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Arkansas Nuclear One Unit 2 Heatup and Cooldown Limit Curves for Normal Operation

Westinghouse

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Benjamin E. Mays*
Structural Design & Analysis III

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Reviewers: D. Brett Lynch*
Structural Design & Analysis III

Eugene T. Hayes*
Radiation Engineering & Analysis

Approved: Lynn A. Patterson*, Manager
Structural Design & Analysis III

Laurent P. Houssay*, Manager
Radiation Engineering & Analysis

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

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

RECORD OF REVISION

Revision 0: Original Issue

Revision 1: Revised issue. The purpose of this revision is to remove utilization of the TLR-RES/DE/CIB-2013-01 report methodology. Therefore, calculated ΔRT_{NDT} values less than or equal to 25°F will not be reduced to zero. The pressure-temperature (P-T) limit curves are not affected by the changes. Changes are indicated with change bars.

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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 Arkansas Nuclear One Unit 2 reactor vessel. The heatup and cooldown P-T limit curves were generated using the limiting Adjusted Reference Temperature (ART) values for Arkansas Nuclear One Unit 2. The limiting ART values were those of Lower Shell Plate C-8010-1 (Position 1.1) at both 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations. The P-T limit curves were generated 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 curves were generated for 54 effective full-power years (EFPY) using heatup rates of 50, 60, 70, and 80°F/hr, and cooldown rates of 0 (steady-state), -25, -60, and -100°F/hr. The curves were developed with the flange and lowest service temperature (LST) requirements and without margins for instrumentation errors. They 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 54 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 vessel at the nozzle corner region. As discussed in Appendix B, the P-T limit curves generated based on the limiting cylindrical beltline material (Lower Shell Plate C-8010-1) bound the P-T limit curves for the reactor vessel inlet and outlet nozzles for Arkansas Nuclear One Unit 2 at 54 EFPY.

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

Appendix D contains a credibility evaluation for weld Heat # 10137 considering all applicable sister plant surveillance program data.

1 INTRODUCTION

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} 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. For the purposes of this report, heatup is defined as the process of heating the reactor coolant system (RCS) from ambient temperature to operating temperature. Cooldown is defined as the process of cooling the RCS from operating temperature to ambient temperature. Steady-state is defined as a 0°F/hr cooldown or heatup rate. Under steady-state, the thermal stress intensity factor is considered to be zero. The steady-state curve is necessary from an engineering perspective for comparison with heatup and cooldown.

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} ($RT_{NDT(U)}$). 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 Adjusted Reference Temperature (ART) values ($RT_{NDT(U)}$ + ΔRT_{NDT} + margins for uncertainties) at the 1/4T and 3/4T locations, where T is the thickness of the vessel at the beltline region measured from the clad/base metal interface.

The 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]. 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 purpose of this report is to present the calculations and the development of the Arkansas Nuclear One Unit 2 heatup and cooldown P-T limit curves for 54 EFPY. This report documents the calculated ART values and the development of the P-T limit curves for normal operation. The calculated ART values for 54 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 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, along with the lowest service temperature (LST) requirements of ASME Code, Section III [Ref. 9]. As discussed in Appendix B, the P-T limit curves generated in Section 8 bound the P-T limit curves for the reactor vessel inlet and outlet nozzles for Arkansas Nuclear One Unit 2 at 54 EFPY. Discussion of the other non-reactor vessel ferritic RCPB components relative to P-T limits is contained in Appendix C.

2 CALCULATED NEUTRON FLUENCE

2.1 INTRODUCTION

A discrete ordinates (S_N) transport analysis was performed for the Arkansas Nuclear One Unit 2 reactor to determine the neutron radiation environment within the reactor pressure vessel. In this analysis, radiation exposure parameters were established on a plant- and fuel-cycle-specific basis. An evaluation of the dosimetry sensor sets from the 284° and 97° surveillance capsules is provided in WCAP-18166-NP [Ref. 21]. The dosimetry analysis documented in WCAP-18166-NP showed that the ±20% (1σ) acceptance criterion specified in Regulatory Guide 1.190 [Ref. 5] is met. The validated calculations form the basis for providing projections of the neutron exposure of the reactor pressure vessel for operating periods extending to 54 EFPY.

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, the neutron transport evaluation methodologies follow the guidance of Regulatory Guide 1.190 [Ref. 5]. Additionally, the methods used to develop the calculated pressure vessel fluence are consistent with the NRC-approved methodology described in WCAP-14040-A, Revision 4 [Ref. 2].

2.2 DISCRETE ORDINATES ANALYSIS

In performing the fast neutron exposure evaluations for the Arkansas Nuclear One Unit 2 reactor vessel, a series of fuel-cycle-specific forward transport calculations were performed using the following three-dimensional fluence rate synthesis technique:

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

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

For the Arkansas Nuclear One Unit 2 transport calculations, the r,θ model depicted in Figure 2-1 and Figure 2-2 were utilized since, with the exception of the capsules, the reactor is octant symmetric. These r,θ models included the core, the reactor internals, octants with surveillance capsules at 7° and 14° and octants without surveillance capsules, the pressure vessel cladding and vessel wall, the insulation external to the pressure vessel, and the primary biological shield wall. These models formed the basis for the calculated results and enabled making comparisons to the surveillance capsule dosimetry evaluations. Specifically, the r,θ model with surveillance capsules was utilized to perform capsule dosimetry evaluations and subsequent comparisons with calculated results, while the r,θ model without surveillance capsules was used to generate the maximum fluence levels at the pressure vessel wall. In developing these analytical models, nominal design dimensions were generally employed for the various structural components. Note that for the pressure vessel inner radius, however, the average of the as-built inner radii

was used. In addition, 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 reactor core itself was treated as a homogeneous mixture of fuel, cladding, water, and miscellaneous core structures such as fuel assembly grids, guide tubes, et cetera. The geometric mesh description of the r,θ reactor model in Figure 2-1 consisted of 166 radial by 123 azimuthal intervals. The geometric mesh description of the r,θ reactor model in Figure 2-2 consisted of 166 radial by 116 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 fluence rate convergence criterion utilized in the r,θ calculations was set at a value of 0.001.

The r,z model used for the Arkansas Nuclear One Unit 2 calculations is shown in Figure 2-3. The model extends radially from the centerline of the reactor core out to the primary biological shield and axially from an elevation approximately 4.9 feet below to 6 feet above the active core. As in the case of the r,θ models, nominal design dimensions, with the exception of the pressure vessel inner radius, and full-power coolant densities were employed in the calculations. In the r,z model, the homogenous core region was treated as an equivalent cylinder with a volume equal to that of the active core zone. The stainless steel girth ribs located between the core shroud and core barrel regions were also explicitly included in the model. The geometric mesh description of the r,z reactor model in Figure 2-3 consisted of 161 radial by 222 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 fluence rate convergence criterion utilized in the r,z calculations was set at a value of 0.001.

The one-dimensional radial model used in the synthesis procedure consisted of the same 161 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 25 fuel cycles at Arkansas Nuclear One Unit 2 included cycle-dependent fuel assembly initial enrichments, burnups, and axial power distributions (note that Cycles 1–24 have been completed; Cycle 25 is based on the expected core design for this cycle and an assumed cycle length of 1.37 EFPY). 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 supporting this analysis were performed using the DORT discrete ordinates code [Ref. 23] and the BUGLE-96 cross-section library [Ref. 7]. The BUGLE-96 library provides a 67-group coupled neutron-gamma ray cross-section data set produced specifically for light-water reactor (LWR) applications. In these analyses, anisotropic scattering was treated with a P_5 Legendre expansion and angular discretization was modeled with an S_{16} order of angular quadrature. Energy- and space-dependent core power distributions, as well as system operating temperatures, were treated on a fuel-cycle-specific basis.

The locations of the Arkansas Nuclear One Unit 2 vessel welds and plates are provided in Table 2-1. The axial position of each material is indexed to z = 0.0 cm, which corresponds to the midplane of the active fuel stack.

These data tabulations include both plant- and fuel-cycle-specific calculated neutron exposures at the end of Cycle 24, at the end of projected Cycle 25, and at further projections to 54 EFPY. The calculations account for the uprate from 2815 MWt to 3026 MWt that occurred at the beginning of Cycle 16. The projections are based on the assumption that the core power distributions and associated plant operating characteristics from Cycle 23, Cycle 24, and the design of Cycle 25 are representative of future plant operation. The future projections are based on the current reactor power level of 3026 MWt.

Selected results from the neutron transport analyses are provided in Table 2-2 through Table 2-7. In Table 2-2, the calculated maximum fast neutron (E > 1.0 MeV) fluence values for the reactor pressure vessel materials are provided at future projections to 32, 36, 40, 48 and 54 EFPY. The projections are based on the assumption that the core power distributions and associated plant operating characteristics from Cycles 23–25 are representative of future plant operation. In Table 2-3, the calculated maximum iron atom displacement values for the reactor pressure vessel materials are provided at future projections to 32, 36, 40, 48 and 54 EFPY.

The calculated fast neutron (E > 1.0 MeV) fluence rate, fast neutron (E > 1.0 MeV) fluence, iron atom displacement rate, and iron atom displacements are provided in Table 2-4 through Table 2-7, respectively, for the reactor pressure vessel inner radius at four azimuthal locations, as well as the maximum exposure observed within the octant. The vessel data given in Table 2-4 through Table 2-7 were taken at the clad/base metal interface and represent maximum calculated exposure levels on the vessel.

2.3 CALCULATIONAL UNCERTAINTIES

The uncertainty associated with the calculated neutron exposure of the Arkansas Nuclear One Unit 2 reactor pressure vessel materials is based on the recommended approach provided in Regulatory Guide 1.190. In particular, the qualification of the methodology was carried out in the following four stages:

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

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

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

4. Comparisons of the plant-specific calculations with all available dosimetry results from the Arkansas Nuclear One Unit 2 surveillance program.

The first phase of the methods qualification (PCA comparisons) addressed the adequacy of basic transport calculation and dosimetry evaluation techniques and associated cross-sections. This phase, however, did not test the accuracy of commercial core neutron source calculations nor did it address uncertainties in operational or geometric variables that impact power reactor calculations.

The second phase of the qualification (H.B. Robinson comparisons) addressed uncertainties in these additional areas that are primarily methods related and would tend to apply generically to all fast neutron exposure evaluations.

The third phase of the qualification (analytical sensitivity study) identified the potential uncertainties introduced into the overall evaluation due to calculational methods approximations as well as to a lack of knowledge relative to various plant-specific input parameters. The overall calculational uncertainty applicable to the Arkansas Nuclear One Unit 2 analysis was established from results of these three phases of the methods qualification.

The fourth phase of the uncertainty assessment (comparisons with Arkansas Nuclear One Unit 2 measurements) was used solely to demonstrate the validity of the transport calculations and to confirm the uncertainty estimates associated with the analytical results. The comparison was used only as a check and was not used in any way to modify the calculated surveillance capsule or pressure vessel neutron exposures.

Table 2-8 summarizes the uncertainties developed from the first three phases of the methodology qualification. Additional information pertinent to these evaluations is provided in Reference 6. The net calculational uncertainty was determined by combining the individual components in quadrature. Therefore, the resultant uncertainty was treated as random and no systematic bias was applied to the analytical results. The plant-specific measurement comparisons given in Appendix A of Reference 21 support these uncertainty assessments for Arkansas Nuclear One Unit 2.

Table 2-1 Pressure Vessel Material Locations

Material	Axial Location Relative to Core Midplane at 0 cm (cm)	Azimuthal Location (degrees)
Inlet Nozzle to Upper Shell Welds – Lowest Extent		
Nozzle 1	301.625(a)	60
Nozzle 2	301.625(a)	120
Nozzle 3	301.625(a)	240
Nozzle 4	301.625(a)	300
Outlet Nozzle to Upper Shell Welds – Lowest Extent		
Nozzle 1	301.625(a)	0
Nozzle 2	301.625(a)	180
Upper Shell to Intermediate Shell Circumferential Weld		
8-203	248.722	0 to 360
Intermediate Shell Plates		
C-8009-1, -2, -3	-33.973 to 248.722	0 to 360(b)
Intermediate Shell Longitudinal Welds		
2-203 A	-33.973 to 248.722	90
2-203 B	-33.973 to 248.722	210
2-203 C	-33.973 to 248.722	330
Intermediate Shell to Lower Shell Circumferential Weld		
9-203	-33.973	0 to 360
Lower Shell Plates		
C-8010-1, -2, -3	-315.892 to -33.973	0 to 360(c)
Lower Shell Longitudinal Welds		
3-203 A	-315.892 to -33.973	90
3-203 B	-315.892 to -33.973	210
3-203 C	-315.892 to -33.973	330
Lower Shell to Bottom Head Circumferential Weld		
10-203	-315.892	0 to 360

Notes:

(a) This axial location corresponds to the bottom of the vessel support pad of the inlet nozzle, instead of the nozzle to upper shell weld. This provides a bounding fluence for the nozzle to upper shell weld.

(b) Intermediate shell plates C-8009-1, -2, and -3 extend from azimuthal angles of 330° to 90°, 90° to 210°, and 210° to 330°, respectively.

(c) Lower shell plates C-8010-1, -2, and -3 extend from azimuthal angles of 210° to 330°, 330° to 90°, and 90° to 210°, respectively.

Table 2-2 Calculated Maximum Fast Neutron (E > 1.0 MeV) Fluence of the Pressure Vessel Clad/Base Metal Interface

Material	Fluence[b] (n/cm^2)				
	32 EFPY	36 EFPY	40 EFPY	48 EFPY	54 EFPY
Inlet Nozzle to Upper Shell Welds – Lowest Extent					
Nozzle 1[a]	4.78E+16	5.36E+16	5.93E+16	7.09E+16	7.96E+16
Nozzle 2[a]	4.78E+16	5.36E+16	5.93E+16	7.09E+16	7.96E+16
Nozzle 3[a]	4.78E+16	5.36E+16	5.93E+16	7.09E+16	7.96E+16
Nozzle 4[a]	4.78E+16	5.36E+16	5.93E+16	7.09E+16	7.96E+16
Outlet Nozzle to Upper Shell Welds – Lowest Extent					
Nozzle 1[a]	6.05E+16	6.73E+16	7.41E+16	8.77E+16	9.80E+16
Nozzle 2[a]	6.05E+16	6.73E+16	7.41E+16	8.77E+16	9.80E+16
Upper Shell to Intermediate Shell Circumferential Weld					
8-203	3.53E+17	3.96E+17	4.38E+17	5.24E+17	5.89E+17
Intermediate Shell Plates					
C-8009-1, -2, -3	3.02E+19	3.36E+19	3.70E+19	4.39E+19	4.91E+19
Intermediate Shell Longitudinal Welds					
2-203 A	2.89E+19	3.21E+19	3.53E+19	4.16E+19	4.64E+19
2-203 B	2.22E+19	2.48E+19	2.75E+19	3.28E+19	3.68E+19
2-203 C	2.22E+19	2.48E+19	2.75E+19	3.28E+19	3.68E+19
Intermediate Shell to Lower Shell Circumferential Weld					
9-203	3.00E+19	3.35E+19	3.69E+19	4.38E+19	4.89E+19
Lower Shell Plates					
C-8010-1, -2, -3	3.02E+19	3.37E+19	3.73E+19	4.44E+19	4.98E+19
Lower Shell Longitudinal Welds					
3-203 A	2.89E+19	3.22E+19	3.55E+19	4.21E+19	4.71E+19
3-203 B	2.22E+19	2.49E+19	2.77E+19	3.32E+19	3.73E+19
3-203 C	2.22E+19	2.49E+19	2.77E+19	3.32E+19	3.73E+19
Lower Shell to Bottom Head Circumferential Weld					
10-203	5.41E+16	6.06E+16	6.71E+16	8.01E+16	8.98E+16

Notes:

(a) The axial location used corresponds to the bottom of the vessel support pad of the inlet nozzle, instead of the nozzle to upper shell weld. This provides a bounding fluence for the nozzle to upper shell weld.

