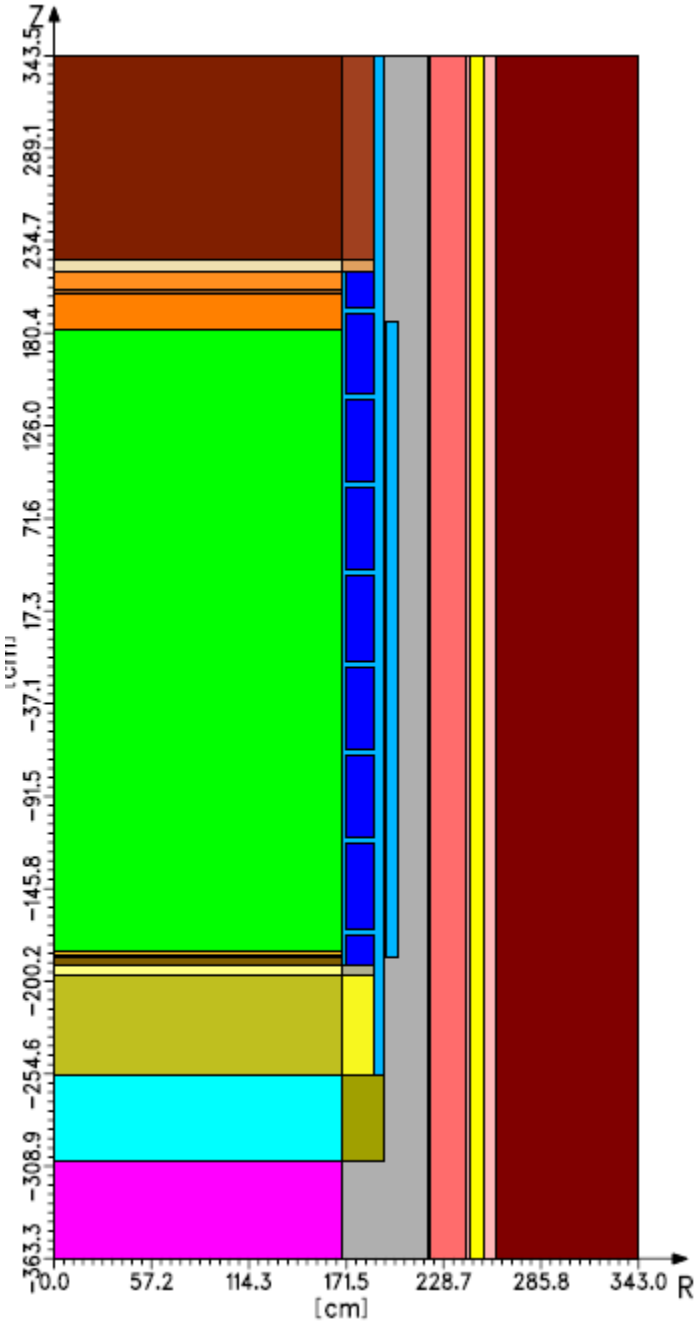

Figure 2-4 Watts Bar Unit 2 Reactor Geometry; [r,z] Section View without Surveillance Capsules

3 FRACTURE TOUGHNESS PROPERTIES

The requirements for P-T limit curve development are specified in 10 CFR 50, Appendix G [Ref. 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."*

The Watts Bar Unit 2 beltline materials traditionally included Intermediate Shell Forging 05, Lower Shell Forging 04, and Intermediate to Lower Shell Circumferential Weld W05; however, as described in NRC Regulatory Issue Summary (RIS) 2014-11 [Ref. 9] and TLR-RES/DE/CIB-2013-01 [Ref. 10], any reactor vessel materials that are predicted to experience a neutron fluence exposure greater than 1.0×10^{17} n/cm^2 (E > 1.0 MeV) at the end of the licensed operating period should be considered in the development of P-T limit curves. The additional materials that exceed this fluence threshold are referred to as extended beltline materials and are evaluated to ensure that the applicable acceptance criteria are met. As seen from Tables 2-7, 2-9, and 2-10 of this report, the extended beltline materials include Upper Shell Forging 06, Bottom Head Ring 03, Upper to Intermediate Shell Circumferential Weld W06, and Lower Shell to Bottom Head Ring Circumferential Weld W04. Note that the Upper to Intermediate Shell Circumferential Weld W06 fluence value is conservatively applied to the Upper Shell Forging 06 herein; this approach is conservative. Although the reactor vessel nozzles are not a part of the extended beltline, per NRC RIS 2014-11, the nozzle materials must be evaluated for their potential effect on P-T limit curves regardless of exposure – See Appendix B for more details.

As part of this P-T limit curve development effort, the initial RT_{NDT} and initial USE values for the Watts Bar Unit 2 reactor vessel beltline and extended beltline base metal materials are required. Since this document represents the first definition of the extended beltline materials for Watts Bar Unit 2, the initial RT_{NDT} and USE values for these materials are determined herein. The initial RT_{NDT} values for each of the extended beltline forgings were determined per BTP 5-3, Paragraph B1.1(3) [Ref. 11] in conjunction with ASME Code, Section III, Subsection NB-2300 [Ref. 12]. These initial RT_{NDT} values were determined using both BTP 5-3 Position 1.1(3)(a) and Position 1.1(3)(b), and the more limiting initial RT_{NDT} value was chosen for each material. The initial USE values were determined in accordance with the methodology described in ASTM E185-82 [Ref. 19]. For each of the extended beltline forgings, use of BTP 5-3 Paragraph 1.2 [Ref. 11] was necessary. A summary of the best-estimate copper (Cu), nickel (Ni), Manganese (Mn), and Phosphorus (P) contents, in units of weight percent (wt. %), as well as initial RT_{NDT} and initial USE values for the reactor vessel beltline and extended beltline materials are provided in Table 3-1 for Watts Bar Unit 2. Table 3-2 contains a summary of the initial RT_{NDT} values of the reactor vessel flange and reactor vessel closure head flange. These values serve as input to the P-T limit curves "flange-notch" per Appendix G of 10 CFR 50 – See Section 6.3 for details.

Note that data from a surveillance program at another plant is often called 'sister-plant' data; this term will be utilized herein.

Table 3-1 Summary of the Best-Estimate Chemistry, Initial RT_{NDT}, and Initial USE Values for the Watts Bar Unit 2 Reactor Vessel Materials

Reactor Vessel Material and Identification Number	Heat Number	Chemical Composition[a]				Fracture Toughness Property	
		Wt. % Cu	Wt. % Ni	Wt. % Mn	Wt. % P	Initial RT_{NDT} (°F)	Initial Upper-Shelf Energy (ft-lb)
Reactor Vessel Beltline Materials							
Intermediate Shell Forging 05	527828	0.05[b]	0.78[b]	0.72	0.012	14[b]	90[b]
Lower Shell Forging 04	528658	0.05[b]	0.81[b]	0.72	0.006	5[b]	105[b]
Intermediate to Lower Shell Circumferential Weld W05	895075	0.05[b]	0.70[b]	1.97	0.010	-50[b]	127[b]
Reactor Vessel Extended Beltline Materials							
Upper Shell Forging 06	411572	0.07	0.91	0.70	0.005	-14[c]	94[c]
Upper to Intermediate Shell Circumferential Weld Seam W06	899680	0.03	0.75	1.97	0.009	10[d]	92[e]
Lower Shell to Bottom Head Ring Circumferential Weld Seam W04	899680	0.03	0.75	1.97	0.009	10[d]	92[e]
Bottom Head Ring 03	5329	0.06	0.86	0.72	0.009	-40[c]	105[c]
Surveillance Materials[f]							
Intermediate Shell Forging 05	527828	0.05	0.78	0.72	0.012	- - -	90
Watts Bar Unit 2 Surveillance Weld	895075[g]	0.033	0.70	1.89	0.013	- - -	144.3
Catawba Unit 1 Surveillance Weld		0.05	0.73	- - -	- - -	- - -	- - -
Watts Bar Unit 1 Surveillance Weld		0.03	0.75	- - -	- - -	- - -	- - -
McGuire Unit 2 Surveillance Weld		0.04	0.74	- - -	- - -	- - -	- - -

Notes (continued on following page):

(a) All chemistry values are obtained from the Watts Bar Unit 2 Certified Material Test Reports (CMTRs), unless otherwise noted. Wt. % Mn and wt. % P values are not necessary for reactor vessel integrity evaluations; these values are provided for future use, if needed.

(b) Chemistry and initial RT_{NDT} values are taken from Table 2-1 of WCAP-17035-NP, Revision 2 [Ref. 13]. Initial USE values are taken from Table B-1 of WCAP-17035-NP, Revision 2.

(c) The extended beltline forging $RT_{NDT(U)}$ and initial USE values were calculated as a part of this evaluation. The $RT_{NDT(U)}$ values are based on drop-weight data, tangentially-oriented Charpy V-notch test data, and NUREG-0800, BTP 5-3 [Ref. 11] Position 1.1(3)(a) and (b), with the more limiting $RT_{NDT(U)}$ value being selected. The initial USE values are the average of all impact energy values with ≥ 95% shear reduced to 65% of their original values to conservatively approximate transverse data per NUREG-0800, BTP 5-3 Position 1.2 methodology.

(d) Value consistent with the initial RT_{NDT} value for weld Heat # 899680 as reported in WCAP-17455-NP [Ref. 14] for McGuire Unit 2 and WCAP-17669-NP, Revision 1 [Ref. 15] for Catawba Unit 1. See Note (e) below.

(e) The CMTRs for weld Heat # 899680 report only three impact energy values at a single test temperature (-12°C or 10.4°F) that did not reach greater than 55% shear. No other information is available for this weld heat. However, weld Heat # 895075 does have USE data and is a Rotterdam weld of the same type (Grau L.O., LW 320). Therefore, in absence of USE data for weld Heat # 899680, the weld Heat # 895075 test results from the first surveillance capsule should be used in accordance with Section B.1.2 of NUREG-0800 Branch Technical Position 5-3 [Ref. 11]. However, since the first capsule has not yet been tested for Watts Bar Unit 2, the limiting USE value from sister-plant Heat # 899680 analyses is used herein. The sister-plant analyses are contained in WCAP-17455-NP [Ref. 14] for McGuire Unit 2 and WCAP-17669-NP, Revision 1 [Ref. 15] for Catawba Unit 1. This value should be revisited after test results from the first capsule are available.

(f) The Watts Bar Unit 2 surveillance forging material was made from reactor vessel Intermediate Shell Forging 05. Material properties for the surveillance forging material were taken to be identical to those of the vessel forging, since the surveillance material was cut from a prolongation of the actual vessel material. The Watts Bar Unit 2 surveillance weld material was made using Heat # 895075 with Grau L.O. (LW 320) flux, lot P46, which is identical to the Watts Bar Unit 2 Intermediate to Lower Shell Circumferential Weld per WCAP-9455, Revision 3 [Ref. 16]. The chemistry values for the Watts Bar Unit 2 surveillance weld are the average of the values in Table A-2 of WCAP-9455, Revision 3. The USE value for the Watts Bar Unit 2 surveillance weld is taken from Section 3 of WCAP-9455, Revision 3.

(g) Surveillance data exists for weld Heat # 895075 from multiple sources; see Section 4 for more details. The data for the Catawba Unit 1, Watts Bar Unit 1, and McGuire Unit 2 surveillance welds was taken from Table 3-1 of WCAP-17669-NP, Revision 1 [Ref. 15].