(b) Values are based on the average power distributions and core operating conditions of Cycles 23–25.

Table 2-3 Calculated Maximum Iron Atom Displacements at the Pressure Vessel Clad/Base Metal Interface

Material	Iron Atom Displacements[b] (dpa)				
	32 EFPY	36 EFPY	40 EFPY	48 EFPY	54 EFPY
Inlet Nozzle to Upper Shell Welds – Lowest Extent					
Nozzle 1[a]	2.83E-04	3.17E-04	3.51E-04	4.19E-04	4.70E-04
Nozzle 2[a]	2.83E-04	3.17E-04	3.51E-04	4.19E-04	4.70E-04
Nozzle 3[a]	2.83E-04	3.17E-04	3.51E-04	4.19E-04	4.70E-04
Nozzle 4[a]	2.83E-04	3.17E-04	3.51E-04	4.19E-04	4.70E-04
Outlet Nozzle to Upper Shell Welds – Lowest Extent					
Nozzle 1[a]	3.37E-04	3.75E-04	4.14E-04	4.91E-04	5.48E-04
Nozzle 2[a]	3.37E-04	3.75E-04	4.14E-04	4.91E-04	5.48E-04
Upper Shell to Intermediate Shell Circumferential Weld					
8-203	6.41E-04	7.18E-04	7.95E-04	9.50E-04	1.07E-03
Intermediate Shell Plates					
C-8009-1, -2, -3	4.59E-02	5.11E-02	5.64E-02	6.69E-02	7.47E-02
Intermediate Shell Longitudinal Welds					
2-203 A	4.42E-02	4.90E-02	5.39E-02	6.36E-02	7.09E-02
2-203 B	3.39E-02	3.80E-02	4.20E-02	5.02E-02	5.63E-02
2-203 C	3.39E-02	3.80E-02	4.20E-02	5.02E-02	5.63E-02
Intermediate Shell to Lower Shell Circumferential Weld					
9-203	4.57E-02	5.10E-02	5.62E-02	6.67E-02	7.45E-02
Lower Shell Plates					
C-8010-1, -2, -3	4.59E-02	5.13E-02	5.67E-02	6.76E-02	7.58E-02
Lower Shell Longitudinal Welds					
3-203 A	4.41E-02	4.92E-02	5.42E-02	6.43E-02	7.19E-02
3-203 B	3.39E-02	3.81E-02	4.23E-02	5.08E-02	5.71E-02
3-203 C	3.39E-02	3.81E-02	4.23E-02	5.08E-02	5.71E-02
Lower Shell to Bottom Head Circumferential Weld					
10-203	2.85E-04	3.19E-04	3.53E-04	4.22E-04	4.73E-04

Notes:

(a) The axial location used corresponds to the bottom of the vessel support pad of the inlet nozzle, instead of the nozzle to upper shell weld. This provides a bounding fluence for the nozzle to upper shell weld.

(b) Values are based on the average power distributions and core operating conditions of Cycles 23–25.

Table 2-4 Calculated Azimuthal Variation of the Maximum Fast Neutron (E > 1.0 MeV) Fluence Rate at the Reactor Vessel Clad/Base Metal Interface

Cycle	Cycle Length (EFPY)	Cumulative Operating Time (EFPY)	Fluence Rate (n/cm^2-s)				
			0°	15°	30°	45°	Maximum
1	0.89	0.89	3.62E+10	3.60E+10	2.71E+10	2.63E+10	3.81E+10
2	0.80	1.69	4.59E+10	4.51E+10	3.38E+10	3.22E+10	4.80E+10
3	0.64	2.33	4.34E+10	4.21E+10	3.18E+10	3.12E+10	4.52E+10
4	0.97	3.31	4.36E+10	4.12E+10	3.34E+10	3.33E+10	4.46E+10
5	0.85	4.16	4.68E+10	4.52E+10	3.34E+10	3.30E+10	4.86E+10
6	1.22	5.38	3.40E+10	3.40E+10	2.54E+10	2.34E+10	3.64E+10
7	1.13	6.51	3.17E+10	3.03E+10	2.39E+10	2.20E+10	3.30E+10
8	1.15	7.66	3.25E+10	3.20E+10	2.42E+10	2.17E+10	3.44E+10
9	1.18	8.84	3.20E+10	3.15E+10	2.38E+10	2.09E+10	3.38E+10
10	1.32	10.16	2.98E+10	2.98E+10	2.35E+10	2.26E+10	3.19E+10
11	1.33	11.49	2.36E+10	2.06E+10	1.73E+10	1.83E+10	2.36E+10
12	1.31	12.81	2.41E+10	2.30E+10	1.77E+10	1.75E+10	2.48E+10
13	1.47	14.27	2.29E+10	1.99E+10	1.63E+10	1.78E+10	2.29E+10
14	1.41	15.69	2.35E+10	2.28E+10	1.79E+10	1.78E+10	2.42E+10
15	1.29	16.98	2.31E+10	2.23E+10	1.90E+10	1.85E+10	2.34E+10
16	1.35	18.33	2.73E+10	2.49E+10	1.90E+10	1.92E+10	2.75E+10
17	1.36	19.69	2.56E+10	2.47E+10	1.96E+10	1.89E+10	2.63E+10
18	1.43	21.12	2.77E+10	2.83E+10	2.21E+10	2.13E+10	2.95E+10
19	1.34	22.46	2.71E+10	2.72E+10	2.09E+10	2.06E+10	2.85E+10
20	1.36	23.82	2.72E+10	2.72E+10	2.10E+10	2.11E+10	2.85E+10
21	1.35	25.17	2.72E+10	2.72E+10	2.12E+10	2.12E+10	2.86E+10
22	1.45	26.61	2.59E+10	2.59E+10	2.03E+10	2.05E+10	2.72E+10
23	1.36	27.98	2.50E+10	2.68E+10	2.13E+10	2.16E+10	2.76E+10
24	1.26	29.24	2.40E+10	2.57E+10	2.14E+10	2.20E+10	2.64E+10
25[(a)]	1.37	30.60	2.97E+10	2.96E+10	2.29E+10	2.32E+10	3.11E+10

Note:

(a) Cycle 25 is the current operating cycle. Values listed for this cycle are projections based on the Cycle 25 design.

Table 2-5 Calculated Azimuthal Variation of the Maximum Fast Neutron (E > 1.0 MeV) Fluence at the Reactor Vessel Clad/Base Metal Interface

Cycle	Cycle Length (EFPY)	Cumulative Operating Time (EFPY)	Fluence (n/cm^2)				
			0°	15°	30°	45°	Maximum
1	0.89	0.89	1.02E+18	1.01E+18	7.61E+17	7.37E+17	1.07E+18
2	0.80	1.69	2.16E+18	2.14E+18	1.61E+18	1.54E+18	2.27E+18
3	0.64	2.33	3.04E+18	2.99E+18	2.25E+18	2.17E+18	3.18E+18
4	0.97	3.31	4.38E+18	4.26E+18	3.27E+18	3.19E+18	4.55E+18
5	0.85	4.16	5.64E+18	5.48E+18	4.17E+18	4.08E+18	5.86E+18
6	1.22	5.38	6.95E+18	6.78E+18	5.15E+18	4.98E+18	7.26E+18
7	1.13	6.51	8.08E+18	7.86E+18	6.00E+18	5.77E+18	8.44E+18
8	1.15	7.66	9.26E+18	9.02E+18	6.88E+18	6.56E+18	9.68E+18
9	1.18	8.84	1.04E+19	1.02E+19	7.76E+18	7.33E+18	1.09E+19
10	1.32	10.16	1.17E+19	1.14E+19	8.74E+18	8.27E+18	1.23E+19
11	1.33	11.49	1.27E+19	1.23E+19	9.47E+18	9.04E+18	1.32E+19
12	1.31	12.81	1.37E+19	1.32E+19	1.02E+19	9.76E+18	1.43E+19
13	1.47	14.27	1.47E+19	1.42E+19	1.09E+19	1.06E+19	1.53E+19
14	1.41	15.69	1.57E+19	1.51E+19	1.17E+19	1.14E+19	1.63E+19
15	1.29	16.98	1.67E+19	1.60E+19	1.25E+19	1.21E+19	1.73E+19
16	1.35	18.33	1.78E+19	1.71E+19	1.33E+19	1.29E+19	1.84E+19
17	1.36	19.69	1.89E+19	1.81E+19	1.41E+19	1.37E+19	1.95E+19
18	1.43	21.12	2.01E+19	1.94E+19	1.51E+19	1.46E+19	2.08E+19
19	1.34	22.46	2.12E+19	2.05E+19	1.59E+19	1.55E+19	2.20E+19
20	1.36	23.82	2.23E+19	2.16E+19	1.68E+19	1.64E+19	2.32E+19
21	1.35	25.17	2.35E+19	2.27E+19	1.77E+19	1.72E+19	2.43E+19
22	1.45	26.61	2.46E+19	2.39E+19	1.86E+19	1.81E+19	2.55E+19
23	1.36	27.98	2.57E+19	2.50E+19	1.95E+19	1.91E+19	2.67E+19
24	1.26	29.24	2.66E+19	2.60E+19	2.03E+19	1.99E+19	2.77E+19
25[(a)]	1.37	30.60	2.78E+19	2.72E+19	2.12E+19	2.08E+19	2.90E+19
Future[(b)]		32.00	2.89E+19	2.84E+19	2.22E+19	2.18E+19	3.02E+19
Future[(b)]		36.00	3.22E+19	3.18E+19	2.49E+19	2.46E+19	3.37E+19
Future[(b)]		40.00	3.55E+19	3.53E+19	2.77E+19	2.74E+19	3.73E+19
Future[(b)]		48.00	4.21E+19	4.22E+19	3.32E+19	3.30E+19	4.44E+19
Future[(b)]		54.00	4.71E+19	4.74E+19	3.73E+19	3.72E+19	4.98E+19

Notes:

(a) Cycle 25 is the current operating cycle. Values listed for this cycle are projections based on the Cycle 25 design.

(b) Values beyond Cycle 25 are based on the average power distributions and core operating conditions of Cycles 23–25.

Table 2-6 Calculated Iron Atom Displacement Rate at the Pressure Vessel Clad/Base Metal Interface

Cycle	Cycle Length (EFPY)	Cumulative Operating Time (EFPY)	Iron Atom Displacement Rate (dpa/s)				
			0°	15°	30°	45°	Maximum
1	0.89	0.89	5.52E-11	5.47E-11	4.14E-11	4.01E-11	5.79E-11
2	0.80	1.69	6.99E-11	6.86E-11	5.16E-11	4.91E-11	7.29E-11
3	0.64	2.33	6.62E-11	6.40E-11	4.85E-11	4.76E-11	6.87E-11
4	0.97	3.31	6.65E-11	6.27E-11	5.10E-11	5.08E-11	6.78E-11
5	0.85	4.16	7.13E-11	6.87E-11	5.10E-11	5.04E-11	7.39E-11
6	1.22	5.38	5.20E-11	5.17E-11	3.88E-11	3.58E-11	5.53E-11
7	1.13	6.51	4.84E-11	4.61E-11	3.66E-11	3.37E-11	5.02E-11
8	1.15	7.66	4.97E-11	4.86E-11	3.69E-11	3.33E-11	5.24E-11
9	1.18	8.84	4.89E-11	4.79E-11	3.64E-11	3.21E-11	5.15E-11
10	1.32	10.16	4.56E-11	4.54E-11	3.59E-11	3.45E-11	4.84E-11
11	1.33	11.49	3.60E-11	3.14E-11	2.65E-11	2.80E-11	3.60E-11
12	1.31	12.81	3.68E-11	3.51E-11	2.71E-11	2.68E-11	3.78E-11
13	1.47	14.27	3.50E-11	3.03E-11	2.50E-11	2.73E-11	3.50E-11
14	1.41	15.69	3.59E-11	3.48E-11	2.73E-11	2.73E-11	3.69E-11
15	1.29	16.98	3.53E-11	3.41E-11	2.91E-11	2.83E-11	3.57E-11
16	1.35	18.33	4.17E-11	3.80E-11	2.91E-11	2.95E-11	4.19E-11
17	1.36	19.69	3.91E-11	3.76E-11	3.01E-11	2.89E-11	4.01E-11
18	1.43	21.12	4.23E-11	4.31E-11	3.38E-11	3.26E-11	4.49E-11
19	1.34	22.46	4.13E-11	4.14E-11	3.20E-11	3.16E-11	4.34E-11
20	1.36	23.82	4.15E-11	4.13E-11	3.21E-11	3.23E-11	4.34E-11
21	1.35	25.17	4.16E-11	4.14E-11	3.24E-11	3.25E-11	4.35E-11
22	1.45	26.61	3.95E-11	3.95E-11	3.10E-11	3.13E-11	4.14E-11
23	1.36	27.98	3.82E-11	4.08E-11	3.26E-11	3.29E-11	4.20E-11
24	1.26	29.24	3.67E-11	3.91E-11	3.27E-11	3.36E-11	4.02E-11
25[(a)]	1.37	30.60	4.53E-11	4.51E-11	3.50E-11	3.55E-11	4.74E-11

Note:

(a) Cycle 25 is the current operating cycle. Values listed for this cycle are projections based on the Cycle 25 design.

Table 2-7 Calculated Iron Atom Displacements at the Pressure Vessel Clad/Base Metal Interface

Cycle	Cycle Length (EFPY)	Cumulative Operating Time (EFPY)	Iron Atom Displacements (dpa)				
			0°	15°	30°	45°	Maximum
1	0.89	0.89	1.55E-03	1.54E-03	1.16E-03	1.13E-03	1.63E-03
2	0.80	1.69	3.30E-03	3.25E-03	2.45E-03	2.36E-03	3.45E-03
3	0.64	2.33	4.63E-03	4.54E-03	3.43E-03	3.32E-03	4.83E-03
4	0.97	3.31	6.68E-03	6.47E-03	5.00E-03	4.88E-03	6.91E-03
5	0.85	4.16	8.60E-03	8.32E-03	6.37E-03	6.24E-03	8.90E-03
6	1.22	5.38	1.06E-02	1.03E-02	7.86E-03	7.61E-03	1.10E-02
7	1.13	6.51	1.23E-02	1.20E-02	9.17E-03	8.82E-03	1.28E-02
8	1.15	7.66	1.41E-02	1.37E-02	1.05E-02	1.00E-02	1.47E-02
9	1.18	8.84	1.59E-02	1.55E-02	1.19E-02	1.12E-02	1.66E-02
10	1.32	10.16	1.78E-02	1.74E-02	1.34E-02	1.27E-02	1.86E-02
11	1.33	11.49	1.94E-02	1.87E-02	1.45E-02	1.38E-02	2.01E-02
12	1.31	12.81	2.09E-02	2.02E-02	1.56E-02	1.49E-02	2.17E-02
13	1.47	14.27	2.25E-02	2.15E-02	1.67E-02	1.62E-02	2.32E-02
14	1.41	15.69	2.40E-02	2.31E-02	1.79E-02	1.74E-02	2.48E-02
15	1.29	16.98	2.54E-02	2.44E-02	1.91E-02	1.85E-02	2.63E-02
16	1.35	18.33	2.72E-02	2.60E-02	2.03E-02	1.97E-02	2.80E-02
17	1.36	19.69	2.89E-02	2.76E-02	2.16E-02	2.10E-02	2.97E-02
18	1.43	21.12	3.07E-02	2.95E-02	2.31E-02	2.24E-02	3.17E-02
19	1.34	22.46	3.24E-02	3.12E-02	2.44E-02	2.37E-02	3.35E-02
20	1.36	23.82	3.41E-02	3.29E-02	2.57E-02	2.50E-02	3.53E-02
21	1.35	25.17	3.58E-02	3.46E-02	2.70E-02	2.63E-02	3.70E-02
22	1.45	26.61	3.76E-02	3.64E-02	2.84E-02	2.77E-02	3.89E-02
23	1.36	27.98	3.92E-02	3.81E-02	2.98E-02	2.91E-02	4.06E-02
24	1.26	29.24	4.06E-02	3.96E-02	3.11E-02	3.04E-02	4.22E-02
25[(a)]	1.37	30.60	4.25E-02	4.14E-02	3.25E-02	3.19E-02	4.41E-02
Future[(b)]		32.00	4.42E-02	4.32E-02	3.39E-02	3.33E-02	4.59E-02
Future[(b)]		36.00	4.92E-02	4.84E-02	3.81E-02	3.76E-02	5.13E-02
Future[(b)]		40.00	5.42E-02	5.37E-02	4.23E-02	4.19E-02	5.67E-02
Future[(b)]		48.00	6.43E-02	6.42E-02	5.08E-02	5.05E-02	6.76E-02
Future[(b)]		54.00	7.19E-02	7.21E-02	5.71E-02	5.69E-02	7.58E-02

Notes:

(a) Cycle 25 is the current operating cycle. Values listed for this cycle are projections based on the Cycle 25 design.

(b) Values beyond Cycle 25 are based on the average power distributions and core operating conditions of Cycles 23–25.

Table 2-8 Calculational Uncertainties

Description	Capsule and Vessel IR
PCA Comparisons	3%
H.B. Robinson Comparisons	3%
Analytical Sensitivity Studies	11%
Additional Uncertainty for Factors not Explicitly Evaluated	5%
Net Calculational Uncertainty	13%

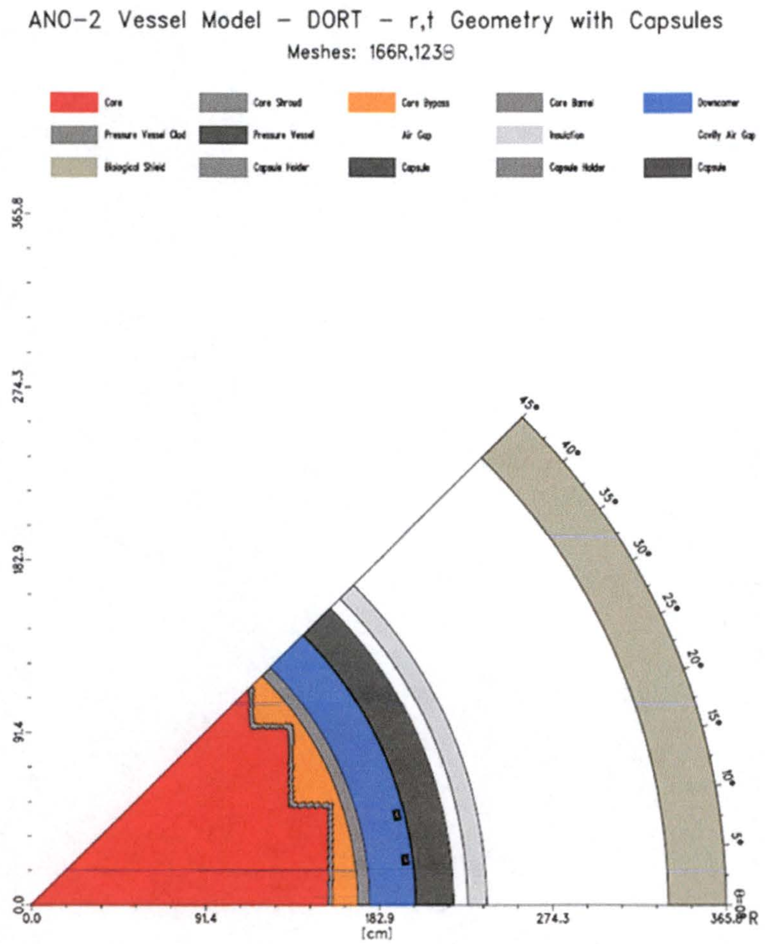

Figure 2-1 Arkansas Nuclear One Unit 2 r,θ Reactor Geometry Plan View at the Core Midplane with Surveillance Capsules

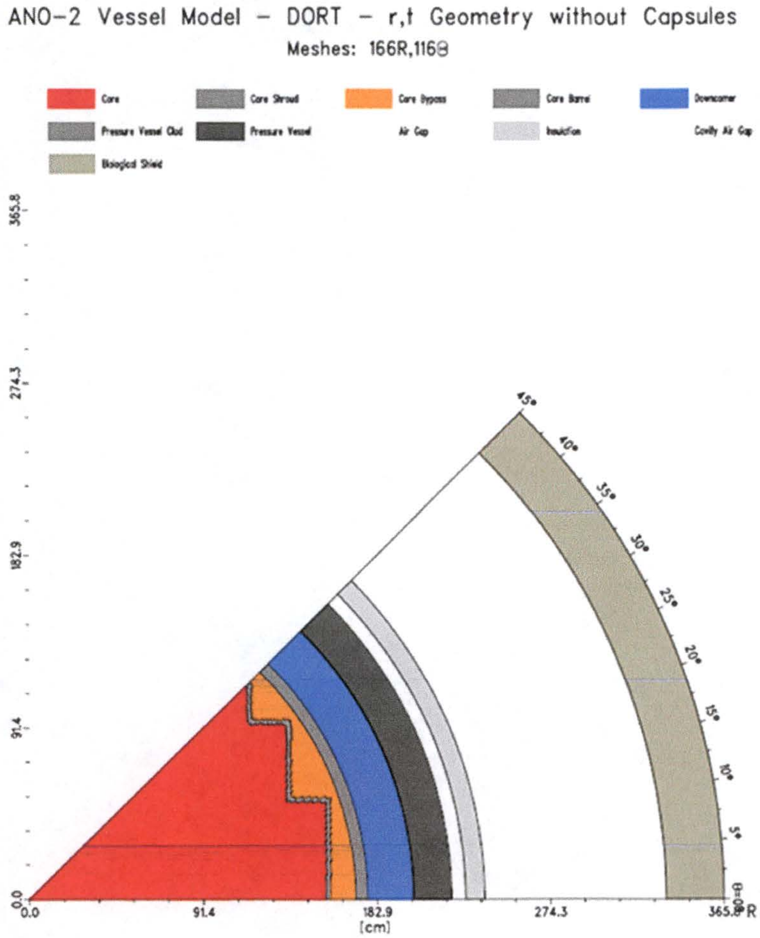

Figure 2-2 Arkansas Nuclear One Unit 2 r,θ Reactor Geometry Plan View at the Core Midplane without Surveillance Capsules

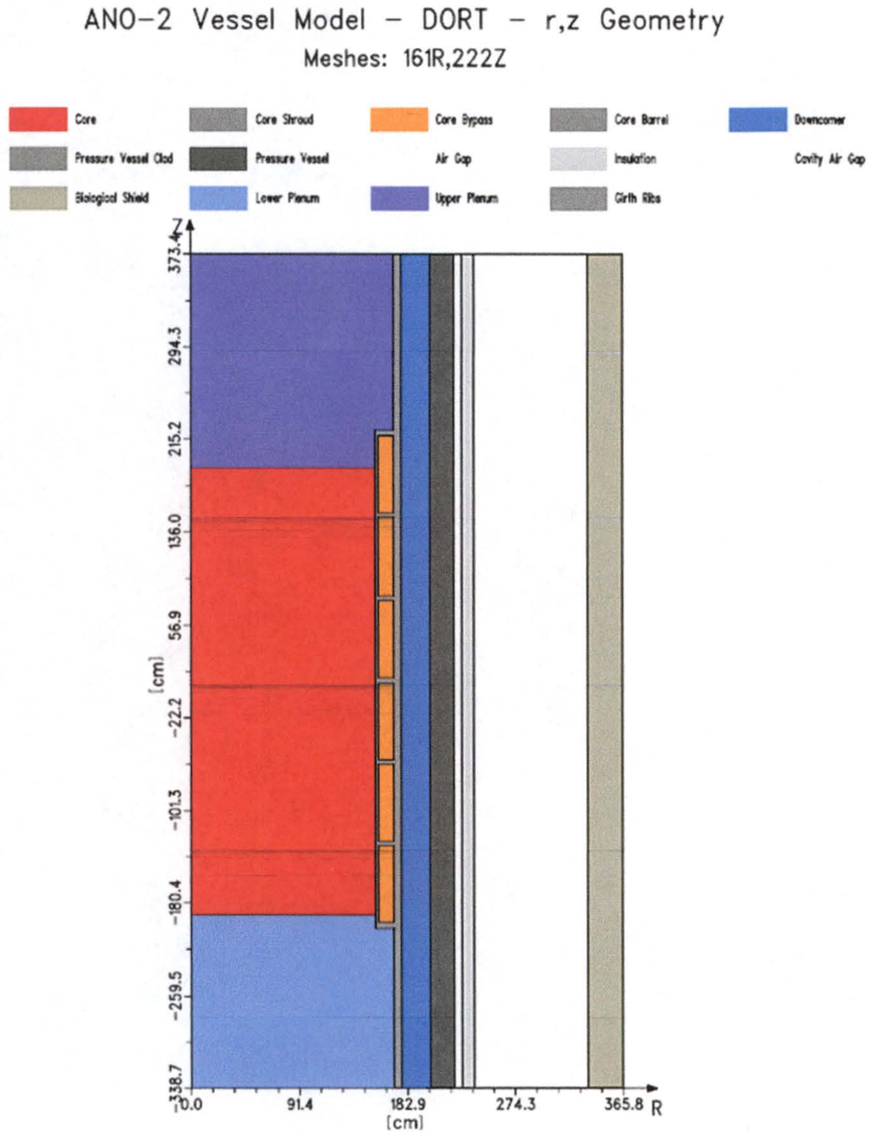

Figure 2-3 Arkansas Nuclear One Unit 2 r,z Reactor Geometry Section View

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 Arkansas Nuclear One Unit 2 beltline materials traditionally included the intermediate and lower shell plate and weld materials; however, as described in NRC Regulatory Issue Summary (RIS) 2014-11 [Ref. 8], any reactor vessel materials that are predicted to experience a neutron fluence exposure greater than 1.0 x 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 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 Table 2-2 of this report, the extended beltline materials include the upper shell plates, upper shell longitudinal welds, and the upper to intermediate shell girth weld. Note that for reactor vessel welds, the terms "girth" and "circumferential" are used interchangeably; herein, these welds shall be referred to as girth welds. Similarly, for reactor vessel welds, the terms "axial" and "longitudinal" are used interchangeably; herein, these welds shall be referred to as longitudinal welds. 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 methodology and evaluations used to determine the initial RT_{NDT} values for the Arkansas Nuclear One Unit 2 reactor vessel beltline and extended beltline base metal materials were reviewed and updated, as appropriate. The initial RT_{NDT} value of Intermediate Shell Plate C-8009-3 was determined per ASME Code, Section III, Subsection NB-2300 [Ref. 9]. The initial RT_{NDT} values for each of the other eight reactor vessel plates were determined per BTP 5-3, Paragraph B1.1(3) [Ref. 10] in conjunction with ASME Code, Section III, Subsection NB-2300 [Ref. 9]. 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. 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} values for the reactor vessel beltline and extended beltline materials are provided in Table 3-1 for Arkansas Nuclear One Unit 2. Table 3-2 contains a summary of the initial RT_{NDT} values of the reactor vessel flange, reactor vessel closure head, replacement reactor vessel closure head, and balance of the reactor coolant system (RCS). These values serve as input to the P-T limit curves "flange-notch" and LST per Appendix G of 10 CFR 50 and ASME Code, Section III, respectively – See Sections 6.3 and 6.4 for details.

Table 3-1 Summary of the Best-Estimate Chemistry and Initial RT_{NDT} Values for the Arkansas Nuclear One Unit 2 Reactor Vessel Materials

Reactor Vessel Material and Identification Number[(a)]	Heat Number[(a)]	Chemical Composition[(b)]				Fracture Toughness Property
		Wt. % Cu	Wt. % Ni	Wt. % Mn	Wt. % P	Initial RT_{NDT}[(c)] (°F)
Reactor Vessel Beltline Materials						
Intermediate Shell Plate C-8009-1	C8161-3	0.098	0.605	1.35	0.010	-1.4
Intermediate Shell Plate C-8009-2	C8161-1	0.085	0.600	1.37	0.010	0.5
Intermediate Shell Plate C-8009-3	C8182-2	0.096	0.580	1.36	0.012	0.0[(g)]
Lower Shell Plate C-8010-1	C8161-2	0.085	0.585	1.33	0.009	12.0
Lower Shell Plate C-8010-2	B2545-1	0.083	0.668	1.36	0.008	-16.7
Lower Shell Plate C-8010-3	B2545-2	0.080	0.653	1.36	0.008	-22.6
Intermediate Shell Longitudinal Welds 2-203A, B, & C	Multiple[(d)]	0.05[(d)]	1.00[(d)]	1.16[(d)]	0.014[(d)]	-56
Lower Shell Longitudinal Welds 3-203A, B, & C	10120	0.046[(e)]	0.082[(e)]	1.21	0.012	-56
Intermediate to Lower Shell Girth Weld 9-203	83650	0.045[(e)]	0.087[(e)]	1.24	0.006	-40[(g)]
Reactor Vessel Extended Beltline Materials						
Upper Shell Plate[(j)] C-8008-1	C8182-1	0.13	0.60	1.36	0.011	12.2
Upper Shell Plate C-8008-2	C7605-1	0.13	0.55	1.36	0.013	60.5
Upper Shell Plate C-8008-3	C8571-2	0.08	0.55	1.29	0.014	27.3
Upper Shell Longitudinal Welds 1-203A, B, & C	BOLA	0.02	0.93	1.02	0.010	-60[(g)]
Upper to Intermediate Shell Girth Weld 8-203	10137	0.22	0.02	0.94	0.015	-56
	6329637	0.21	0.11[(f)]	1.22	0.011	-56
	FAGA	0.03	0.95	1.00	0.008	-24[(h)]
Surveillance Weld Data[(i)]						
Arkansas Nuclear One Unit 2	83650	0.045	0.083	1.33	0.007	- - -
Calvert Cliffs Unit 2	10137	0.21	0.06	- - -	- - -	- - -
Millstone Unit 2		0.21	0.06	- - -	- - -	- - -
J.M. Farley Unit 2	BOLA	0.028	0.89	- - -	- - -	- - -

Notes on following page.