Table 3-2 Summary of Watts Bar Unit 2 Reactor Vessel Closure Head Flange and Vessel Flange Initial RT_{NDT} Values

Reactor Vessel Material	Initial RT_{NDT}[(a)] (°F)
Closure Head Flange	-40
Vessel Flange	-22

Notes:

(a) Values taken from WCAP-17035-NP, Revision 2 [Ref. 13] and are consistent with those documented in WCAP-13830, Revision 1 [Ref. 17].

4 SURVEILLANCE DATA

Per Regulatory Guide 1.99, Revision 2 [Ref. 1], calculation of Position 2.1 chemistry factors requires data from the plant-specific surveillance program. In addition to the plant-specific surveillance data, surveillance data generated from surveillance programs at other plants which include a reactor vessel beltline or extended beltline material should also be considered when calculating Position 2.1 chemistry factors.

The surveillance capsule plate material for Watts Bar Unit 2 is from Intermediate Shell Forging 05. The surveillance capsule weld material for Watts Bar Unit 2 is Heat # 895075, which is applicable to the intermediate to lower shell circumferential weld.

While no surveillance capsules have yet to be removed and tested from the Watts Bar Unit 2 reactor, weld Heat # 895075 is contained in the Catawba Unit 1, Watts Bar Unit 1, and McGuire Unit 2 surveillance programs. Thus, the Catawba Unit 1, Watts Bar Unit 1, and McGuire Unit 2 data will be used in the calculation of the Position 2.1 chemistry factor value for Watts Bar Unit 2 weld Heat # 895075. Table 4-1 summarizes the applicable surveillance capsule data pertaining to weld Heat # 895075. Per Appendix A of WCAP-17669-NP, Revision 1 [Ref. 15], the combined Catawba Unit 1, Watts Bar Unit 1, and McGuire Unit 2 surveillance weld data (Heat # 895075) is deemed credible. This credibility conclusion is applicable to the Watts Bar Unit 2 weld Heat # 895075 evaluated herein, since no additional surveillance data was included in this analysis. Therefore, a reduced margin term will be utilized in the ART and PTS calculations contained in Section 7 and Appendix E, respectively.

Table 4-1 Catawba Unit 1, Watts Bar Unit 1, and McGuire Unit 2 Surveillance Capsule Data for Weld Heat # 895075[(a)]

Weld Metal Heat # 895075	Capsule	Capsule Fluence (x 10^{19} n/cm^2, E > 1.0 MeV)	Measured 30 ft-lb Transition Temperature Shift (°F)	Inlet Temperature (°F)	Temperature Adjustment[(b)] (°F)
Catawba Unit 1 Data	Z	0.286	1.91	553	-3
	Y	1.29	17.79		
	V	2.27	26.5		
Watts Bar Unit 1 Data	U	0.447	0.0	560	4
	W	1.08	30.5		
	X	1.71	25.8		
	Z	2.40	13.9		
McGuire Unit 2 Data	V	0.302	38.51	557	1
	X	1.38	35.93		
	U	1.90	23.81		
	W	2.82	43.76		

Notes:

(a) All data taken from WCAP-17669-NP, Revision 1 [Ref. 15] Table 4-1 or 4-2, unless otherwise noted.

(b) Temperature adjustment = 1.0*($T_{capsule}$ - T_{plant}), where T_{plant} = 556°F for Watts Bar Unit 2 (applied to the weld ΔRT_{NDT} data for each of the Catawba Unit 1, Watts Bar Unit 1, and McGuire Unit 2 capsules in the Position 2.1 chemistry factor calculation – See Section 5 for more details).

5 CHEMISTRY FACTORS

The chemistry factors (CFs) were calculated using Regulatory Guide 1.99, Revision 2, Positions 1.1 and 2.1. Position 1.1 chemistry factors for each reactor vessel material are calculated using the best-estimate copper and nickel weight percent of the material and Tables 1 and 2 of Regulatory Guide 1.99, Revision 2. The best-estimate copper and nickel weight percent values for the Watts Bar Unit 2 reactor vessel materials are provided in Table 3-1 of this report.

The Position 2.1 chemistry factor calculation is presented in Table 5-1 for the Watts Bar Unit 2 Heat # 895075 material. This value was calculated using the surveillance data summarized in Section 4 of this report, which includes available surveillance program results from sister plants. The calculation is performed using the method described in Regulatory Guide 1.99, Revision 2. All of the surveillance data is adjusted for irradiation temperature and chemical composition differences in accordance with the guidance presented at an industry meeting held by the NRC on February 12 and 13, 1998 [Ref. 18]. Margin will be applied to the ART calculations in Section 7 and PTS calculations in Appendix E according to the conclusions of the credibility evaluation for the surveillance material, as documented in Section 4.

The Position 1.1 chemistry factors are summarized along with the Position 2.1 chemistry factors in Table 5-2 for Watts Bar Unit 2.

Table 5-1 Calculation of Watts Bar Unit 2 Chemistry Factor Value for Weld Heat # 895075 Using Surveillance Capsule Data

Weld Metal Heat # 895075	Capsule	Capsule f[(a)] (x 10^{19} n/cm^2, E > 1.0 MeV)	FF[(b)]	ΔRT_{NDT}[(c)] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Catawba Unit 1 Data	Z	0.286	0.658	0.00[(d)] (1.91)	0.00	0.433
	Y	1.29	1.071	14.79 (17.79)	15.84	1.147
	V	2.27	1.222	23.50 (26.5)	28.71	1.493
Watts Bar Unit 1 Data	U	0.447	0.776	6.64 (0.0)	5.15	0.602
	W	1.08	1.022	57.27 (30.5)	58.50	1.044
	X	1.71	1.148	49.47 (25.8)	56.77	1.317
	Z	2.40	1.236	29.71 (13.9)	36.73	1.528
McGuire Unit 2 Data	V	0.302	0.672	49.78 (38.51)	33.45	0.452
	X	1.38	1.089	46.53 (35.93)	50.69	1.187
	U	1.90	1.176	31.26 (23.81)	36.75	1.382
	W	2.82	1.276	56.40 (43.76)	71.95	1.628
				SUM:	394.56	12.211
	$CF_{Weld\ Heat\ \#\ 895075} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (394.56) \div (12.211) =$ **32.3°F**					

Notes:

(a) f = fluence.

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

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values. The ΔRT_{NDT} values are adjusted first by the difference in operating temperature then using the ratio procedure to account for differences in the surveillance weld chemistry and the reactor vessel weld chemistry (pre-adjusted values are listed in parentheses and were taken from Table 4-1 of this report). The temperature adjustments are listed in Table 4-1. Ratio applied to the Catawba Unit 1 surveillance data = $CF_{Vessel\ Weld}$ / $CF_{Surv.\ Weld}$ = 68°F / 68°F = 1.00. Ratio applied to the Watts Bar Unit 1 surveillance data = $CF_{Vessel\ Weld}$ / $CF_{Surv.\ Weld}$ = 68°F / 41°F = 1.66. Ratio applied to the McGuire Unit 2 surveillance data = $CF_{Vessel\ Weld}$ / $CF_{Surv.\ Weld}$ = 68°F / 54°F = 1.26.

(d) A negative ΔRT_{NDT} value was calculated (-1.09°F). Physically, this should not occur; thus, a conservative value of 0°F was utilized for this calculation.

Table 5-2 Summary of Watts Bar Unit 2 Positions 1.1 and 2.1 Chemistry Factors

Reactor Vessel Material and Identification Number	Heat Number	Chemistry Factor (°F)	
		Position 1.1[(a)]	Position 2.1
Reactor Vessel Beltline Materials			
Intermediate Shell Forging 05	527828	31.0	- - -
Lower Shell Forging 04	528658	31.0	- - -
Intermediate to Lower Shell Circumferential Weld W05	895075	68.0	32.3[(b)]
Reactor Vessel Extended Beltline Materials			
Upper Shell Forging 06	411572	44.0	- - -
Upper to Intermediate Shell Circumferential Weld Seam W06	899680	41.0	- - -
Lower Shell to Bottom Head Ring Circumferential Weld Seam W04	899680	41.0	- - -
Bottom Head Ring 03	5329	37.0	- - -
Surveillance Weld Data			
Watts Bar Unit 2	895075	44.9	- - -
Catawba Unit 1		68.0	- - -
Watts Bar Unit 1		41.0	- - -
McGuire Unit 2		54.0	- - -

Notes:

(a) Position 1.1 chemistry factors were calculated using the copper and nickel weight percent values presented in Table 3-1 of this report and Tables 1 and 2 of Regulatory Guide 1.99, Revision 2 [Ref. 1].

(b) Position 2.1 chemistry factor was taken from Table 5-1 of this report. As discussed in Section 4, the surveillance weld data was deemed credible.

6 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

6.1 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 the 2000 Addenda of Section XI, Appendix G of the ASME Code [Ref. 3]. The K_{Ic} curve is given by the following equation:

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

where,

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

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

6.2 METHODOLOGY FOR PRESSURE-TEMPERATURE LIMIT CURVE DEVELOPMENT

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

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

where,

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

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

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

C = 2.0 for Level A and Level B service limits

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

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

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

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:

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

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

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

where HU is the heatup rate in °F/hr.

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

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

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

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

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

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

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:

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

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

Note that Equations 3, 7, and 8 were implemented in the OPERLIM computer code, which is the program used to generate the pressure-temperature (P-T) limit curves. The P-T curve methodology is the same as that described in WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" [Ref. 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.

6.3 CLOSURE HEAD/VESSEL FLANGE REQUIREMENTS

10 CFR Part 50, Appendix G [Ref. 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 3-2. The limiting unirradiated RT_{NDT} of -22°F is associated with the vessel flange of the Watts Bar Unit 2 vessel, so the minimum allowable temperature of this region is 98°F at pressures greater than 621 psig (without margins for instrument uncertainties). Consistent with the previous P-T limit curves [Ref. 13], the minimum allowable temperature of the flange region was rounded up to 100°F. This limit is shown in Figures 8-1 and 8-2.

6.4 BOLTUP TEMPERATURE REQUIREMENTS

The minimum boltup temperature is the minimum allowable temperature at which the reactor vessel closure head bolts can be preloaded. It is determined by the highest reference temperature, RT_{NDT}, in the closure flange region. This requirement is established in Appendix G to 10 CFR 50 [Ref. 4]. Per the NRC-approved methodology in WCAP-14040-A, Revision 4 [Ref. 2], the minimum boltup temperature should be 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 3-2, the minimum boltup temperature for the Watts Bar Unit 2 reactor vessel is 60°F (without margins for instrument uncertainties). This limit is shown in Figures 8-1 and 8-2.

7 CALCULATION OF ADJUSTED REFERENCE TEMPERATURE

From Regulatory Guide 1.99, Revision 2 [Ref. 1], the adjusted reference temperature (ART) for each material in the beltline region is given by the following expression:

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

Initial RT_{NDT} is the reference temperature for the unirradiated material as defined in Paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code [Ref. 12]. 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 if 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:

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

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

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

where x inches (reactor vessel cylindrical shell beltline thickness is 8.465 inches) is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then placed in Equation 11 to calculate the ΔRT_{NDT} at the specific depth.

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

Table 7-1 contains the surface fluence values at 32 EFPY, which were used for the development of the P-T limit curves contained in this report. Table 7-1 also contains the 1/4T and 3/4T calculated fluence values and fluence factors (FFs), per Regulatory Guide 1.99, Revision 2. The values in this table will be used to calculate the 32 EFPY ART values for the Watts Bar Unit 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. 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}.