Notes:

(a) The reactor vessel plate and weld material identification and heat numbers were taken from the Arkansas Nuclear One Unit 2 Certified Material Test Reports (CMTRs) and/or A-PENG-ER-002 [Ref. 11], unless otherwise noted.

(b) All chemistry values obtained from A-PENG-ER-002 and/or the Arkansas Nuclear One Unit 2 CMTRs, unless otherwise noted. Chemistry values for plates are the average of all available analyses. Chemistry values for welds are the average of all coated electrode deposit chemistry (CEDC) for the E-8018 stick electrodes or weld flux deposit chemistry (WFDC) for Linde 0091 welds, unless otherwise noted. Where available, additional chemistry analysis results from BMI-0584 [Ref. 12] were also included in the average. The chemistry values for the beltline plates reported in this table are identical to those previously reported in the Arkansas Nuclear One Unit 2 License Renewal Application (LRA) [Ref. 13] and the previous capsule report, BAW-2399, Revision 1 [Ref. 14].

(c) The $RT_{NDT(U)}$ values for the plates are based on drop-weight data, longitudinally-oriented Charpy V-notch test data and NUREG-0800, BTP 5-3 Position 1.1(3)(a) and (b) [Ref. 10], with the more limiting $RT_{NDT(U)}$ value being selected, unless otherwise noted. The $RT_{NDT(U)}$ values for welds are the generic value for Linde 0091 flux type welds (-56°F) per 10 CFR 50.61 [Ref. 15], unless otherwise noted.

(d) The material Heat numbers for the Intermediate Shell Longitudinal Welds 2-203A, B, & C are unclear in the historical data. For conservatism, the material properties for the Intermediate Shell Longitudinal Welds 2-203A, B, & C reported are the most limiting values from welds relevant to the Intermediate Shell Longitudinal Welds 2-203A, B, & C per A-PENG-ER-002. These welds include Heat # 10120, Flux Type Linde 0091, Lot # 3999 (sister plant weld), Heat # 10120,10120, Heat # AAGC, and the analysis of the in-process weld deposit chemistry.

(e) The Cu and Ni wt. % values for the Lower Shell Longitudinal Welds 3-203A, B, & C and the Intermediate to Lower Shell Girth Weld 9-203 are consistent with BAW-2399, Revision 1 [Ref. 14] and were originally taken from CE NPSD-1039, Revision 2 [Ref. 16].

(f) Weld Heat # 6329637 does not contain any WFDC Ni wt. % values, thus the bare wire chemical analysis (BWCA) value of 0.11% from A-PENG-ER-002 [Ref. 11] was used.

(g) This $RT_{NDT(U)}$ value for the surveillance plate, Intermediate Shell Plate C-8009-3, is based on drop-weight data, transverse orientation Charpy V-notch test data taken from the baseline capsule test report, TR-MCD-002 [Ref. 17] and ASME Code Section III Subarticle NB-2331 [Ref. 9]. The $RT_{NDT(U)}$ value for the two weld materials, Heat numbers 83650 and BOLA, is based on drop-weight data, Charpy V-notch test data taken from A-PENG-ER-002 and ASME Code Section III Subarticle NB-2331 [Ref. 9].

(h) Drop-weight test data is not available for this E-8018 weld heat. Therefore, to assign an $RT_{NDT(U)}$ value to this E-8018 stick electrode weld, the data in Table 8 of A-PENG-ER-002 was analyzed. The average T_{NDT} value for the 17 E-8018 weld heats is -57°F with a standard deviation of 16.5°F. This yields a bounding value of -24°F using a mean plus two sigma model; therefore, a value of -24°F is acceptable for the initial RT_{NDT} value of this weld material, with consideration that its Charpy impact energy at 10°F, which is less than T_{NDT} + 60°F, was greater than 100 ft-lb. Furthermore, -24°F bounds all of the E-8018 stick electrode T_{NDT} values present in A-PENG-ER-002 [Ref. 11].

(i) Surveillance data exists for weld Heat # 83650, # 10137, and # BOLA from multiple sources; see Section 4 for more details. The data for Arkansas Nuclear One Unit 2 weld metal Heat # 83650 was taken as the average of the data available from TR-MCD-002, as well as the subsequent analyses completed during testing of the first capsule, BMI-0584 [Ref. 12]. The data for the Calvert Cliffs Unit 2 weld metal Heat # 10137 was taken from Table 4-3 of WCAP-17501-NP [Ref. 18]. The data for the Millstone Unit 2 weld metal Heat # 10137 was taken from Table 4-1 of WCAP-16012 [Ref. 19]. The data for the J.M. Farley Unit 2 weld metal Heat # BOLA was taken from Table 4-1 of WCAP-16918, Revision 1 [Ref. 20].

(j) Upper Shell Plate C-8008-1 shares the same material heat number as the Arkansas Nuclear One Unit 2 surveillance plate material, Intermediate Shell Plate C-8009-3; therefore, surveillance program test results apply to this material as well.

Table 3-2 Summary of Arkansas Nuclear One Unit 2 Reactor Vessel Closure Heads, Vessel Flange and Balance of RCS Initial RT_{NDT} Values

Reactor Vessel Material	Initial RT_{NDT} (°F)	Methodology
Current Closure Head (Heat # 125B404)	10	BTP 5-3, Paragraph B1.1(3)(a) and (b) [Ref. 10]
Replacement Closure Head[(a)] (Heat # R3781/R3782)	-22	ASME Code, Section III, Subsection NB-2300 [Ref. 9]
Vessel Flange (Heat # 122A440)	30	BTP 5-3, Paragraph B1.1(3)(a) and (b) [Ref. 10]
Balance of RCS[(b)]	50	Note (b)

Notes:

(a) A replacement, single forging, closure head has been fabricated; however, it has not yet been installed at Arkansas Nuclear One Unit 2. The vessel flange initial RT_{NDT} value is higher than both the current and replacement closure head initial RT_{NDT} values. Thus, the results contained herein are conservative for the current and replacement closure heads.

(b) 50°F was conservatively assigned to all RCS material not specifically tested per Section 5.2.4.3 of the Arkansas Nuclear One Unit 2 updated Final Safety Analysis Report (FSAR).

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, data 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. Data from a surveillance program at another plant is often called 'sister plant' data.

The surveillance capsule plate material for Arkansas Nuclear One Unit 2 is from Intermediate Shell Plate C-8009-3. Surveillance results from this plate also apply to Upper Shell Plate C-8008-1, because the two plates were made from the same heat of material (Heat # C8182). The surveillance capsule weld material for Arkansas Nuclear One Unit 2 is Heat # 83650, which is applicable to the intermediate to lower shell girth weld. Table 4-1 summarizes the Arkansas Nuclear One Unit 2 surveillance data for the plate material and weld material (Heat # 83650) that will be used in the calculation of the Position 2.1 chemistry factor values for these materials. The results of the last withdrawn and tested surveillance capsule, Capsule 284°, were documented in WCAP-18166-NP [Ref. 21]. Appendix D of WCAP-18166-NP concluded that the surveillance plate and weld (Heat # 83650) data are credible; therefore, a reduced margin term will be utilized in the ART calculations contained in Section 7.

The Arkansas Nuclear One Unit 2 reactor vessel upper to intermediate shell girth weld seam was fabricated using weld Heat # 10137. Weld Heat # 10137 is contained in the Calvert Cliffs Unit 2 and Millstone Unit 2 surveillance programs. Thus, the Calvert Cliffs Unit 2 and Millstone Unit 2 data will be used in the calculation of the Position 2.1 chemistry factor value for Arkansas Nuclear One Unit 2 weld Heat # 10137. Note that no surveillance data is available for the other two Heats (# 6329637 and # FAGA) which were also used to make the Upper to Intermediate Shell Girth Weld 8-203. Table 4-2 summarizes the applicable surveillance capsule data pertaining to weld Heat # 10137. The combined surveillance data is deemed credible per Appendix D; however, as a result of the Millstone Unit 2 surveillance data including both weld Heat # 10137 and 90136, the Position 2.1 chemistry factor calculations for weld Heat # 10137 will utilize a full margin term for conservatism. See Appendix D for details.

The Arkansas Nuclear One Unit 2 reactor vessel upper shell longitudinal weld seams were fabricated using weld Heat # BOLA. Weld Heat # BOLA is contained in the J.M. Farley Unit 2 surveillance program. Thus, the J.M. Farley Unit 2 data will be used in the calculation of the Position 2.1 chemistry factor value for Arkansas Nuclear One Unit 2 weld Heat # BOLA. Table 4-3 summarizes the applicable surveillance capsule data pertaining to weld Heat # BOLA. Per Appendix D of WCAP-16918-NP, Revision 1 [Ref. 20], the J.M. Farley Unit 2 surveillance weld data is deemed non-credible. Since the J.M. Farley Unit 2 surveillance weld is not analyzed with any additional surveillance capsule material herein, this credibility conclusion is applicable to the Arkansas Nuclear One Unit 2 weld Heat # BOLA. Therefore, a full margin term will be utilized in the ART calculations contained in Section 7.

Table 4-1 Arkansas Nuclear One Unit 2 Surveillance Capsule Data

Material	Capsule[a]	Capsule Fluence[a] (x 10^{19} n/cm^2, E > 1.0 MeV)	Measured 30 ft-lb Transition Temperature Shift[a] (°F)
Intermediate Shell Plate C-8009-3 (Longitudinal)	97°	0.303	23.5
	284°	3.67	85.7
Intermediate Shell Plate C-8009-3 (Transverse)	97°	0.303	33.4
	104°	2.15	52.9
	284°	3.67	85.6
Surveillance Weld Material (Heat # 83650)	97°	0.303	13.2
	104°	2.15	16.1
	284°	3.67	12.0

Note:

(a) Surveillance data was taken from Table 5-10 of WCAP-18166-NP [Ref. 21].

Table 4-2 Calvert Cliffs Unit 2 and Millstone Unit 2 Surveillance Capsule Data for Weld Heat # 10137

Material	Capsule[(a)]	Capsule Fluence[(a)] (x 10^{19} n/cm^2, E > 1.0 MeV)	Measured 30 ft-lb Transition Temperature Shift[(a)] (°F)	Inlet Temperature[(b)] (°F)	Temperature Adjustment[(c)] (°F)
Calvert Cliffs Unit 2 Data	263°	0.825	72.7	550	-1.0
	97°	1.95	82.9	549	-2.0
	104°	2.44	69.7	548	-3.0
Millstone Unit 2 Data	97°	0.324	65.93	544.3	-6.7
	104°	0.949	52.12	547.6	-3.4
	83°	1.74	56.09	548.0	-3.0

Notes:

(a) For surveillance weld Heat # 10137, data pertaining to Calvert Cliffs Unit 2 were taken from Table 5-10 of WCAP-17501-NP [Ref. 18]. Data pertaining to Millstone Unit 2 were taken from Table 5-10 of WCAP-16012 [Ref. 19].

(b) Inlet temperatures were calculated as the average inlet temperature from all the previously completed cycles at the time of capsule withdrawal.

(c) Temperature adjustment = 1.0*($T_{capsule}$ - T_{plant}), where T_{plant} = 551.0°F for Arkansas Nuclear One Unit 2 (applied to the weld ΔRT_{NDT} data for each of the Calvert Cliffs Unit 2 and Millstone Unit 2 capsules in the Position 2.1 chemistry factor calculation – See Section 5 for more details).

Table 4-3 J.M. Farley Unit 2 Surveillance Capsule Data for Weld Heat # BOLA

Material	Capsule[(a)]	Capsule Fluence[(a)] (x 10^{19} n/cm^2, E > 1.0 MeV)	Measured 30 ft-lb Transition Temperature Shift[(a)] (°F)	Inlet Temperature[(b)] (°F)	Temperature Adjustment[(c)] (°F)
J.M. Farley Unit 2	U	0.605	-28.4	544	-7.0
	W	1.73	7.0	542	-9.0
	X	2.98	-15.6	543	-8.0
	Z	4.92	10.2	543	-8.0
	Y	6.79	69.1	543	-8.0
	V	8.73	56.5	542	-9.0

Notes:

(a) For surveillance weld Heat # BOLA, data pertaining to J.M. Farley Unit 2 were taken from Table 5-10 of WCAP-16918-NP, Revision 1 [Ref. 20].

(b) Inlet temperatures were calculated as the average inlet temperature from all the previously completed cycles at the time of capsule withdrawal.

(c) Temperature adjustment = 1.0*($T_{capsule}$ - T_{plant}), where T_{plant} = 551.0°F for Arkansas Nuclear One Unit 2 (applied to the weld ΔRT_{NDT} data for each of the J.M. Farley 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 Arkansas Nuclear One Unit 2 reactor vessel materials are provided in Table 3-1 of this report.

The Position 2.1 chemistry factors are calculated for the materials that have available surveillance program results. The calculation is performed using the method described in Regulatory Guide 1.99, Revision 2. The Arkansas Nuclear One Unit 2 surveillance data as well as the applicable sister plant data was summarized in Section 4 of this report, and will be utilized in the Position 2.1 chemistry factor calculations in this Section.

The Position 2.1 chemistry factor calculations are presented in Tables 5-1 through 5-4 for Arkansas Nuclear One Unit 2 reactor vessel materials that have associated surveillance data. These values were calculated using the surveillance data summarized in Section 4 of this report. 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. 22]. Margin will be applied to the ART calculations in Section 7 according to the conclusions of the credibility evaluation for each of the surveillance materials, as documented in Section 4.

The Position 1.1 chemistry factors are summarized along with the Position 2.1 chemistry factors in Table 5-5 for Arkansas Nuclear One Unit 2.

Table 5-1 Calculation of Arkansas Nuclear One Unit 2 Chemistry Factor Value for Intermediate Shell Plate C-8009-3 Using Surveillance Capsule Data

Intermediate Shell (IS) Plate C-8009-3 Data[a]	Capsule	Capsule f[b] (x 10^{19} n/cm^2, E > 1.0 MeV)	FF[c]	ΔRT_{NDT}[d] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Longitudinal Orientation	97°	0.303	0.6728	23.5	15.81	0.453
	284°	3.67	1.3373	85.7	114.60	1.788
Transverse Orientation	97°	0.303	0.6728	33.4	22.47	0.453
	104°	2.15	1.2079	52.9	63.90	1.459
	284°	3.67	1.3373	85.6	114.47	1.788
				SUM:	331.26	5.941
	$CF_{IS\ Plate\ C\text{-}8009\text{-}3} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (331.26) \div (5.941) = \mathbf{55.8°F}$					

Notes:

(a) This surveillance data applies to both Intermediate Shell Plate C-8009-3 and Upper Shell Plate C-8008-1, since the two plates were made from the same heat of material (Heat # C8182).