Contained in Tables 7-2 and 7-3 are the 32 EFPY ART calculations at the 1/4T and 3/4T locations for generation of the Watts Bar Unit 2 heatup and cooldown curves.

The inlet and outlet nozzle forging weld materials for Watts Bar Unit 2 have projected fluence values that do not exceed the 1 x 10^{17} n/cm^2 fluence threshold at 32 EFPY per Table 2-9. The projected fluence values for the inlet and outlet nozzle forging weld materials provide conservative estimates of the fluence values of the inlet and outlet nozzles at the lowest extent of the nozzle; therefore, per NRC RIS 2014-11 [Ref. 9], neutron radiation embrittlement need not be considered herein for the nozzle forging or weld materials. Thus, ART calculations for the inlet and outlet nozzle forging and weld materials utilizing the 1/4T and 3/4T fluence values are excluded from Tables 7-2 and 7-3, respectively. Limiting ART values for the nozzle forging materials are contained in Appendix B. Finally, the second conclusion of TLR-RES/DE/CIB-2013-01 [Ref. 10] states that if ΔRT_{NDT} is calculated to be less than 25°F, then embrittlement need not be considered. This conclusion was applied, as necessary, to the ART calculations documented in Tables 7-2 and 7-3 for the extended beltline materials.

The limiting ART values for Watts Bar Unit 2 to be used in the generation of the P-T limit curves are based on Intermediate Shell Forging 05 (Position 1.1). In order to provide an additional margin of conservatism, the limiting calculated ART values were rounded and increased by 10°F. The increased limiting ART values, using the "Axial Flaw" methodology, for Intermediate Shell Forging 05 are summarized in Table 7-4.

Table 7-1 Fluence Values and Fluence Factors for the Vessel Surface, 1/4T and 3/4T Locations for the Watts Bar Unit 2 Reactor Vessel Materials at 32 EFPY

Reactor Vessel Material	Surface Fluence, f [(a)] (n/cm^2, E > 1.0 MeV)	1/4T f (n/cm^2, E > 1.0 MeV)	1/4T FF	3/4T f (n/cm^2, E > 1.0 MeV)	3/4T FF
Reactor Vessel Beltline Materials					
Intermediate Shell Forging 05	1.861×10^{19}	1.120×10^{19}	1.032	0.4055×10^{19}	0.7497
Lower Shell Forging 04	1.891×10^{19}	1.138×10^{19}	1.036	0.4121×10^{19}	0.7540
Intermediate to Lower Shell Circumferential Weld Seam W05 (Heat # 895075)	1.861×10^{19}	1.120×10^{19}	1.032	0.4055×10^{19}	0.7497
Reactor Vessel Extended Beltline Materials					
Upper Shell Forging 06	0.1097×10^{19}	0.0660×10^{19}	0.3390	0.02390×10^{19}	0.1919
Upper to Intermediate Shell Circumferential Weld Seam W06 (Heat # 899680)	0.1097×10^{19}	0.0660×10^{19}	0.3390	0.02390×10^{19}	0.1919
Lower Shell to Bottom Head Ring Circumferential Weld Seam W04 (Heat # 899680)	0.2454×10^{19}	0.1477×10^{19}	0.4993	0.05347×10^{19}	0.3035
Bottom Head Ring 03	0.2454×10^{19}	0.1477×10^{19}	0.4993	0.05347×10^{19}	0.3035

Note:

(a) 32 EFPY fluence values are documented in Tables 2-7 and 2-10.

Table 7-2 Adjusted Reference Temperature Evaluation for the Watts Bar Unit 2 Reactor Vessel Beltline Materials through 32 EFPY at the 1/4T Location

Reactor Vessel Material and ID Number	Heat Number	CF (°F)	1/4T Fluence (n/cm^2, E > 1.0 MeV)	1/4T FF	$RT_{NDT(U)}$[(a)] (°F)	ΔRT_{NDT}[(b)] (°F)	σ_I[(a)] (°F)	σ_Δ[(c)] (°F)	Margin (°F)	ART[(d)] (°F)
Reactor Vessel Beltline Materials										
Intermediate Shell Forging 05	527828	31.0	1.120×10^{19}	1.032	14	32.0	0.0	16.0	32.0	78.0
Lower Shell Forging 04	528658	31.0	1.138×10^{19}	1.036	5	32.1	0.0	16.1	32.1	69.2
Intermediate to Lower Shell Circumferential Weld Seam W05	895075	68.0	1.120×10^{19}	1.032	-50	70.2	0.0	28.0	56.0	76.2
Using credible Catawba Unit 1, Watts Bar Unit 1, and McGuire Unit 2 surveillance data	895075	32.3	1.120×10^{19}	1.032	-50	33.3	0.0	14.0	28.0	11.3
Reactor Vessel Extended Beltline Materials										
Upper Shell Forging 06	411572	44.0	0.0660×10^{19}	0.3390	-14	0.0 (14.9)	0.0	0.0	0.0	-14
Upper to Intermediate Shell Circumferential Weld Seam W06	899680	41.0	0.0660×10^{19}	0.3390	10	0.0 (13.9)	0.0	0.0	0.0	10
Lower Shell to Bottom Head Ring Weld Seam W04	899680	41.0	0.1477×10^{19}	0.4993	10	0.0 (20.5)	0.0	0.0	0.0	10
Bottom Head Ring 03	5329	37.0	0.1477×10^{19}	0.4993	-40	0.0 (18.5)	0.0	0.0	0.0	-40

Notes:

(a) Initial RT_{NDT} values taken from Table 3-1. All initial RT_{NDT} values are measured value; thus, σ_I = 0°F for each material.

(b) As discussed in Section 7, calculated extended beltline ΔRT_{NDT} values less than 25°F have been reduced to zero per TLR-RES/DE/CIB-2013-01 [Ref. 10]. Actual calculated ΔRT_{NDT} values are listed in parentheses.

(c) As discussed in Section 4, the surveillance weld data for Heat # 895075 data was deemed credible. Per the guidance of Regulatory Guide 1.99, Revision 2 [Ref. 1], the base metal σ_Δ = 17°F for Position 1.1. Also per Regulatory Guide 1.99, Revision 2 [Ref. 1], the weld metal σ_Δ = 28°F for Position 1.1, and the weld metal σ_Δ = 14°F for Position 2.1 with credible surveillance data. However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$.

(d) The Regulatory Guide 1.99, Revision 2 [Ref. 1] methodology was used to calculate ART values. ART = $RT_{NDT(U)} + \Delta RT_{NDT}$ + Margin.

Table 7-3 Adjusted Reference Temperature Evaluation for the Watts Bar Unit 2 Reactor Vessel Beltline Materials through 32 EFPY at the 3/4T Location

Reactor Vessel Material and ID Number	Heat Number	CF (°F)	3/4T Fluence (n/cm^2, E > 1.0 MeV)	3/4T FF	$RT_{NDT(U)}$[(a)] (°F)	ΔRT_{NDT}[(b)] (°F)	σ_I[(a)] (°F)	σ_Δ[(c)] (°F)	Margin (°F)	ART[(d)] (°F)
Reactor Vessel Beltline Materials										
Intermediate Shell Forging 05	527828	31.0	0.4055×10^{19}	0.7497	14	23.2	0.0	11.6	23.2	60.5
Lower Shell Forging 04	528658	31.0	0.4121×10^{19}	0.7540	5	23.4	0.0	11.7	23.4	51.7
Intermediate to Lower Shell Circumferential Weld Seam W05	895075	68.0	0.4055×10^{19}	0.7497	-50	51.0	0.0	25.5	51.0	52.0
Using credible Catawba Unit 1, Watts Bar Unit 1, and McGuire Unit 2 surveillance data	895075	32.3	0.4055×10^{19}	0.7497	-50	24.2	0.0	12.1	24.2	-1.6
Reactor Vessel Extended Beltline Materials										
Upper Shell Forging 06	411572	44.0	0.02390×10^{19}	0.1919	-14	0.0 (8.4)	0.0	0.0	0.0	-14
Upper to Intermediate Shell Circumferential Weld Seam W06	899680	41.0	0.02390×10^{19}	0.1919	10	0.0 (7.9)	0.0	0.0	0.0	10
Lower Shell to Bottom Head Ring Weld Seam W04	899680	41.0	0.05347×10^{19}	0.3035	10	0.0 (12.4)	0.0	0.0	0.0	10
Bottom Head Ring 03	5329	37.0	0.05347×10^{19}	0.3035	-40	0.0 (11.2)	0.0	0.0	0.0	-40

Notes:

(a) Initial RT_{NDT} values taken from Table 3-1. All initial RT_{NDT} values are measured value; thus, σ_I = 0°F for each material.

(b) As discussed in Section 7, calculated extended beltline ΔRT_{NDT} values less than 25°F have been reduced to zero per TLR-RES/DE/CIB-2013-01 [Ref. 10]. Actual calculated ΔRT_{NDT} values are listed in parentheses.

(c) As discussed in Section 4, the surveillance weld data for Heat # 895075 data was deemed credible. Per the guidance of Regulatory Guide 1.99, Revision 2 [Ref. 1], the base metal σ_Δ = 17°F for Position 1.1. Also per Regulatory Guide 1.99, Revision 2 [Ref. 1], the weld metal σ_Δ = 28°F for Position 1.1, and the weld metal σ_Δ = 14°F for Position 2.1 with credible surveillance data. However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$.

(d) The Regulatory Guide 1.99, Revision 2 [Ref. 1] methodology was used to calculate ART values. ART = $RT_{NDT(U)} + \Delta RT_{NDT}$ + Margin.

Table 7-4 Summary of the Increased Limiting ART Values Used in the Generation of the Watts Bar Unit 2 Heatup and Cooldown Curves at 32 EFPY

1/4T Limiting ART[(a)]	3/4T Limiting ART[(a)]
88°F	71°F
Intermediate Shell Forging 05 (Position 1.1)	

Notes:

(a) The ART values used for P-T limit curve development in this report are the limiting ART values calculated in Tables 7-2 and 7-3 rounded and increased by 10°F to add additional margin; this approach is conservative.

8 HEATUP AND COOLDOWN PRESSURE-TEMPERATURE LIMIT CURVES

Pressure-temperature limit curves for normal heatup and cooldown of the primary reactor coolant system have been calculated for the pressure and temperature in the reactor vessel cylindrical beltline region using the methods discussed in Sections 6 and 7 of this report. This approved methodology is also presented in WCAP-14040-A, Revision 4 [Ref. 2].