(b) f = fluence.

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

(d) ΔRT_{NDT} values are the measured 30 ft-lb shift values. All values are taken from Table 4-1 of this report.

Table 5-2 Calculation of Arkansas Nuclear One Unit 2 Chemistry Factor Value for Weld Heat # 83650 Using Surveillance Capsule Data

Weld Metal Heat # 83650	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
Arkansas Nuclear One Unit 2 Data	97°	0.303	0.6728	13.3 (13.2)	8.97	0.453
	104°	2.15	1.2079	16.3 (16.1)	19.64	1.459
	284°	3.67	1.3373	12.1 (12.0)	16.21	1.788
				SUM:	44.82	3.700
	$CF_{Weld\ Heat\ \#\ 83650} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (44.82) \div (3.70) = \mathbf{12.1°F}$					

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 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). Ratio applied to the Arkansas Nuclear One Unit 2 surveillance data = $CF_{Vessel\ Weld} / CF_{Surv.\ Weld}$ = 34.1°F / 33.7°F = 1.01.

Table 5-3 Calculation of Arkansas Nuclear One Unit 2 Chemistry Factor Value for Weld Heat # 10137 Using Surveillance Capsule Data

Weld Metal Heat # 10137	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
Calvert Cliffs Unit 2 Data	263°	0.825	0.9460	73.1 (72.7)	69.19	0.895
	97°	1.95	1.1825	82.5 (82.9)	97.58	1.398
	104°	2.44	1.2401	68.0 (69.7)	84.37	1.538
Millstone Unit 2 Data[d]	97°	0.324	0.6902	60.4 (65.93)	41.70	0.476
	104°	0.949	0.9853	49.7 (52.12)	48.97	0.971
	83°	1.740	1.1523	54.2 (56.09)	62.40	1.328
				SUM:	404.20	6.606
	$CF_{Weld\ Heat\ \#\ 10137} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (404.20) \div (6.606) = \mathbf{61.2°F}$					

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 4-2 of this report). The temperature adjustments are listed in Table 4-2. Ratio applied to the Calvert Cliffs Unit 2 surveillance data = $CF_{Vessel\ Weld}$ / $CF_{Surv.\ Weld}$ = 98.5°F / 96.8°F = 1.02. Ratio applied to the Millstone Unit 2 surveillance data = $CF_{Vessel\ Weld}$ / $CF_{Surv.\ Weld}$ = 98.5°F / 96.8°F = 1.02.

(d) Millstone Unit 2 surveillance data contains specimens from both weld Heat # 10137 and weld Heat # 90136. However, this inclusion of an additional heat is not expected to negatively impact the subsequent reactor vessel integrity calculation results, as additional conservatisms are in place. See Appendix D for more details.

Table 5-4 Calculation of Arkansas Nuclear One Unit 2 Chemistry Factor Value for Weld Heat # BOLA Using Surveillance Capsule Data

Weld Metal Heat # BOLA	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
J.M. Farley Unit 2	U	0.605	0.8593	0.0[d] (-28.4)	0.00	0.738
	W	1.73	1.1508	0.0[d] (7.0)	0.00	1.324
	X	2.98	1.2891	0.0[d] (-15.6)	0.00	1.662
	Z	4.92	1.3992	2.2 (10.2)	3.08	1.958
	Y	6.79	1.4579	61.1 (69.1)	89.08	2.125
	V	8.73	1.4960	47.5 (56.5)	71.06	2.238
				SUM:	163.22	10.046
	$CF_{Heat\ \#\ BOLA} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (163.22) \div (10.046) =$ **16.2°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 4-3 of this report). The temperature adjustments are listed in Table 4-3. A ratio of 1.00 was conservatively applied to the J.M. Farley Unit 2 surveillance data, since $CF_{Vessel\ Weld} < CF_{Surv.\ Weld}$.
(d) A negative ΔRT_{NDT} value was calculated after temperature adjustment. Physically, this should not occur; thus a conservative value of 0.0°F was used.

Table 5-5 Summary of Arkansas Nuclear One 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 Plate C-8009-1	C8161-3	63.6	---
Intermediate Shell Plate C-8009-2	C8161-1	54.5	---
Intermediate Shell Plate C-8009-3	C8182-2	62.2	55.8[(b)]
Lower Shell Plate C-8010-1	C8161-2	54.5	---
Lower Shell Plate C-8010-2	B2545-1	53.1	---
Lower Shell Plate C-8010-3	B2545-2	51.0	---
Intermediate Shell Longitudinal Welds 2-203A, B, & C	Multiple	68.0	---
Lower Shell Longitudinal Welds 3-203A, B, & C	10120	34.0	---
Intermediate to Lower Shell Girth Weld 9-203	83650	34.1	12.1[(c)]
Reactor Vessel Extended Beltline Materials			
Upper Shell Plate C-8008-1	C8182-1	91.0	55.8[(b)]
Upper Shell Plate C-8008-2	C7605-1	89.5	---
Upper Shell Plate C-8008-3	C8571-2	51.0	---
Upper Shell Longitudinal Welds 1-203A, B, & C	BOLA	27.0	16.2[(d)]
Upper to Intermediate Shell Girth Weld 8-203	10137	98.5	61.2[(e)]
	6329637	100.8	---
	FAGA	41.0	---
Surveillance Weld Data			
Arkansas Nuclear One Unit 2	83650	33.7	---
Calvert Cliffs Unit 2	10137	96.8	---
Millstone Unit 2		96.8	---
J.M. Farley Unit 2	BOLA	38.2	---

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.

(b) Position 2.1 chemistry factor was taken from Table 5-1 of this report. As discussed in Section 4, the surveillance plate data is credible and applicable to both Intermediate Shell Plate C-8009-3 and Upper Shell Plate C-8008-1.

(c) Position 2.1 chemistry factor was taken from Table 5-2 of this report. As discussed in Section 4, the surveillance weld data for Heat # 83650 is credible.

(d) Position 2.1 chemistry factor was taken from Table 5-4 of this report. As discussed in Section 4, the surveillance weld data for Heat # BOLA is not credible.

(e) Position 2.1 chemistry factor was taken from Table 5-3 of this report. As discussed in Section 4, the surveillance weld data for Heat # 10137 is credible; however no reduction in the margin term will be taken.

6 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

6.1 OVERALL APPROACH

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{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:

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

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 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:

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

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

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

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

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

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

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:

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

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 622 psig. The initial RT_{NDT} values of the reactor vessel closure head, replacement reactor vessel closure head, and vessel flange are documented in Table 3-2. The limiting unirradiated RT_{NDT} of 30°F is associated with the vessel flange of the Arkansas Nuclear One Unit 2 vessel, so the minimum allowable temperature of this region is 150°F at pressures greater than 622 psig (without margins for instrument uncertainties). This limit is shown in Figures 8-1 and 8-2.

6.4 LOWEST SERVICE TEMPERATURE REQUIREMENTS

The lowest service temperature (LST) is the minimum allowable temperature at which pressure can exceed 20% of the pre-service hydrostatic test pressure (3110 psig). This temperature is defined by Paragraphs NB-3211 and NB-2332 of ASME Code Section III [Ref. 9] as the most limiting RT_{NDT} for the balance of the RCS components plus 100°F. The balance of the reactor coolant system components includes consideration of the ferritic materials outside the reactor vessel cylindrical shell beltline, nozzle corner (see Appendix B), closure head, and vessel flange regions, but within the primary system. Per Table 3-2, the most limiting RT_{NDT} for the balance of RCS is 50°F. Therefore, without margins for instrument errors, the LST for Arkansas Nuclear One Unit 2 is 150°F. For Arkansas Nuclear One Unit 2, this limit is identical to the vessel flange limit described in Section 6.3 and is shown in Figures 8-1 and 8-2.

6.5 BOLTUP TEMPERATURE REQUIREMENTS

The minimum boltup temperature is the minimum allowable temperature at which the reactor vessel closure head bolts can be preloaded. It is determined by the highest reference temperature, RT_{NDT}, in the closure flange region. This requirement is established in Appendix G to 10 CFR 50 [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 Arkansas Nuclear One 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, 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. 9]. 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 7.875 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 54 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 [Ref. 1]. The values in this table will be used to calculate the 54 EFPY ART values for the Arkansas Nuclear One 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 54 EFPY ART calculations at the 1/4T and 3/4T locations for generation of the Arkansas Nuclear One Unit 2 heatup and cooldown curves.

The inlet and outlet nozzle forging materials for Arkansas Nuclear One Unit 2 have projected fluence values that do not exceed the 1 x 10^{17} n/cm^2 fluence threshold at 54 EFPY per Table 2-2 at the lowest extent of the nozzle; therefore, per NRC RIS 2014-11 [Ref. 8], neutron radiation embrittlement need not be considered herein for these materials. Thus, ART calculations for the inlet and outlet nozzle forging 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 materials are contained in Appendix B.

The limiting ART values for Arkansas Nuclear One Unit 2 to be used in the generation of the P-T limit curves are based on Lower Shell Plate C-8010-1 (Position 1.1). In order to provide an additional margin of conservatism, the limiting calculated ART values were rounded up and increased by 5°F. The increased limiting ART values, using the "Axial Flaw" methodology, for Lower Shell Plate C-8010-1 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 Arkansas Nuclear One Unit 2 Reactor Vessel Materials at 54 EFPY

Reactor Vessel Region	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 Plates	4.91×10^{19}	3.06×10^{19}	1.2955	1.19×10^{19}	1.0485
Lower Shell Plates	4.98×10^{19}	3.10×10^{19}	1.2988	1.21×10^{19}	1.0524
Intermediate Shell Longitudinal Welds	4.64×10^{19}	2.89×10^{19}	1.2820	1.12×10^{19}	1.0327
Lower Shell Longitudinal Welds	4.71×10^{19}	2.94×10^{19}	1.2856	1.14×10^{19}	1.0369
Intermediate to Lower Shell Girth Weld	4.89×10^{19}	3.05×10^{19}	1.2945	1.18×10^{19}	1.0474
Reactor Vessel Extended Beltline Materials					
Upper Shell Plates	5.89×10^{17}	3.67×10^{17}	0.2467	1.43×10^{17}	0.1389
Upper Shell Longitudinal Welds	5.89×10^{17}	3.67×10^{17}	0.2467	1.43×10^{17}	0.1389
Upper to Intermediate Shell Girth Weld	5.89×10^{17}	3.67×10^{17}	0.2467	1.43×10^{17}	0.1389

Note:

(a) 54 EFPY fluence values are documented in Table 2-2.

Table 7-2 Adjusted Reference Temperature Evaluation for the Arkansas Nuclear One Unit 2 Reactor Vessel Beltline Materials through 54 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} (°F)	σ_I[(a)] (°F)	σ_Δ[(b)] (°F)	Margin (°F)	ART[(c)] (°F)
Reactor Vessel Beltline Materials										
Intermediate Shell Plate C-8009-1	C8161-3	63.6	3.06×10^{19}	1.2955	-1.4	82.4	0.0	17.0	34.0	115.0
Intermediate Shell Plate C-8009-2	C8161-1	54.5	3.06×10^{19}	1.2955	0.5	70.6	0.0	17.0	34.0	105.1
Intermediate Shell Plate C-8009-3	C8182-2	62.2	3.06×10^{19}	1.2955	0.0	80.6	0.0	17.0	34.0	114.6
Using credible Arkansas Nuclear One Unit 2 surveillance data	C8182-2	55.8	3.06×10^{19}	1.2955	0.0	72.3	0.0	8.5	17.0	89.3
Lower Shell Plate C-8010-1	C8161-2	54.5	3.10×10^{19}	1.2988	12.0	70.8	0.0	17.0	34.0	116.8
Lower Shell Plate C-8010-2	B2545-1	53.1	3.10×10^{19}	1.2988	-16.7	69.0	0.0	17.0	34.0	86.3
Lower Shell Plate C-8010-3	B2545-2	51.0	3.10×10^{19}	1.2988	-22.6	66.2	0.0	17.0	34.0	77.6
Intermediate Shell Longitudinal Welds 2-203A, B, & C	Multiple	68.0	2.89×10^{19}	1.2820	-56	87.2	17.0	28.0	65.5	96.7
Lower Shell Longitudinal Welds 3-203A, B, & C	10120	34.0	2.94×10^{19}	1.2856	-56	43.7	17.0	21.9	55.4	43.1
Intermediate to Lower Shell Girth Weld 9-203	83650	34.1	3.05×10^{19}	1.2945	-40	44.1	0.0	22.1	44.1	48.3
Using credible Arkansas Nuclear One Unit 2 surveillance data	83650	12.1	3.05×10^{19}	1.2945	-40	15.7	0.0	7.8	15.7	-8.7
Reactor Vessel Extended Beltline Materials										
Upper Shell Plate C-8008-1	C8182-1	91.0	0.0367×10^{19}	0.2467	12.2	22.5	0.0	11.2	22.5	57.1
Using credible Arkansas Nuclear One Unit 2 surveillance data	C8182-1	55.8	0.0367×10^{19}	0.2467	12.2	13.8	0.0	6.9	13.8	39.7
Upper Shell Plate C-8008-2	C7605-1	89.5	0.0367×10^{19}	0.2467	60.5	22.1	0.0	11.0	22.1	104.7
Upper Shell Plate C-8008-3	C8571-2	51.0	0.0367×10^{19}	0.2467	27.3	12.6	0.0	6.3	12.6	52.5
Upper Shell Longitudinal Welds 1-203A, B, & C	BOLA	27.0	0.0367×10^{19}	0.2467	-60	6.7	0.0	3.3	6.7	-46.7
Using non-credible J.M. Farley Unit 2 surveillance data	BOLA	16.2	0.0367×10^{19}	0.2467	-60	4.0	0.0	2.0	4.0	-52.0

Table 7-2 Adjusted Reference Temperature Evaluation for the Arkansas Nuclear One Unit 2 Reactor Vessel Beltline Materials through 54 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} (°F)	σ_I[a] (°F)	σ_Δ[b] (°F)	Margin (°F)	ART[c] (°F)
Upper to Intermediate Shell Girth Weld 8-203	10137	98.5	0.0367 x 10^{19}	0.2467	-56	24.3	17.0	12.2	41.8	10.1
	Using credible Calvert Cliffs Unit 2 and Millstone Unit 2 data	61.2	0.0367 x 10^{19}	0.2467	-56	15.1	17.0	28.0	65.5	24.6
	6329637	100.8	0.0367 x 10^{19}	0.2467	-56	24.9	17.0	12.4	42.1	11.0
	FAGA	41.0	0.0367 x 10^{19}	0.2467	-24	10.1	0.0	5.1	10.1	-3.8

Notes:

(a) The plate material initial RT_{NDT} values are measured values. For weld materials with generic initial RT_{NDT} values, σ_I = 17°F. For weld materials with measured initial RT_{NDT} values, σ_I = 0°F.