Figure 8-1 presents the limiting heatup curves without margins for possible instrumentation errors using heatup rates of 60 and 100°F/hr applicable for 32 EFPY, with the flange requirements and using the "Axial Flaw" methodology. Figure 8-2 presents the limiting cooldown curves without margins for possible instrumentation errors using cooldown rates of 0, 20, 40, 60, and 100°F/hr applicable for 32 EFPY, with the flange requirements and using the "Axial Flaw" methodology. The heatup and cooldown curves were generated using the 1998 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 8-1 and 8-2. 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 8-1 (heatup curve only). The straight-line portion of the criticality limit is at the minimum permissible temperature for the 2485 psig inservice hydrostatic test as required by Appendix G to 10 CFR Part 50. The governing equation for the hydrostatic test is defined in the 1998 through the 2000 Addenda ASME Code Section XI, Appendix G as follows:

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

where,

K_{Im} is the stress intensity factor covered by membrane (pressure) stress [see page 6-2, Equation (3)],

$K_{Ic} = 33.2 + 20.734\ e^{[0.02\ (T - RT_{NDT})]}$ [see page 6-1 Equation (1)],

T is the minimum permissible metal temperature, and

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

The criticality limit curve specifies pressure-temperature limits for core operation in order to provide additional margin during actual power production. The pressure-temperature limits for core operation (except for low power physics tests) are that: 1) the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and 2) the reactor vessel must be at least 40°F higher than the minimum permissible temperature in the corresponding pressure-temperature curve for heatup and cooldown calculated as described in Section 6 of this report. For the heatup and cooldown curves without margins for instrumentation errors, the minimum temperature for the inservice hydrostatic leak tests for the Watts Bar Unit 2 reactor vessel at 32 EFPY is 149°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 8-1 and 8-2 define all of the above limits for ensuring prevention of non-ductile failure for the Watts Bar Unit 2 reactor vessel for 32 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 8-1 and 8-2 are presented in Tables 8-1 and 8-2. The P-T limit curves shown in Figures 8-1 and 8-2 were generated based on the limiting ART values for the cylindrical beltline and extended beltline reactor vessel materials rounded and increased by 10°F to add additional margin; this approach is conservative. As discussed in Appendix B, the P-T limits developed for the cylindrical beltline region bound the P-T limits for the reactor vessel inlet and outlet nozzles for Watts Bar Unit 2 at 32 EFPY.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Intermediate Shell Forging 05 using Regulatory Guide 1.99 Position 1.1 data
LIMITING ART VALUES AT 32 EFPY: 1/4T, 88°F (Axial Flaw)
3/4T, 71°F (Axial Flaw)

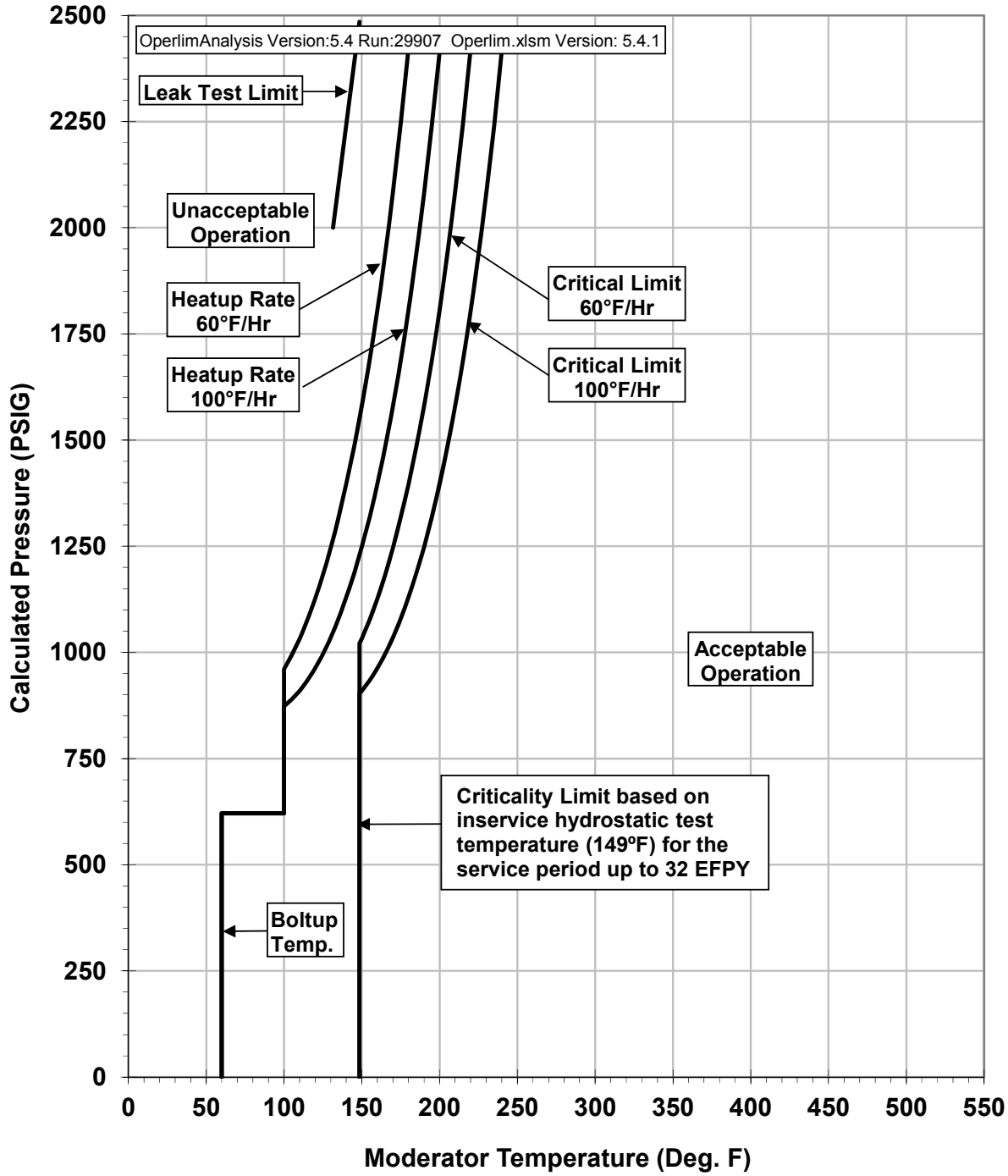

Figure 8-1 Watts Bar Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rates of 60 and 100°F/hr) Applicable for 32 EFPY (with Flange Requirements and without Margins for Instrumentation Errors) using the 1998 through the 2000 Addenda App. G Methodology (w/ K_{Ic})

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Intermediate Shell Forging 05 using Regulatory Guide 1.99 Position 1.1 data
LIMITING ART VALUES AT 32 EFPY: 1/4T, 88°F (Axial Flaw)
3/4T, 71°F (Axial Flaw)

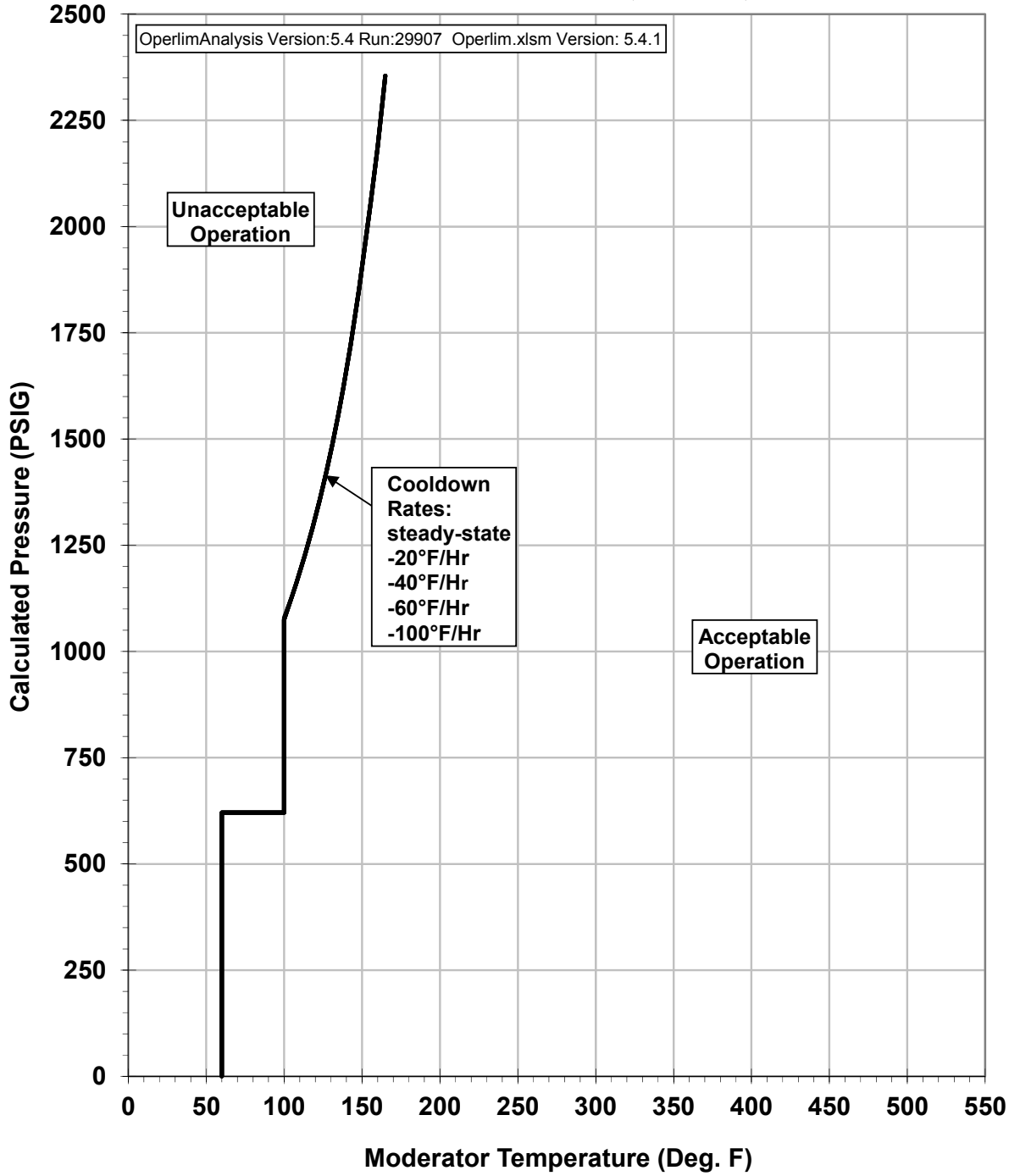

Figure 8-2 Watts Bar Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 20, 40, 60, and 100°F/hr) Applicable for 32 EFPY (with Flange Requirements and without Margins for Instrumentation Errors) using the 1998 through the 2000 Addenda App. G Methodology (w/ K_{Ic})

Table 8-1 Watts Bar Unit 2 32 EFPY Heatup Curve Data Points using the 1998 through the 2000 Addenda App. G Methodology (w/ K_{Ic}, w/ Flange Requirements, and w/o Margins for Instrumentation Errors)

60°F/hr Heatup		60°F/hr Criticality		100°F/hr Heatup		100°F/hr Criticality	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
60	0	149	0	60	0	149	0
60	621	149	1021	60	621	149	904
65	621	150	1032	65	621	150	909
70	621	155	1076	70	621	155	934
75	621	160	1126	75	621	160	963
80	621	165	1183	80	621	165	996
85	621	170	1246	85	621	170	1035
90	621	175	1317	90	621	175	1079
95	621	180	1395	95	621	180	1129
100	621	185	1483	100	621	185	1185
100	960	190	1580	100	873	190	1248
105	993	195	1687	105	889	195	1318
110	1032	200	1806	110	909	200	1396
115	1076	205	1938	115	934	205	1483
120	1126	210	2083	120	963	210	1580
125	1183	215	2244	125	996	215	1686
130	1246	220	2421	130	1035	220	1805
135	1317			135	1079	225	1936
140	1395			140	1129	230	2080
145	1483			145	1185	235	2240
150	1580			150	1248	240	2417
155	1687			155	1318		
160	1806			160	1396		
165	1938			165	1483		
170	2083			170	1580		
175	2244			175	1686		
180	2421			180	1805		
				185	1936		
				190	2080		
				195	2240		
				200	2417		