(b) As discussed in Section 4, the surveillance plate and weld Heat # 83650 data were deemed credible, while the weld Heat # BOLA data were deemed non-credible. The surveillance weld data for Heat # 10137 was deemed credible; however, per Section 4 and Appendix D, a full margin term will be used. Per the guidance of Regulatory Guide 1.99, Revision 2 [Ref. 1], the base metal σ_Δ = 17°F for Position 1.1 and σ_Δ = 8.5°F for Position 2.1 with credible surveillance data. The weld metal σ_Δ = 28°F for Position 1.1 and 2.1 with non-credible surveillance data (Heat # BOLA), and the weld metal σ_Δ = 14°F for Position 2.1 with credible surveillance data (Heat # 83650). Since a full margin term will be used for Heat # 10137, σ_Δ = 28°F with credible surveillance data for Position 2.1 for this weld heat. However, σ_Δ need not exceed 0.5*ΔRT_{NDT}.

(c) The Regulatory Guide 1.99, Revision 2 methodology was used to calculate ART values. ART = $RT_{NDT(U)}$ + ΔRT_{NDT} + Margin.

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Table 7-3 Adjusted Reference Temperature Evaluation for the Arkansas Nuclear One Unit 2 Reactor Vessel Beltline Materials through 54 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} (°F)	σ_I[(a)] (°F)	σ_Δ[(b)] (°F)	Margin (°F)	ART[(c)] (°F)
Reactor Vessel Beltline Materials										
Intermediate Shell Plate C-8009-1	C8161-3	63.6	1.19×10^{19}	1.0485	-1.4	66.7	0.0	17.0	34.0	99.3
Intermediate Shell Plate C-8009-2	C8161-1	54.5	1.19×10^{19}	1.0485	0.5	57.1	0.0	17.0	34.0	91.6
Intermediate Shell Plate C-8009-3	C8182-2	62.2	1.19×10^{19}	1.0485	0.0	65.2	0.0	17.0	34.0	99.2
Using credible Arkansas Nuclear One Unit 2 surveillance data	C8182-2	55.8	1.19×10^{19}	1.0485	0.0	58.5	0.0	8.5	17.0	75.5
Lower Shell Plate C-8010-1	C8161-2	54.5	1.21×10^{19}	1.0524	12.0	57.4	0.0	17.0	34.0	103.4
Lower Shell Plate C-8010-2	B2545-1	53.1	1.21×10^{19}	1.0524	-16.7	55.9	0.0	17.0	34.0	73.2
Lower Shell Plate C-8010-3	B2545-2	51.0	1.21×10^{19}	1.0524	-22.6	53.7	0.0	17.0	34.0	65.1
Intermediate Shell Longitudinal Welds 2-203A, B, & C	Multiple	68.0	1.12×10^{19}	1.0327	-56	70.2	17.0	28.0	65.5	79.7
Lower Shell Longitudinal Welds 3-203A, B, & C	10120	34.0	1.14×10^{19}	1.0369	-56	35.3	17.0	17.6	49.0	28.2
Intermediate to Lower Shell Girth Weld 9-203	83650	34.1	1.18×10^{19}	1.0474	-40	35.7	0.0	17.9	35.7	31.4
Using credible Arkansas Nuclear One Unit 2 surveillance data	83650	12.1	1.18×10^{19}	1.0474	-40	12.7	0.0	6.3	12.7	-14.7
Reactor Vessel Extended Beltline Materials										
Upper Shell Plate C-8008-1	C8182-1	91.0	0.0143×10^{19}	0.1389	12.2	12.6	0.0	6.3	12.6	37.5
Using credible Arkansas Nuclear One Unit 2 surveillance data	C8182-1	55.8	0.0143×10^{19}	0.1389	12.2	7.8	0.0	3.9	7.8	27.7
Upper Shell Plate C-8008-2	C7605-1	89.5	0.0143×10^{19}	0.1389	60.5	12.4	0.0	6.2	12.4	85.4
Upper Shell Plate C-8008-3	C8571-2	51.0	0.0143×10^{19}	0.1389	27.3	7.1	0.0	3.5	7.1	41.5
Upper Shell Longitudinal Welds 1-203A, B, & C	BOLA	27.0	0.0143×10^{19}	0.1389	-60	3.8	0.0	1.9	3.8	-52.5
Using non-credible J.M. Farley Unit 2 surveillance data	BOLA	16.2	0.0143×10^{19}	0.1389	-60	2.3	0.0	1.1	2.3	-55.5

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Table 7-3 Adjusted Reference Temperature Evaluation for the Arkansas Nuclear One Unit 2 Reactor Vessel Beltline Materials through 54 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} (°F)	σ_I[(a)] (°F)	σ_Δ[(b)] (°F)	Margin (°F)	ART[(c)] (°F)
Upper to Intermediate Shell Girth Weld 8-203	10137	98.5	0.0143×10^{19}	0.1389	-56	13.7	17.0	6.8	36.6	-5.7
	Using credible Calvert Cliffs Unit 2 and Millstone Unit 2 data	61.2	0.0143×10^{19}	0.1389	-56	8.5	17.0	28.0	65.5	18.0
	6329637	100.8	0.0143×10^{19}	0.1389	-56	14.0	17.0	7.0	36.8	-5.2
	FAGA	41.0	0.0143×10^{19}	0.1389	-24	5.7	0.0	2.8	5.7	-12.6

Notes:

(a) The plate material initial RT_{NDT} values are measured values. For weld materials with generic initial RT_{NDT} values, σ_I = 17°F. For weld materials with measured initial RT_{NDT} values, σ_I = 0°F.

(b) As discussed in Section 4, the surveillance plate and weld Heat # 83650 data were deemed credible, while the weld Heat # BOLA data were deemed non-credible. The surveillance weld data for Heat # 10137 was deemed credible; however, per Section 4 and Appendix D, a full margin term will be used. Per the guidance of Regulatory Guide 1.99, Revision 2 [Ref. 1], the base metal σ_Δ = 17°F for Position 1.1 and σ_Δ = 8.5°F for Position 2.1 with credible surveillance data. The weld metal σ_Δ = 28°F for Position 1.1 and 2.1 with non-credible surveillance data (Heat # BOLA), and the weld metal σ_Δ = 14°F for Position 2.1 with credible surveillance data (Heat # 83650). Since a full margin term will be used for Heat # 10137, σ_Δ = 28°F with credible surveillance data for Position 2.1 for this weld heat. However, σ_Δ need not exceed $0.5^*\Delta RT_{NDT}$.

(c) The Regulatory Guide 1.99, Revision 2 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 Arkansas Nuclear One Unit 2 Heatup and Cooldown Curves at 54 EFPY

1/4T Limiting ART[a]	3/4T Limiting ART[a]
122°F	109°F
Lower Shell Plate C-8010-1 (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 up and increased by 5°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 50, 60, 70, and 80°F/hr applicable for 54 EFPY, with the flange and lowest service temperature 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, 25, 60, and 100°F/hr applicable for 54 EFPY, with the flange and lowest service temperature 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 first 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 (see Figure 8-3 and Table 8-3). The governing equation for the hydrostatic test is defined in the 1998 through the 2000 Addenda ASME Code Section XI, Appendix G as follows:

$$1.5\ K_{Im} < K_{Ic}$$

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 test at 2485 psig for the Arkansas Nuclear One Unit 2 reactor vessel at 54 EFPY is

179°F. 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, 8-2, and 8-3 define all of the above limits for ensuring prevention of non-ductile failure for the Arkansas Nuclear One Unit 2 reactor vessel for 54 EFPY with the flange and lowest service temperature 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 data points used for developing the inservice hydrostatic and leak test P-T limit curve shown in Figure 8-3 are presented in Table 8-3. The P-T limit curves shown in Figures 8-1, 8-2, and 8-3 were generated based on the limiting ART values for the cylindrical beltline and extended beltline reactor vessel materials rounded up and increased by 5°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 Arkansas Nuclear One Unit 2 at 54 EFPY.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Lower Shell Plate C-8010-1 using Regulatory Guide 1.99 Position 1.1 data
LIMITING ART VALUES AT 54 EFPY: 1/4T, 122°F (Axial Flaw)
3/4T, 109°F (Axial Flaw)

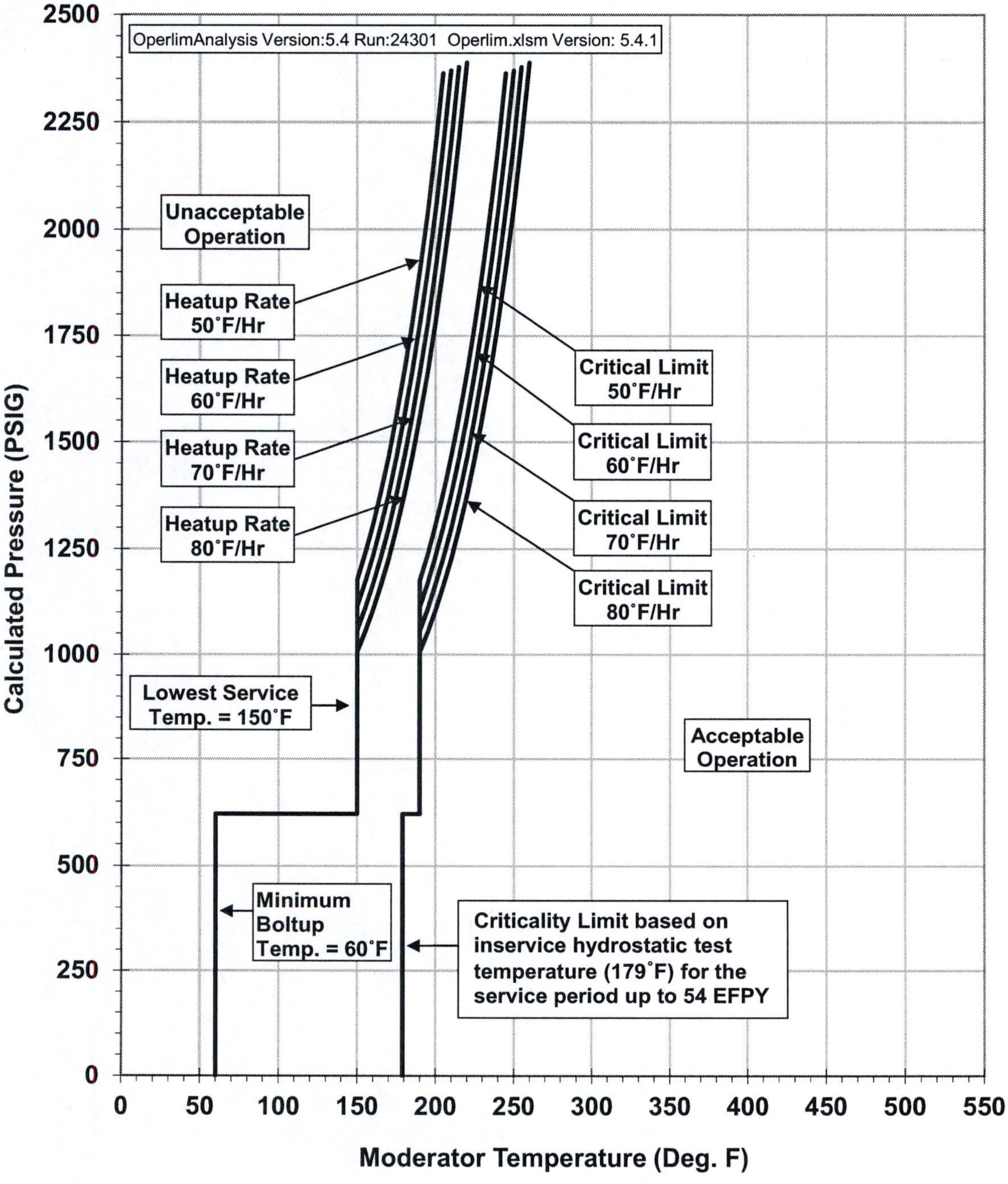

Figure 8-1 Arkansas Nuclear One Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rates of 50, 60, 70, and 80°F/hr) Applicable for 54 EFPY (with Flange and LST Requirements and without Margins for Instrumentation Errors) using the 1998 through the 2000 Addenda App. G Methodology (w/ K_{Ic})

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Lower Shell Plate C-8010-1 using Regulatory Guide 1.99 Position 1.1 data
LIMITING ART VALUES AT 54 EFPY: 1/4T, 122°F (Axial Flaw)
3/4T, 109°F (Axial Flaw)

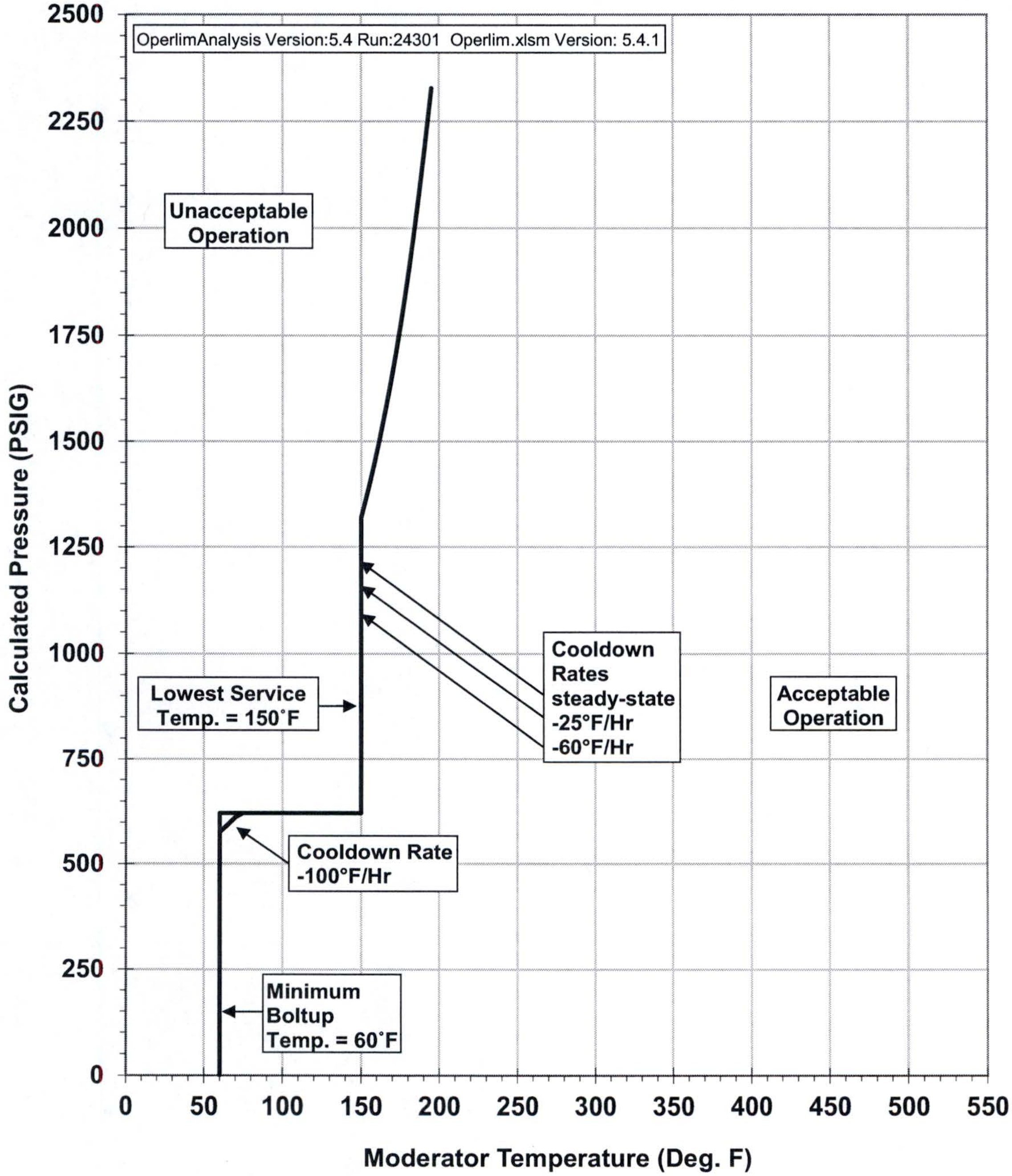

Figure 8-2 Arkansas Nuclear One Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 60, and 100°F/hr) Applicable for 54 EFPY (with Flange and LST Requirements and without Margins for Instrumentation Errors) using the 1998 through the 2000 Addenda App. G Methodology (w/ K_{Ic})