Leak Test Limit	
T (°F)	P (psig)
132	2000
149	2485

Table 8-2 Watts Bar Unit 2 32 EFPY Cooldown Curve Data Points using the 1998 through the 2000 Addenda App. G Methodology (w/ K_{Ic}, w/ Flange Requirements, and w/o Margins for Instrumentation Errors)

Steady-State		20°F/hr Cooldown		40°F/hr Cooldown		60°F/hr Cooldown		100°F/hr Cooldown	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
60	0	60	0	60	0	60	0	60	0
60	621	60	621	60	621	60	621	60	621
65	621	65	621	65	621	65	621	65	621
70	621	70	621	70	621	70	621	70	621
75	621	75	621	75	621	75	621	75	621
80	621	80	621	80	621	80	621	80	621
85	621	85	621	85	621	85	621	85	621
90	621	90	621	90	621	90	621	90	621
95	621	95	621	95	621	95	621	95	621
100	621	100	621	100	621	100	621	100	621
100	1080	100	1075	100	1075	100	1075	100	1075
105	1130	105	1130	105	1130	105	1130	105	1130
110	1185	110	1185	110	1185	110	1185	110	1185
115	1247	115	1247	115	1247	115	1247	115	1247
120	1315	120	1315	120	1315	120	1315	120	1315
125	1390	125	1390	125	1390	125	1390	125	1390
130	1473	130	1473	130	1473	130	1473	130	1473
135	1564	135	1564	135	1564	135	1564	135	1564
140	1665	140	1665	140	1665	140	1665	140	1665
145	1777	145	1777	145	1777	145	1777	145	1777
150	1901	150	1901	150	1901	150	1901	150	1901
155	2037	155	2037	155	2037	155	2037	155	2037
160	2188	160	2188	160	2188	160	2188	160	2188
165	2355	165	2355	165	2355	165	2355	165	2355

9 REFERENCES

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4. 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.
5. Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," U.S. Nuclear Regulatory Commission, March 2001.
6. Westinghouse Report WCAP-16083-NP, Revision 1, "Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry," April 2013.
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14. Westinghouse Report WCAP-17455-NP, Revision 0, "McGuire Units 1 and 2 Measurement Uncertainty Recapture (MUR) Power Uprate: Reactor Vessel Integrity and Neutron Fluence Evaluations," February 2012.
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18. K. Wichman, M. Mitchell, and A. Hiser, U.S. NRC, Generic Letter 92-01 and RPV Integrity Workshop Handouts, *NRC/Industry Workshop on RPV Integrity Issues*, February 12, 1998. *[ADAMS Accession Number ML110070570]*

19. ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," ASTM, July 1982.

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 32 EFPY for Watts Bar Unit 2. The reactor vessel cylindrical shell radii to the 1/4T and 3/4T locations are as follows:

- 1/4T Radius = 88.768 inches
- 3/4T Radius = 93.001 inches

Table A-1 K_{It} Values for Watts Bar Unit 2 at 32 EFPY 100°F/hr Heatup Curves (w/ Flange Requirements, and w/o Margins for Instrument Errors)

Water Temp. (°F)	Vessel Temperature at 1/4T Location for 100°F/hr Heatup (°F)	1/4T Thermal Stress Intensity Factor (ksi √in.)	Vessel Temperature at 3/4T Location for 100°F/hr Heatup (°F)	3/4T Thermal Stress Intensity Factor (ksi √in.)
60	56.015	-0.994	55.047	0.478
65	58.635	-2.438	55.318	1.443
70	61.728	-3.675	56.029	2.419
75	65.038	-4.846	57.225	3.331
80	68.620	-5.851	58.848	4.142
85	72.314	-6.766	60.858	4.864
90	76.193	-7.556	63.213	5.501
95	80.170	-8.276	65.868	6.069
100	84.280	-8.903	68.790	6.572
105	88.475	-9.472	71.945	7.019
110	92.767	-9.970	75.302	7.416
115	97.129	-10.424	78.839	7.773
120	101.561	-10.823	82.530	8.092
125	106.051	-11.189	86.359	8.379
130	110.592	-11.512	90.308	8.637
135	115.181	-11.809	94.363	8.870
140	119.806	-12.074	98.512	9.080
145	124.470	-12.318	102.742	9.271
150	129.161	-12.537	107.045	9.445
155	133.883	-12.740	111.410	9.603
160	138.625	-12.924	115.832	9.748
165	143.391	-13.096	120.303	9.882
170	148.173	-13.252	124.817	10.005
175	152.974	-13.399	129.369	10.119
180	157.787	-13.534	133.955	10.225
185	162.615	-13.662	138.570	10.324
190	167.452	-13.780	143.211	10.417
195	172.301	-13.894	147.875	10.504
200	177.156	-13.999	152.559	10.587
205	182.021	-14.101	157.261	10.665
210	186.891	-14.197	161.979	10.739

Table A-2 K_{It} Values for Watts Bar Unit 2 at 32 EFPY 100°F/hr Cooldown Curves (w/ Flange Requirements, and w/o Margins for Instrument Errors)

Water Temp. (°F)	Vessel Temperature at 1/4T Location for 100°F/hr Cooldown (°F)	100°F/hr Cooldown 1/4T Thermal Stress Intensity Factor (ksi √in.)
210	236.000	16.310
205	230.917	16.244
200	225.833	16.178
195	220.750	16.111
190	215.666	16.046
185	210.582	15.979
180	205.497	15.913
175	200.413	15.846
170	195.329	15.780
165	190.244	15.713
160	185.160	15.647
155	180.075	15.581
150	174.990	15.514
145	169.906	15.448
140	164.821	15.382
135	159.737	15.316
130	154.653	15.250
125	149.568	15.183
120	144.484	15.118
115	139.400	15.052
110	134.316	14.986
105	129.232	14.921
100	124.148	14.855
95	119.064	14.790
90	113.981	14.725
85	108.897	14.659
80	103.814	14.595
75	98.731	14.530
70	93.647	14.465
65	88.565	14.401
60	83.483	14.336

APPENDIX B REACTOR VESSEL INLET AND OUTLET NOZZLES

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

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

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

B.1 CALCULATION OF ADJUSTED REFERENCE TEMPERATURES

The fracture toughness (K_{Ic}) used for the inlet and outlet nozzle material is defined in Appendix G of the Section XI ASME Code, as discussed in Section 6 of this report. The K_{Ic} fracture toughness curve is dependent on the Adjusted Reference Temperature (ART) value for irradiated materials. The ART values for the inlet and outlet nozzle materials are determined using the methodology contained in Regulatory Guide 1.99, Revision 2 [Ref. B-3], which is described in Section 7 of this report, as well as weight percent (wt. %) copper (Cu) and nickel (Ni) values, initial RT_{NDT} values, and projected neutron fluence as inputs. The material properties for each of the reactor vessel inlet and outlet nozzle forging materials are documented in Table B-1 and a summary of the limiting inlet and outlet nozzle ART values for Watts Bar Unit 2 is presented in Table B-2.

Nozzle Material Properties

The Watts Bar Unit 2 nozzle material properties are provided in Table B-1. Copper (Cu), Nickel (Ni), Manganese (Mn), and Phosphorus (P) weight percent (wt. %) values were obtained from the Watts Bar Unit 2 CMTRs for each of the Watts Bar Unit 2 reactor vessel inlet and outlet nozzles.

The Charpy V-Notch forging specimen orientation for the inlet and outlet nozzles was not reported in the Watts Bar Unit 2 CMTRs; thus, it was conservatively assumed that the orientation was the "strong direction" for each nozzle forging. Due to the assumed lack of transverse test data, the initial RT_{NDT} values were determined for each of the Watts Bar Unit 2 reactor vessel inlet and outlet nozzle forging materials using the Branch Technical Position (BTP) 5-3, Position 1.1(3) methodology [Ref. B-4]. The initial RT_{NDT} values for all of the nozzle materials were determined directly from the data or by using CVGRAPH, Version 6.02 hyperbolic tangent curve fits through the minimum data points, in accordance with ASME Code Section III, Subarticle NB-2331, Paragraph (a)(4) [Ref. B-5]. The initial RT_{NDT} values were determined using both BTP 5-3 Position 1.1(3)(a) and Position 1.1(3)(b), and the more limiting initial RT_{NDT} value was chosen for each nozzle forging material. The initial USE values were determined in accordance with the methodology described in ASTM E185-82 [Ref. B-6]. For each of the nozzle forging materials, use of BTP 5-3 Paragraph 1.2 [Ref. B-5] was necessary. The Watts Bar Unit 2 initial RT_{NDT} and initial USE values for the inlet and outlet nozzles materials are summarized in Table B-1.

Nozzle Calculated Neutron Fluence Values

The maximum fast neutron (E > 1 MeV) exposure of the Watts Bar Unit 2 reactor vessel materials is discussed in Section 2 of this report. The fluence values used in the inlet and outlet nozzle ART calculations were calculated at the lowest extent of the nozzles (i.e., the nozzle to nozzle shell weld locations) and were chosen at an elevation lower than the actual elevation of the postulated flaw, which is at the inside corner of the nozzle, for conservatism.

Per Table 2-9, the inlet nozzles are determined to receive a projected maximum fluence of 8.861×10^{16} n/cm^2 (E > 1 MeV) at the lowest extent of the nozzles at 32 EFPY. Similarly, the outlet nozzles are projected to achieve a maximum fluence value of 4.630×10^{16} n/cm^2 (E > 1 MeV) at the lowest extent of the nozzles at 32 EFPY. Thus, the maximum neutron fluence values for the nozzle materials are not projected to exceed a fluence of 1×10^{17} n/cm^2 (E > 1.0 MeV) at 32 EFPY. Per NRC RIS 2014-11 [Ref. B-1], embrittlement of reactor vessel materials, with projected fluence values less than 1×10^{17} n/cm^2 (E > 1.0 MeV), does not need to be considered. Therefore, the initial RT_{NDT} values documented in Table B-1 are identical to the nozzle ART values.

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

The use of the embrittlement conclusion of NRC RIS 2014-11 [Ref. B-1], and thus the limiting ART values summarized in Table B-2, will remain unchanged as long as the fluence values assigned to the inlet and outlet nozzles remain below 1.0×10^{17} n/cm^2 (E > 1.0 MeV). If these fluence values are reached, the Watts Bar Unit 2 nozzle material ART values should be re-evaluated.

Table B-1 Summary of the Watts Bar Unit 2 Reactor Vessel Nozzle Material Initial RT_{NDT}, Chemistry, and Fluence Values at 32 EFPY

Reactor Vessel Material	Chemical Composition[a]				Fracture Toughness Properties[b]		Fluence at Lowest Extent of Nozzle[d] (n/cm^2, E > 1.0 MeV)
	Wt. % Cu	Wt. % Ni	Wt. % Mn	Wt. % P	$RT_{NDT(U)}$ (°F)	Initial USE (ft-lb)	
Inlet Nozzle 11 (Heat # 5328)	0.05	0.83	0.75	0.008	-22	78	8.861×10^{16}
Inlet Nozzle 12 (Heat # 5330)	0.06	0.85	0.77	0.011	-14	67	8.861×10^{16}
Inlet Nozzle 13 (Heat # 5331)	0.06	0.82	0.75	0.010	-8	61	8.861×10^{16}
Inlet Nozzle 14 (Heat # 5335)	0.04	0.79	0.77	0.009	-13	87	8.861×10^{16}
Outlet Nozzle 15 (Heat # 5319)	0.06	0.86	0.69	0.009	-22	90	4.630×10^{16}
Outlet Nozzle 16 (Heat # 5324)	0.06	0.84	0.71	0.011	-13	72	4.630×10^{16}
Outlet Nozzle 17 (Heat # 5327)	0.05	0.86	0.76	0.009	-39	84	4.630×10^{16}
Outlet Nozzle 18 (Heat # 5334)	0.04	0.80	0.77	0.009	-31	>84[c]	4.630×10^{16}

Notes:

(a) Chemistry values were taken from the Watts Bar Unit 2 CMTRs.

(b) The nozzle forging $RT_{NDT(U)}$ and initial USE values were calculated as a part of this evaluation. The $RT_{NDT(U)}$ values are based on drop-weight data, tangentially-oriented Charpy V-notch test data and NUREG-0800, BTP 5-3 [Ref. B-4] Position 1.1(3)(a) and (b), with the more limiting $RT_{NDT(U)}$ value being selected. The initial USE values are the average of all impact energy values with ≥ 95% shear reduced to 65% of their original values to conservatively approximate transverse data per NUREG-0800, BTP 5-3 Position 1.2 methodology, unless otherwise noted.

(c) Since Charpy data with > 95% shear is not available for this material; this initial USE value is estimated based on the highest impact energy obtained with shear < 95%, reduced to 65% of its original value per NUREG-0800, BTP 5-3 Position 1.2 methodology.

(d) Fluence values conservatively correspond to 32 EFPY fluence values at the lowest extent of the nozzle weld.

Table B-2 Summary of the Limiting ART Values for the Watts Bar Unit 2 Inlet and Outlet Nozzle Materials

EFPY	Nozzle Material and ID Number	Limiting ART Value (°F)
32	Inlet Nozzle 13 (Heat # 5331)	-8
	Outlet Nozzle 16 (Heat # 5324)	-13

B.2 NOZZLE COOLDOWN PRESSURE-TEMPERATURE LIMITS

Allowable pressures are determined for a given temperature based on the fracture toughness of the limiting nozzle material along with the appropriate pressure and thermal stress intensity factors. The Watts Bar Unit 2 nozzle fracture toughness used to determine the P-T limits is calculated using the limiting inlet and outlet nozzle ART values from Table B-2. The stress intensity factor correlations used for the nozzle corners are provided in Oak Ridge National Laboratory study, ORNL/TM-2010/246 [Ref. B-7], and are consistent with ASME PVP2011-57015 [Ref. B-8]. The methodology includes postulating an inside surface 1/4T nozzle corner flaw, and calculating through-wall nozzle corner stresses for a cooldown rate of 100ºF/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 were calculated based on the methodology provided in ORNL/TM-2010/246. The stress intensity factor expression for a rounded corner is:

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

where,

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

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

The Watts Bar Unit 2 reactor vessel inlet and outlet nozzle P-T limit curves are shown in Figures B-1 and B-2, respectively, based on the stress intensity factor expression discussed above; also shown in these figures are the traditional cylindrical beltline cooldown P-T limit curves from Figure 8-2. The nozzle P-T limit curves are provided for a cooldown rate of 100°F/hr, along with a steady-state curve.

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

Conclusion

Based on the results shown in Figures B-1 and B-2, it is concluded that the nozzle P-T limits are bounded by the traditional cylindrical beltline curves. Therefore, the P-T limits provided in Section 8 for 32 EFPY remain limiting for the beltline and non-beltline reactor vessel components.

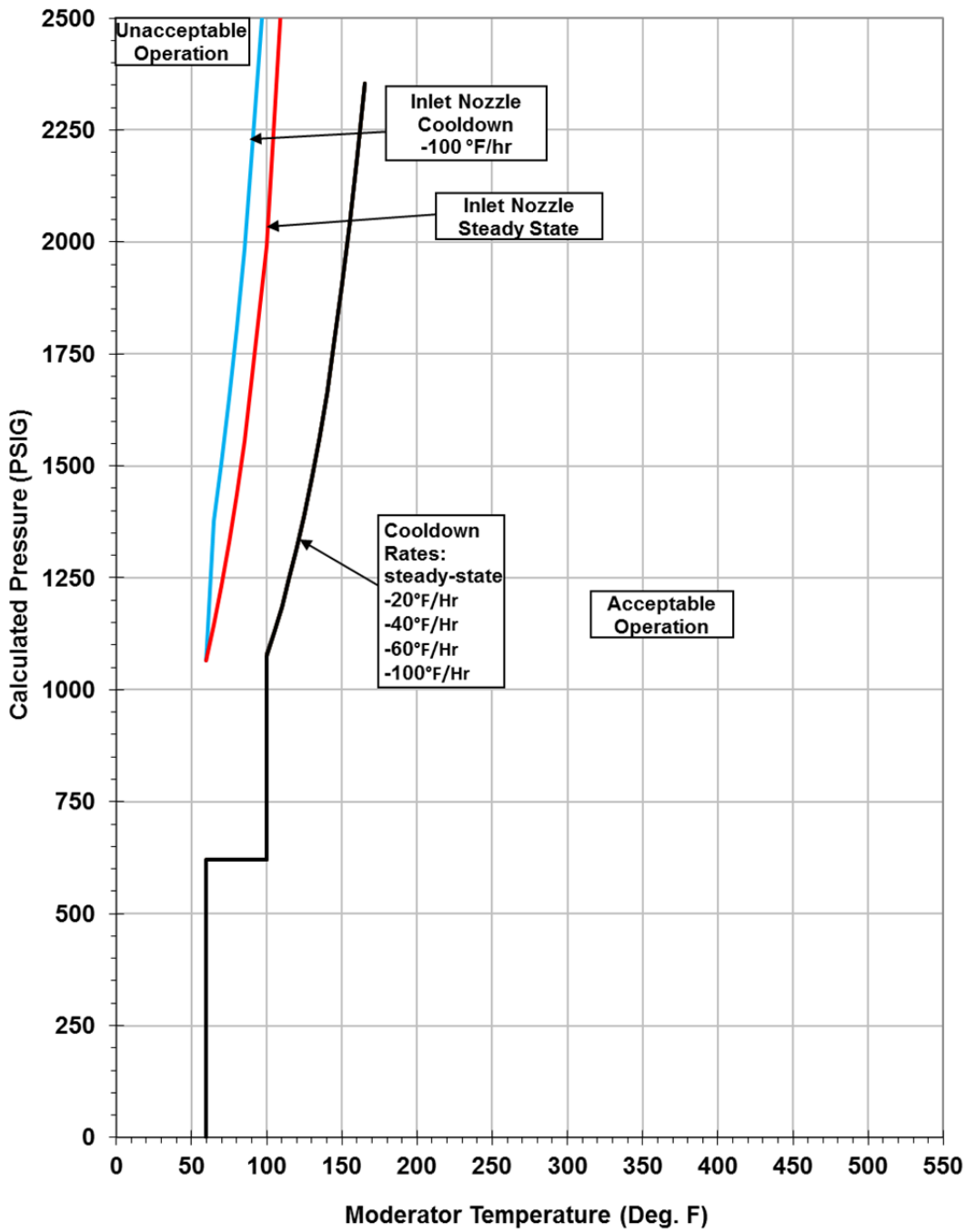

Figure B-1 Comparison of Watts Bar Unit 2 32 EFPY Beltline P-T Limits to 32 EFPY Inlet Nozzle P-T Limits, Without Margins for Instrumentation Errors

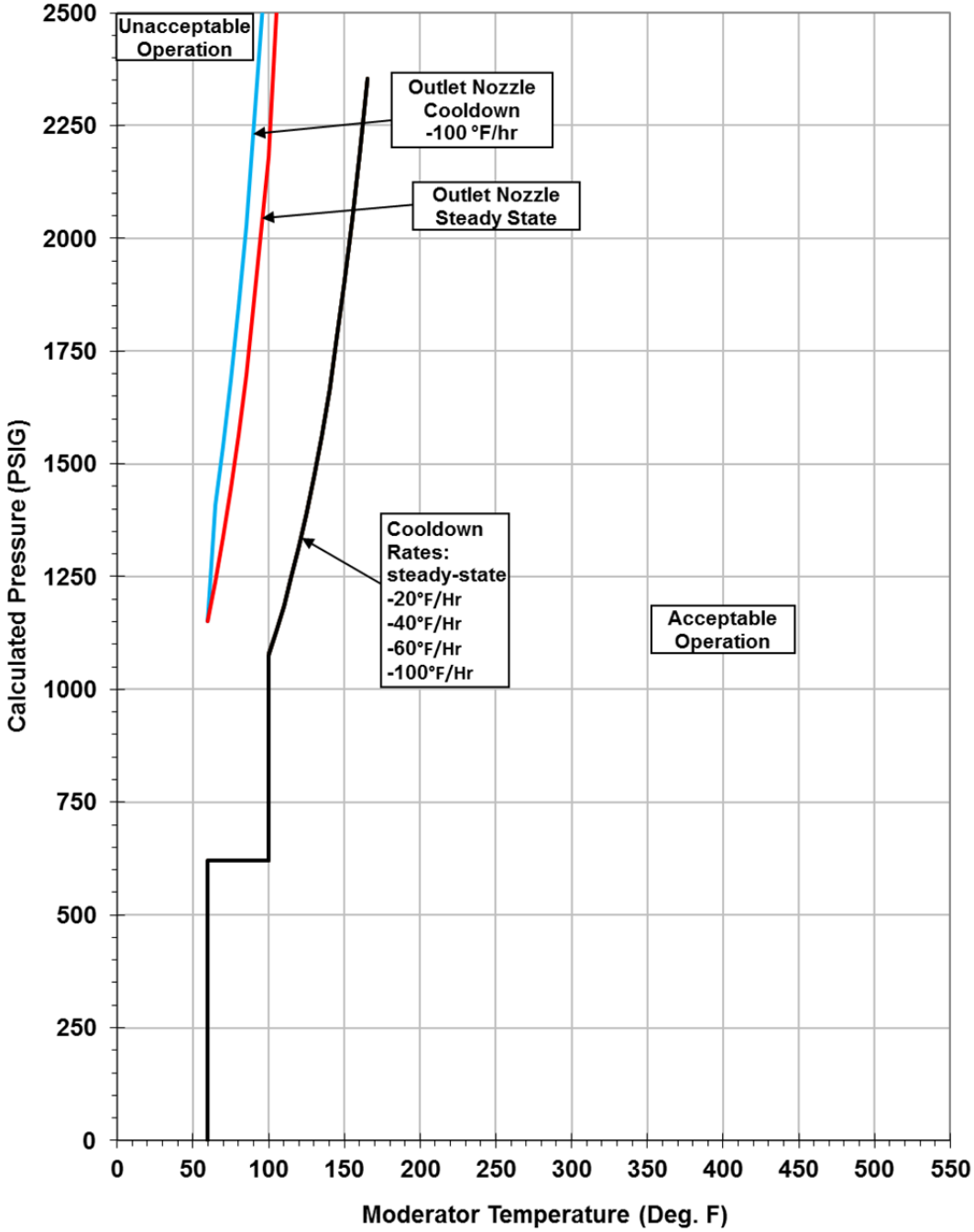

Figure B-2 Comparison of Watts Bar Unit 2 32 EFPY Beltline P-T Limits to 32 EFPY Outlet Nozzle P-T Limits, Without Margins for Instrumentation Errors

B.3 REFERENCES

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

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

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

B-4 NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Chapter 5 LWR Edition, Branch Technical Position 5-3, "Fracture Toughness Requirements," Revision 2, U.S. Nuclear Regulatory Commission, March 2007.