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Lower Shell Plate C-8010-1 using Regulatory Guide 1.99 Position 1.1 data
LIMITING ART VALUES AT 54 EFPY: 1/4T, 122°F (Axial Flaw)
3/4T, 109°F (Axial Flaw)

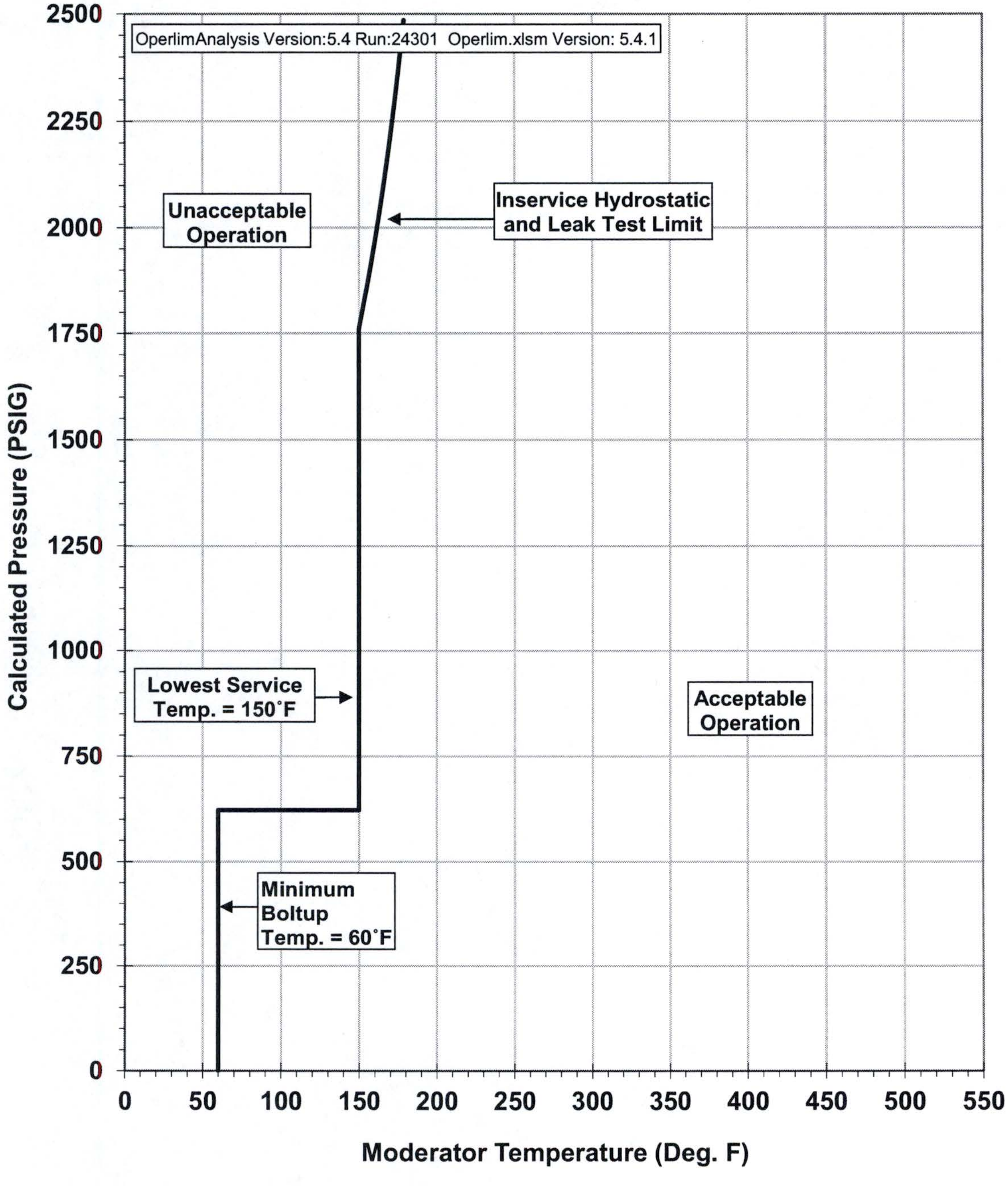

Figure 8-3 Arkansas Nuclear One Unit 2 Reactor Coolant System Inservice Hydrostatic and Leak Test Limitations Applicable for 54 EFPY (with Flange and LST Requirements and without Margins for Instrumentation Errors) using the 1998 through the 2000 Addenda App. G Methodology (w/ K_{Ic})

Table 8-1 Arkansas Nuclear One Unit 2 54 EFPY Heatup Curve Data Points using the 1998 through the 2000 Addenda App. G Methodology (w/ K_{Ic}, w/ Flange and LST Requirements, and w/o Margins for Instrumentation Errors)

50°F/hr Heatup		50°F/hr Criticality		60°F/hr Heatup		60°F/hr Criticality		70°F/hr Heatup		70°F/hr Criticality		80°F/hr Heatup		80°F/hr Criticality	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
60	0	179	0	60	0	179	0	60	0	179	0	60	0	179	0
60	622	179	622	60	622	179	622	60	622	179	622	60	622	179	622
65	622	180	622	65	622	180	622	65	622	180	622	65	622	180	622
70	622	185	622	70	622	185	622	70	622	185	622	70	622	185	622
75	622	190	622	75	622	190	622	75	622	190	622	75	622	190	622
80	622	190	1176	80	622	190	1114	80	622	190	1057	80	622	190	1007
85	622	195	1238	85	622	195	1170	85	622	195	1109	85	622	195	1054
90	622	200	1308	90	622	200	1234	90	622	200	1166	90	622	200	1106
95	622	205	1384	95	622	205	1303	95	622	205	1230	95	622	205	1164
100	622	210	1468	100	622	210	1380	100	622	210	1300	100	622	210	1228
105	622	215	1561	105	622	215	1466	105	622	215	1378	105	622	215	1299
110	622	220	1664	110	622	220	1560	110	622	220	1464	110	622	220	1378
115	622	225	1778	115	622	225	1664	115	622	225	1559	115	622	225	1465
120	622	230	1903	120	622	230	1778	120	622	230	1664	120	622	230	1561
125	622	235	2042	125	622	235	1905	125	622	235	1780	125	622	235	1667
130	622	240	2195	130	622	240	2045	130	622	240	1908	130	622	240	1784
135	622	245	2363	135	622	245	2200	135	622	245	2050	135	622	245	1913
140	622			140	622	250	2370	140	622	250	2206	140	622	250	2056
145	622			145	622			145	622	255	2378	145	622	255	2214
150	622			150	622			150	622			150	622	260	2388
150	1176			150	1114			150	1057			150	1007		
155	1238			155	1170			155	1109			155	1054		

Table 8-1 Arkansas Nuclear One Unit 2 54 EFPY Heatup Curve Data Points using the 1998 through the 2000 Addenda App. G Methodology (w/ K_{Ic}, w/ Flange and LST Requirements, and w/o Margins for Instrumentation Errors)

50°F/hr Heatup		50°F/hr Criticality		60°F/hr Heatup		60°F/hr Criticality		70°F/hr Heatup		70°F/hr Criticality		80°F/hr Heatup		80°F/hr Criticality	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
160	1308			160	1234			160	1166			160	1106		
165	1384			165	1303			165	1230			165	1164		
170	1468			170	1380			170	1300			170	1228		
175	1561			175	1466			175	1378			175	1299		
180	1664			180	1560			180	1464			180	1378		
185	1778			185	1664			185	1559			185	1465		
190	1903			190	1778			190	1664			190	1561		
195	2042			195	1905			195	1780			195	1667		
200	2195			200	2045			200	1908			200	1784		
205	2363			205	2200			205	2050			205	1913		
				210	2370			210	2206			210	2056		
								215	2378			215	2214		
												220	2388		

Table 8-2 Arkansas Nuclear One Unit 2 54 EFPY Cooldown Curve Data Points using the 1998 through the 2000 Addenda App. G Methodology (w/ K_{Ic}, w/ Flange and LST Requirements, and w/o Margins for Instrumentation Errors)

Steady-State		-25°F/hr.		-60°F/hr.		-100°F/hr.	
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	622	60	622	60	622	60	576
65	622	65	622	65	622	65	594
70	622	70	622	70	622	70	613
75	622	75	622	75	622	75	622
80	622	80	622	80	622	80	622
85	622	85	622	85	622	85	622
90	622	90	622	90	622	90	622
95	622	95	622	95	622	95	622
100	622	100	622	100	622	100	622
105	622	105	622	105	622	105	622
110	622	110	622	110	622	110	622
115	622	115	622	115	622	115	622
120	622	120	622	120	622	120	622
125	622	125	622	125	622	125	622
130	622	130	622	130	622	130	622
135	622	135	622	135	622	135	622
140	622	140	622	140	622	140	622
145	622	145	622	145	622	145	622
150	622	150	622	150	622	150	622
150	1321	150	1321	150	1321	150	1321
155	1393	155	1393	155	1393	155	1393
160	1474	160	1474	160	1474	160	1474
165	1562	165	1562	165	1562	165	1562
170	1660	170	1660	170	1660	170	1660
175	1768	175	1768	175	1768	175	1768
180	1888	180	1888	180	1888	180	1888
185	2020	185	2020	185	2020	185	2020
190	2166	190	2166	190	2166	190	2166
195	2328	195	2328	195	2328	195	2328

Table 8-3 Arkansas Nuclear One Unit 2 54 EFPY Inservice Hydrostatic and Leak Test Curve Data Points using the 1998 through the 2000 Addenda App. G Methodology (w/ K_{Ic}, w/ Flange and LST Requirements, and w/o Margins for Instrumentation Errors)

T (°F)	P (psig)
60	0
60	622
65	622
70	622
75	622
80	622
85	622
90	622
95	622
100	622
105	622
110	622
115	622
120	622
125	622
130	622
135	622
140	622
145	622
150	622
150	1761
155	1858
160	1965
162	**2000**
165	2083
170	2214
175	2358
179	**2485**

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APPENDIX A THERMAL STRESS INTENSITY FACTORS (K_{It})

Tables A-1 and A-2 contain the thermal stress intensity factors (K_{It}) for the maximum heatup and cooldown rates at 54 EFPY for Arkansas Nuclear One Unit 2. The reactor vessel cylindrical shell radii to the 1/4T and 3/4T locations are as follows:

- 1/4T Radius = 81.688 inches
- 3/4T Radius = 85.625 inches

Table A-1 K_{It} Values for Arkansas Nuclear One Unit 2 at 54 EFPY 80°F/hr Heatup Curves (w/ Flange and LST Requirements, and w/o Margins for Instrument Errors)

Water Temp. (°F)	Vessel Temperature at 1/4T Location for 80°F/hr Heatup (°F)	1/4T Thermal Stress Intensity Factor (ksi √in.)	Vessel Temperature at 3/4T Location for 80°F/hr Heatup (°F)	3/4T Thermal Stress Intensity Factor (ksi √in.)
60	56.321	-1.028	55.105	0.545
65	59.387	-2.397	55.641	1.540
70	62.751	-3.449	56.862	2.407
75	66.385	-4.400	58.707	3.149
80	70.275	-5.155	61.052	3.764
85	74.297	-5.816	63.832	4.283
90	78.513	-6.351	66.967	4.716
95	82.818	-6.820	70.400	5.083
100	87.260	-7.203	74.080	5.390
105	91.763	-7.540	77.960	5.654
110	96.358	-7.818	82.009	5.878
115	100.997	-8.066	86.196	6.070
120	105.699	-8.271	90.500	6.236
125	110.432	-8.456	94.900	6.379
130	115.207	-8.611	99.381	6.503
135	120.006	-8.753	103.929	6.613
140	124.832	-8.872	108.533	6.709
145	129.675	-8.984	113.183	6.794
150	134.537	-9.080	117.872	6.871
155	139.412	-9.170	122.594	6.940
160	144.298	-9.249	127.343	7.002
165	149.195	-9.324	132.114	7.060
170	154.099	-9.391	136.904	7.113
175	159.011	-9.457	141.710	7.162
180	163.927	-9.516	146.529	7.209
185	168.849	-9.574	151.358	7.253
190	173.774	-9.627	156.198	7.294
195	178.703	-9.681	161.044	7.334
200	183.634	-9.730	165.897	7.373
205	188.568	-9.780	170.756	7.410
210	193.503	-9.826	175.619	7.446

Table A-2 K_{It} Values for Arkansas Nuclear One Unit 2 at 54 EFPY 100°F/hr Cooldown Curves (w/ Flange and LST 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	232.413	13.504
205	227.339	13.448
200	222.265	13.392
195	217.192	13.336
190	212.118	13.281
185	207.044	13.225
180	201.970	13.169
175	196.897	13.113
170	191.823	13.058
165	186.749	13.002
160	181.676	12.947
155	176.602	12.891
150	171.529	12.836
145	166.456	12.781
140	161.383	12.726
135	156.310	12.671
130	151.237	12.616
125	146.164	12.561
120	141.091	12.507
115	136.018	12.452
110	130.946	12.398
105	125.874	12.343
100	120.801	12.289
95	115.729	12.235
90	110.657	12.182
85	105.585	12.128
80	100.514	12.074
75	95.442	12.020
70	90.371	11.967
65	85.299	11.914
60	80.229	11.860

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 Arkansas Nuclear One 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 Arkansas Nuclear One 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, and weight percent (wt. %) copper (Cu) and nickel (Ni), initial RT_{NDT} value, 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 Arkansas Nuclear One Unit 2 is presented in Table B-2.