B-5 ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Division 1, Subsection NB, "Class 1 Components."

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

B-7 Oak Ridge National Laboratory Report, ORNL/TM-2010/246, "Stress and Fracture Mechanics Analyses of Boiling Water Reactor and Pressurized Water Reactor Pressure Vessel Nozzles – Revision 1," June 2012. *[ADAMS Accession Number ML110060164]*

B-8 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.

APPENDIX C OTHER RCPB FERRITIC COMPONENTS

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

The other ferritic RCPB components that are not part of the reactor vessel beltline or extended beltline consist of the reactor vessel closure head, pressurizer, steam generators, and future replacement steam generators.

The reactor vessel closure head materials have been considered in the development of P-T limits, see Section 6.3 of this report for further detail. Furthermore, the reactor vessel closure head was constructed to the 1971 Edition through 1971 Winter Addenda Section III ASME Code and has met all applicable requirements at the time of construction.

The original steam generators and pressurizer were constructed to the 1971 Edition through 1971 Summer Addenda Section III ASME Code and have met all applicable requirements at the time of construction. Furthermore, the steam generators and pressurizer have not undergone neutron embrittlement that would affect P-T limits. Therefore, no further consideration is necessary for this component with regards to P-T limits.

The future replacement steam generators have been designed to the 1989 Edition Section III ASME Code and should meet all applicable requirements at the time of construction. No further consideration is necessary for these components with regards to P-T limits.

C.1 REFERENCES

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

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

APPENDIX D UPPER-SHELF ENERGY EVALUATION

Charpy upper-shelf energy (USE) is associated with the determination of acceptable reactor pressure vessel (RPV) toughness during the licensed operating period.

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

There are two methods that can be used to predict the decrease in USE with irradiation, depending on the availability of credible surveillance capsule data as defined in Regulatory Guide 1.99, Revision 2 [Ref. D-2]. For vessel materials that are not in the surveillance program or have non-credible data, the Charpy USE (Position 1.2) is assumed to decrease as a function of fluence and copper content, as indicated in Regulatory Guide 1.99, Revision 2.

When two or more credible surveillance sets become available from the reactor, they may be used to determine the Charpy USE of the surveillance material. The surveillance data are then used in conjunction with the Regulatory Guide to predict the change in USE (Position 2.2) of the RPV material due to irradiation.

The 32 EFPY (EOL) Position 1.2 USE values of the vessel materials can be predicted using the corresponding 1/4T fluence projection, the copper content of the materials, and Figure 2 in Regulatory Guide 1.99, Revision 2.

The predicted Position 2.2 USE values are determined for the reactor vessel materials that are contained in the surveillance program by using the reduced plant surveillance data along with the corresponding 1/4T fluence projection. However, since no Watts Bar Unit 2 capsules have been tested, no Watts Bar Unit 2 surveillance data yet exists. Therefore, no percent USE reductions are calculated based on Regulatory Guide 1.99, Revision 2, Position 2.2.

The projected USE values were calculated to determine if the Watts Bar Unit 2 beltline and extended beltline materials remain above the 50 ft-lb criterion at 32 EFPY (EOL). These calculations are summarized in Table D-1.

USE Conclusion

For Watts Bar Unit 2, the limiting USE value at 32 EFPY is 72.0 ft-lb (see Table D-1); this value corresponds to Intermediate Shell Forging 05. Therefore, all of the beltline and extended beltline materials in the Watts Bar Unit 2 reactor vessel are projected to remain above the USE screening criterion value of 50 ft-lb (per 10 CFR 50, Appendix G) through EOL (32 EFPY).

Table D-1 Predicted Position 1.2 USE Values at 32 EFPY (EOL) for Watts Bar Unit 2

Reactor Vessel Material and Identification Number	Wt % Cu[(a)]	EOL 1/4T Fluence[(b)] (x 10^{19} n/cm^2, E > 1.0 MeV)	Initial USE[(a)] (ft-lb)	Projected USE Decrease[(c)] (%)	Projected EOL USE (ft-lb)
Reactor Vessel Beltline Materials					
Intermediate Shell Forging 05	0.05	1.120	90	20.0	72.0
Lower Shell Forging 04	0.05	1.138	105	20.0	84.0
Intermediate to Lower Shell Circumferential Weld Seam W05 (Heat # 895075)	0.05	1.120	127	20.0	101.6
Reactor Vessel Extended Beltline Materials					
Upper Shell Forging 06	0.07	0.0660	94	10.0	84.6
Upper to Intermediate Shell Circumferential Weld Seam W06 (Heat # 899680)	0.03	0.0660	92	10.0	82.8
Lower Shell to Bottom Head Ring Weld Seam W04 (Heat # 899680)	0.03	0.1477	92	12.0	81.0
Bottom Head Ring 03	0.06	0.1477	105	12.0	92.4

Notes:

(a) Data taken from Table 3-1 of this report.

(b) The 1/4T fluence was calculated using the Regulatory Guide 1.99, Revision 2 [Ref. D-2] correlation, and the Watts Bar Unit 2 reactor vessel beltline wall thickness of 8.465 inches.

(c) Percentage USE decrease values are based on Position 1.2 of Regulatory Guide 1.99, Revision 2 [Ref. D-2] and were calculated by plotting the 1/4T fluence values on Figure 2 of the Regulatory Guide and using the material-specific Cu wt. % values. In calculating Position 1.2 percent USE decreases, the base metal and weld Cu weight percentages were conservatively rounded up to the nearest line in Regulatory Guide 1.99, Revision 2, Figure 2.

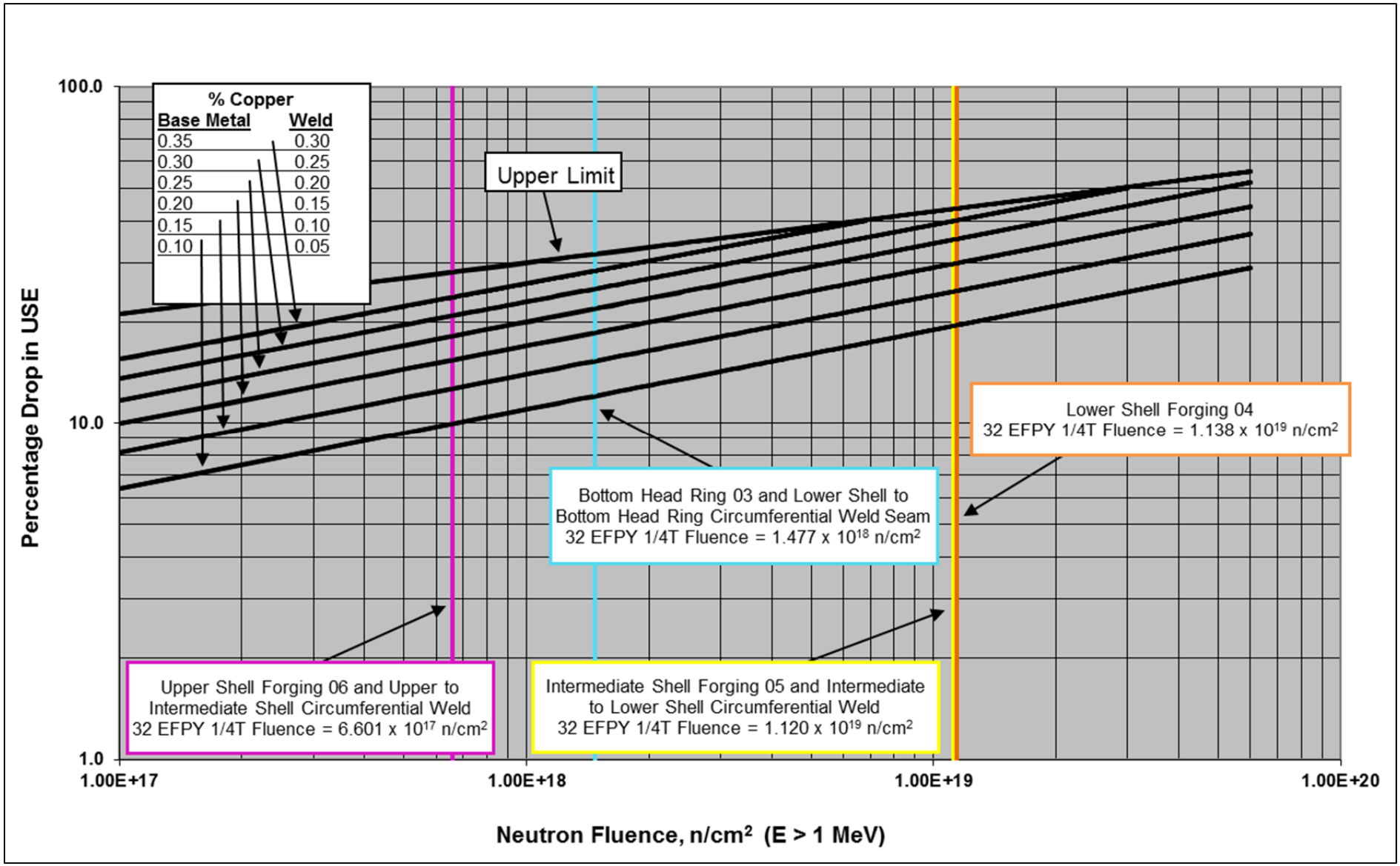

Figure D-1 Regulatory Guide 1.99, Revision 2 Predicted Decrease in Upper-Shelf Energy as a Function of Copper and Fluence

D.1 REFERENCES

D-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.

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.

APPENDIX E PRESSURIZED THERMAL SHOCK AND EMERGENCY RESPONSE GUIDELINE LIMITS EVALUATION

E.1 PRESSURIZED THERMAL SHOCK

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

In 1985, the U.S. NRC issued a formal ruling on PTS (10 CFR 50.61 [Ref. E-1]) that established screening criteria on Pressurized Water Reactor (PWR) vessel embrittlement, as measured by the maximum reference nil-ductility transition temperature in the limiting beltline component at the end of license, termed RT_{PTS}. RT_{PTS} screening values were set by the U.S. NRC for beltline axial welds, forgings or plates, and for beltline circumferential weld seams for plant operation to the end of plant license. All domestic PWR vessels have been required to evaluate vessel embrittlement in accordance with the criteria through the end of license. The U.S. NRC revised 10 CFR 50.61 in 1991 and 1995 to change the procedure for calculating radiation embrittlement. These revisions make the procedure for calculating the reference temperature for pressurized thermal shock (RT_{PTS}) values consistent with the methods given in Regulatory Guide 1.99, Revision 2 [Ref. E-2].

These accepted methods were used with the clad/base metal interface fluence values of Section 2 to calculate the following RT_{PTS} values for the Watts Bar Unit 2 RPV materials at 32 EFPY (EOL). The EOL RT_{PTS} calculations are summarized in Table E-1.

The following two conclusions from Section 4 of TLR-RES/DE/CIB-2013-01 [Ref. E-3] were utilized, as appropriate:

1. *The beltline is defined as the region of the RPV adjacent to the reactor core that is projected to receive a neutron fluence level of $1x10^{17}$ n/cm^2 (E > 1.0 MeV) or higher at the end of the licensed operating period.*
2. *Embrittlement effects may be neglected for any region of the RPV if either of the following conditions are met: (1) neutron fluence is less than $1x10^{17}$ n/cm^2 (E > 1.0 MeV) at EOL, or (2) the mean value of ΔT_{30} estimated using an ETC acceptable to the staff is less than 25°F at EOL. The estimate of ΔT_{30} at EOL shall be made using best-estimate chemistry values.*

Therefore, embrittlement of reactor vessel materials with ΔT_{30} (which is equivalent to ΔRT_{NDT}) values less than 25°F need not be considered in the subsequent RT_{PTS} calculations documented in Table E-1.

Table E-1 RT_{PTS} Calculations for the Watts Bar Unit 2 Reactor Vessel Materials at 32 EFPY

Reactor Vessel Material and Identification Number	Heat Number	CF[a] (°F)	EOL Fluence[a] (x 10^{19} n/cm^2, E > 1.0 MeV)	FF	$RT_{NDT(U)}$[a] (°F)	ΔRT_{NDT}[b] (°F)	σ_U[c] (°F)	σ_Δ[d] (°F)	Margin (°F)	RT_{PTS}[e] (°F)
Reactor Vessel Beltline Materials										
Intermediate Shell Forging 05	527828	31.0	1.861	1.170	14	36.3	0.0	17.0	34.0	84.3
Lower Shell Forging 04	528658	31.0	1.891	1.174	5	36.4	0.0	17.0	34.0	75.4
Intermediate to Lower Shell Circumferential Weld Seam W05	895075	68.0	1.861	1.170	-50	79.6	0.0	28.0	56.0	85.6
Using credible Catawba Unit 1, Watts Bar Unit 1, and McGuire Unit 2 surveillance data	895075	32.3	1.861	1.170	-50	37.8	0.0	14.0	28.0	15.8
Reactor Vessel Extended Beltline Materials										
Upper Shell Forging 06	411572	44.0	0.1097	0.4356	-14	0.0 (19.2)	0.0	0.0	0.0	-14
Upper to Intermediate Shell Circumferential Weld Seam W06	899680	41.0	0.1097	0.4356	10	0.0 (17.9)	0.0	0.0	0.0	10
Lower Shell to Bottom Head Ring Weld Seam W04	899680	41.0	0.2454	0.6194	10	25.4	0.0	12.7	25.4	60.8
Bottom Head Ring 03	5329	37.0	0.2454	0.6194	-40	0.0 (22.9)	0.0	0.0	0.0	-40

Notes:

(a) CF values were taken from Table 5-2, fluence values were taken from Tables 2-7 and 2-10, and $RT_{NDT(U)}$ values were taken from Table 3-1.

(b) Calculated ΔRT_{NDT} values less than 25°F have been reduced to zero per TLR-RES/DE/CIB-2013-01 [Ref. E-3]. Actual calculated ΔRT_{NDT} values are listed in parentheses for these materials.

(c) The initial RT_{NDT} values are measured values; therefore, σ_U = 0°F.

(d) The credibility conclusion for the surveillance material is discussed in Section 4. The sister-plant weld Heat # 895075 data was deemed credible. Per the guidance of 10 CFR 50.61 [Ref. E-1], the base metal σ_Δ = 17°F for Position 1.1. Also per 10 CFR 50.61 [Ref. E-1], the weld metal σ_Δ = 28°F for Position 1.1, and the weld metal σ_Δ = 14°F for Position 2.1 with credible surveillance data (Heat # 895075). However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$.

(e) The 10 CFR 50.61 [Ref. E-1] methodology was utilized in the calculation of the PTS values.

PTS Conclusion

The Watts Bar Unit 2 limiting RT_{PTS} value for base metal or longitudinal weld materials at 32 EFPY is 84.3°F (see Table E-1), which corresponds to Intermediate Shell Forging 05 (using Position 1.1). The Watts Bar Unit 2 limiting RT_{PTS} value for circumferentially oriented welds at 32 EFPY is 85.6°F (see Table E-1), which corresponds to the Intermediate to Lower Shell Circumferential Weld W05 (Heat # 895075, using Position 1.1). However, after applying credible sister-plant surveillance data, the RT_{PTS} value of this material is reduced to 15.8°F. Thus, the actual limiting RT_{PTS} value for circumferentially oriented welds at 32 EFPY is 60.8°F (see Table E-1), which corresponds to the Lower Shell to Bottom Head Ring Circumferential Weld W04 (Heat # 899680, using Position 1.1).

Therefore, all of the beltline and extended beltline materials in the Watts Bar Unit 2 reactor vessel are below the RT_{PTS} screening criteria of 270°F for base metal and/or longitudinal welds, and 300°F for circumferentially oriented welds through EOL (32 EFPY).

The Alternate PTS Rule (10 CFR 50.61a [Ref. E-4]) was published in the Federal Register by the NRC in 2010. This alternate rule is less restrictive than the PTS Rule (10 CFR 50.61) and is intended to be used for situations in which the 10 CFR 50.61 criteria cannot be met. Watts Bar Unit 2 currently meets the criteria for the PTS Rule through EOL and therefore does not need to utilize the Alternate PTS Rule at this time.

E.2 EMERGENCY RESPONSE GUIDELINE LIMITS

The Emergency Response Guideline (ERG) limits were developed to establish guidance for operator action in the event of an emergency situation, such as a PTS event [Ref. E-5]. Generic categories of limits were developed for the guidelines based on the limiting inside surface RT_{NDT}. These generic categories were conservatively generated for the Westinghouse Owners Group (WOG) to be applicable to all Westinghouse plants.

The highest value of RT_{NDT} for which the generic category ERG limits were developed is 250°F for a longitudinal flaw and 300°F for a circumferential flaw. Therefore, if the limiting vessel material has an RT_{NDT} that exceeds 250°F for a longitudinal flaw or 300°F for a circumferential flaw, plant-specific ERG P-T limits must be developed.

The ERG category is determined by the magnitude of the limiting RT_{NDT} value, which is calculated the same way as the RT_{PTS} values are calculated in Section E.1 of this report. The material with the highest RT_{NDT} defines the limiting material, which for Watts Bar Unit 2 is the Intermediate Shell Forging 05 (using Position 1.1). Table E-2 identifies ERG category limits and the limiting material RT_{NDT} values at 32 EFPY for Watts Bar Unit 2.

Table E-2 Evaluation of Watts Bar Unit 2 ERG Limit Category

ERG Pressure-Temperature Limits [Ref. E-5]	
Applicable RT_{NDT} Value[(a)]	**ERG P-T Limit Category**
RT_{NDT} < 200°F	Category I
200°F < RT_{NDT} < 250°F	Category II
250°F < RT_{NDT} < 300°F	Category III b
Limiting RT_{NDT} Value[(b)]	
Reactor Vessel Material	**RT_{NDT} Value @ 32 EFPY**
Intermediate Shell Forging 05 (Position 1.1)	84.3°F

Notes:

(a) Longitudinally oriented flaws are applicable only up to 250°F; circumferentially oriented flaws are applicable up to 300°F.

(b) Value taken from Table E-1.

Per the ERG limit guidance document [Ref. E-5], some vessels do not change categories for operation through the end of license. However, when a vessel does change ERG categories between the beginning and end of operation, a plant-specific assessment must be performed to determine at what operating time the category changes. Thus, the ERG classification need not be changed until the operating cycle during which the maximum vessel value of actual or estimated real-time RT_{NDT} exceeds the limit on its current ERG category.

Per Table E-2, the limiting material for Watts Bar Unit 2 (Intermediate Shell Forging 05 [Position 1.1]) has an RT_{NDT} less than 200°F through 32 EFPY. Therefore, Watts Bar Unit 2 remains in ERG Category I through EOL (32 EFPY).

Conclusion of ERG P-T Limit Categorization

As summarized above, Watts Bar Unit 2 is currently in ERG Category I and will remain in ERG Category Unit I through EOL (32 EFPY).

E.3 REFERENCES

E-1 Code of Federal Regulations 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," Federal Register, Volume 60, No. 243, dated December 19, 1995, effective January 18, 1996.

E-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.

E-3 U.S. NRC Technical Letter Report TLR-RES/DE/CIB-2013-01, "Evaluation of the Beltline Region for Nuclear Reactor Pressure Vessels," Office of Nuclear Regulatory Research [RES], November 2014. *[ADAMS Accession Number ML14318A177]*

E-4 Code of Federal Regulations, 10 CFR 50.61a, "Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," Federal Register, Volume 75, No. 1, dated January 4, 2010, with corrections dated February 3, 2010 (No. 22), March 8, 2010 (No. 44), and November 26, 2010 (No. 227).

E-5 Westinghouse Owners Group Document HF04BG, "Background Information for Westinghouse Owners Group Emergency Response Guidelines, Critical Safety Function Status Tree, F-0.4 Integrity, HP/LP-Rev. 2," April 2005.

APPENDIX F SURVEILLANCE CAPSULE WITHDRAWAL SCHEDULE

The following surveillance capsule removal schedule (Table F-1) meets the recommendations of ASTM E185-82 [Ref. F-1] as required by 10 CFR 50, Appendix H [Ref. F-2]. Note that it is recommended for future capsules to be removed from the Watts Bar Unit 2 reactor vessel.

Table F-1 Surveillance Capsule Withdrawal Schedule

Capsule ID	Capsule Location	Capsule Lead Factor[(a)]	Withdrawal EFPY[(b)]	Capsule Fluence (n/cm^2, E > 1.0 MeV)
U	Dual 34°	4.80	2.61 (EOC 2)	0.7714×10^{19}
W	Single 34°	4.87	6.91 (EOC 5)	1.901×10^{19}
X	Dual 34°	4.80	Note (c)	Note (c)
Z	Single 34°	4.87	Note (d)	Note (d)
V	Dual 31.5°	4.15	Note (d)	Note (d)
Y	Dual 31.5°	4.15	Note (d)	Note (d)

Notes:

(a) Projected capsule fluence values are documented in Table 2-3. The lead factor is the proportional constant by which the capsule fluence leads the vessel's maximum inner wall fluence. This value was calculated using the maximum EOL vessel fluence value and the capsule EOL fluence values documented in Table 2-5 and Table 2-3, respectively.

(b) EFPY from plant startup.

(c) Capsule X must be withdrawn between EOC 6 and 13.5 EFPY in order satisfy the recommendations of the third capsule for EOL per ASTM E185-82 [Ref. F-1]. However, if the capsule is removed after 11.6 EFPY (but still before 13.5 EFPY), this capsule will satisfy the requirements of the third capsule for both EOL (40 years) and end of license extension (60 years) per ASTM E185-82 [Ref. F-1] and NUREG-1801, Revision 2 [Ref. F-3]. Thus, if possible, the capsule should be pulled between 11.6 EFPY and 13.5 EFPY, but the capsule must be pulled between EOC 6 and 13.5 EFPY. The removal EFPY of the third capsule should be revisited at a later date, such as after Capsules U and W are removed.

(d) Capsules Z, V, and Y should remain in the reactor. If additional metallurgical data is needed for Watts Bar Unit 2, such as in support of a second license renewal to 80 total years of operation, withdrawal and testing of these capsules should be considered.

F.1 REFERENCES

F-1 ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," ASTM, July 1982.

F-2 Code of Federal Regulations 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," Federal Register, Volume 60, No. 243, dated December 19, 1995.

F-3 U.S. Nuclear Regulatory Commission, "Generic Aging Lessons Learned (GALL) Report," NUREG-1801, Revision 2, December 2010. *[ADAMS Accession Number ML103490041]*