Nozzle Material Properties

The Arkansas Nuclear One Unit 2 nozzle material properties are provided in Table B-1. Nickel (Ni), Manganese (Mn), and Phosphorus (P) weight percent (wt. %) values were obtained as the average of the material-specific analyses documented in Combustion Engineering report A-PENG-ER-002 [Ref. B-4] for

each of the Arkansas Nuclear One Unit 2 reactor vessel inlet and outlet nozzles. Copper weight percent values for each of the Arkansas Nuclear One Unit 2 outlet nozzles were also taken to be the average of all available analyses contained in Combustion Engineering report A-PENG-ER-002 [Ref. B-4]. However, the A-PENG-ER-002 report did not contain copper weight percent values for the inlet nozzles, because at the time that the Arkansas Nuclear One Unit 2 nozzles were manufactured, these values were not required to be documented for SA-508, Class 2 low-alloy steel. Therefore, no material-specific copper weight percent value is available for the Arkansas Nuclear One Unit 2 inlet nozzles. Per NRC RIS 2014-11 [Ref. B-1], a copper weight percent value is not required for calculation of the Arkansas Nuclear One Unit 2 nozzle material ART values, because the nozzles have fluence values less than 1 x 10^{17} n/cm^2. However, if a copper weight percent value is ever needed, a best-estimate copper weight percent value is available from Section 4 of the NRC-approved Boiling Water Reactor Vessel and Internals Project (BWRVIP [proprietary]) report, BWRVIP-173-A [Ref. B-5], and this value could be utilized for the Arkansas Nuclear One inlet nozzles. A mean plus two standard deviations methodology was applied to the data in BWRVIP-173-A to determine a conservative copper weight percent value. The data in the BWRVIP report was tabulated from an industry-wide database of SA-508, Class 2 forging materials.

The Charpy V-Notch forging specimen orientation for the inlet and outlet nozzles was not reported in A-PENG-ER-002; thus, it was conservatively assumed that the orientation was the "strong direction" for each nozzle forging. The initial RT_{NDT} values were therefore determined for each of the Arkansas Nuclear One 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-6]. The initial RT_{NDT} values for all of the nozzle materials were determined directly from the data or by using a CVGRAPH, Version 6.02 hyperbolic tangent curve fit through the minimum data points, in accordance with ASME Code Section III, Subarticle NB-2331, Paragraph (a)(4) [Ref. B-7]. 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 Arkansas Nuclear One Unit 2 initial RT_{NDT} 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 Arkansas Nuclear One 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 2-2, the inlet nozzles are determined to receive a projected maximum fluence of 7.96 x 10^{16} n/cm^2 (E > 1 MeV) at the lowest extent of the nozzles at 54 EFPY. Similarly, the outlet nozzles are projected to achieve a maximum fluence value of 9.80 x 10^{16} n/cm^2 (E > 1 MeV) at the lowest extent of the nozzles at 54 EFPY. Thus, the maximum neutron fluence values for the nozzle materials are not projected to exceed a fluence of 1 x 10^{17} n/cm^2 at 54 EFPY. Per NRC RIS 2014-11 [Ref. B-1], embrittlement of reactor vessel materials, with projected fluence values less than 1 x 10^{17} n/cm^2, 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 Arkansas Nuclear One Unit 2 inlet and outlet nozzle forging materials are summarized in Table B-1.

Table B-1 Summary of the Arkansas Nuclear One Unit 2 Reactor Vessel Nozzle Material Initial RT_{NDT}, Chemistry, and Fluence Values at 54 EFPY

Reactor Vessel Material	Chemical Composition[(a)]				$RT_{NDT(U)}$[(c)] (°F)	Fluence at Lowest Extent of Nozzle[(d)] (n/cm^2, E > 1.0 MeV)
	Wt. % Cu	Wt. % Ni	Wt. % Mn	Wt. % P		
Inlet Nozzle C-8015-1	Note (b)	0.69	0.65	0.007	30	7.96×10^{16}
Inlet Nozzle C-8015-2	Note (b)	0.60	0.70	0.010	10	7.96×10^{16}
Inlet Nozzle C-8015-3	Note (b)	0.69	0.65	0.006	10	7.96×10^{16}
Inlet Nozzle C-8015-4	Note (b)	0.65	0.70	0.007	30	7.96×10^{16}
Outlet Nozzle C-8016-1	0.12	0.63	0.64	0.006	0	9.80×10^{16}
Outlet Nozzle C-8016-2	0.17	0.76	0.83	0.017	13.5	9.80×10^{16}

Notes:

(a) Chemistry values are the average of all available material-specific chemical analyses, unless otherwise noted.
(b) The Arkansas Nuclear One Unit 2 copper weight percent values are not documented in the historical records. This value is not needed for the current analysis per NRC RIS 2014-11 [Ref. B-1], since the fluence values for these materials are below 1.0×10^{17} n/cm^2. If a copper weight percent value is needed in the future, a best-estimate copper weight percent value is available from Section 4 of the NRC-approved BWRVIP (proprietary) report, BWRVIP-173-A [Ref. B-5].
(c) $RT_{NDT(U)}$ values were determined using NUREG-0800, BTP 5-3 Position 1.1(3)(a) and (b) [Ref. B-6] methodology with the more limiting $RT_{NDT(U)}$ value being selected for each nozzle material.
(d) Fluence values conservatively correspond to 54 EFPY fluence values at the lowest extent of the nozzle weld.

Table B-2 Summary of the Limiting ART Values for the Arkansas Nuclear One Unit 2 Inlet and Outlet Nozzle Materials

EFPY	Nozzle Material and ID Number	Limiting ART Value (°F)
54	Inlet Nozzle C-8015-1 and C-8015-4	30
	Outlet Nozzle C-8016-2	13.5

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 Arkansas Nuclear One Unit 2 nozzle material ART values should be re-evaluated.

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 Arkansas Nuclear One 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 ORNL study, ORNL/TM-2010/246 [Ref. B-8], and are consistent with ASME PVP2011-57015 [Ref. B-9]. The methodology includes postulating an inside surface 1/4T nozzle corner flaw, and calculating through-wall nozzle corner stresses for a cooldown rate of 100°F/hour.

The through-wall stresses at the nozzle corner location were fitted based on a third-order polynomial of the form:

$$\sigma = A_0 + A_1 x + A_2 x^2 + A_3 x^3$$

where,

σ = through-wall stress distribution

x = through-wall distance from inside surface

A_0, A_1, A_2, A_3 = coefficients of polynomial fit for the third-order polynomial, used in the stress intensity factor expression discussed below

The stress intensity factors generated for a rounded nozzle corner for the pressure and thermal gradient 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 Arkansas Nuclear One 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 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 beltline curves. Therefore, the P-T limits provided in Section 8 for 54 EFPY remain limiting for the beltline and non-beltline reactor vessel components.

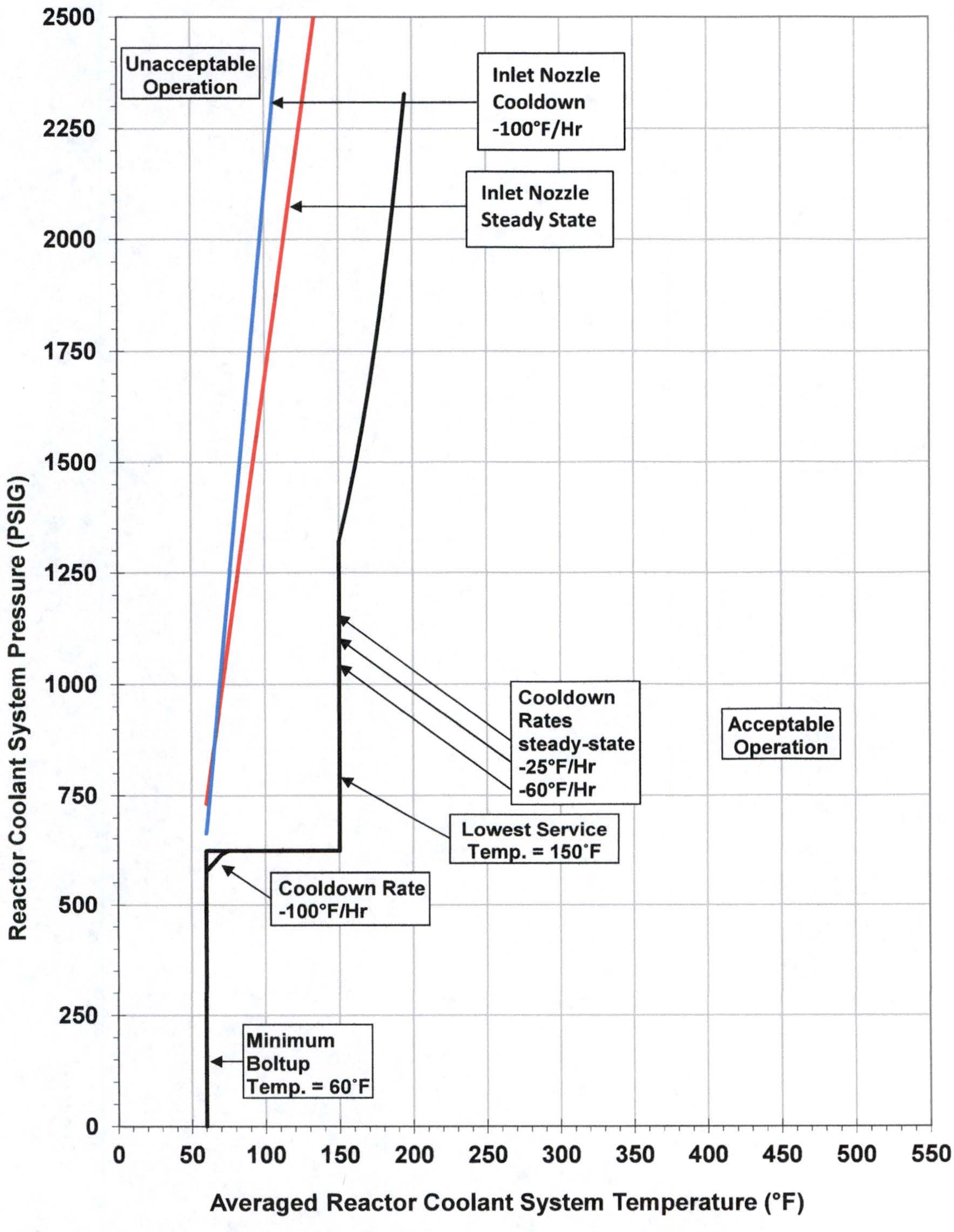

Figure B-1 Comparison of Arkansas Nuclear One Unit 2 Beltline P-T Limits to Inlet Nozzle Limits

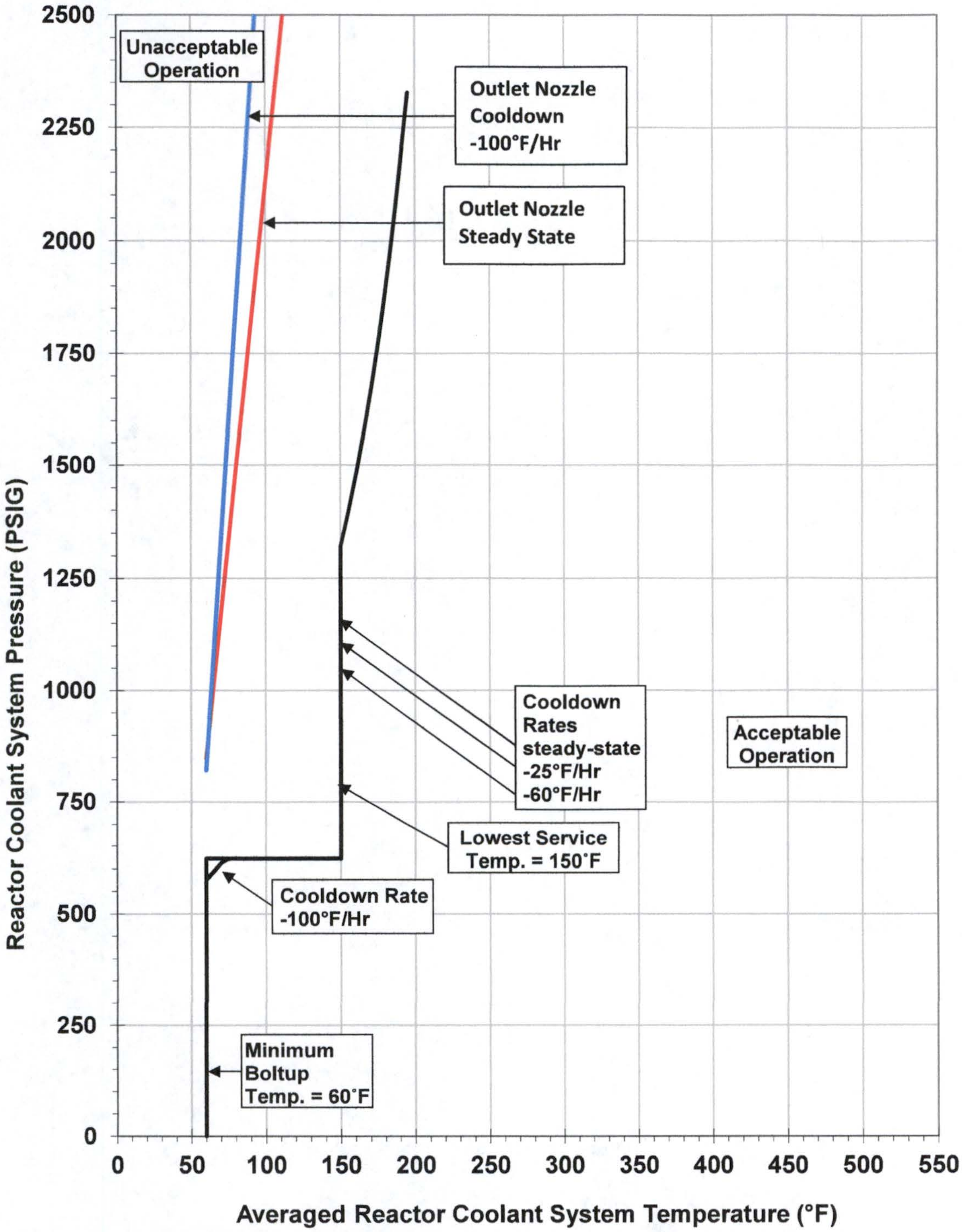

Figure B-2 Comparison of Arkansas Nuclear One Unit 2 Beltline P-T Limits to Outlet Nozzle Limits

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 Combustion Engineering Report A-PENG-ER-002, Revision 0, "The Reactor Vessel Group Records Evaluation Program Phase II Final Report for the ANO 2 Reactor Pressure Vessel Plates, Forgings, Welds and Cladding," October 1995.

B-5 *BWRVIP-173-A: BWR Vessel and Internals Project: Evaluation of Chemistry Data for BWR Vessel Nozzle Forging Materials.* EPRI, Palo Alto, CA: 2011. 1022835.

B-6 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-7 ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Division 1, Subsection NB, "Class 1 Components."

B-8 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-9 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 NON-REACTOR VESSEL 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 for all RCPB components, which is specified in NB-3211 and NB-2332 of the Section III ASME Code, is the relevant requirement that would affect the pressure-temperature (P-T) limits. The lowest service temperature (LST) requirement of NB-3211 and NB-2332 of the Section III ASME Code is applicable to material for ferritic piping, pumps and valves with a nominal wall thickness greater than 2 ½ inches [Ref. C-2]. Arkansas Nuclear One Unit 2 reactor coolant system piping contains ferritic materials in the Class 1 piping; the pumps and valves do not contain ferritic material. The LST requirements of NB-3211 and NB-2332 are considered in Section 6.4 of this report. The other ferritic RCPB components that are not part of the reactor vessel consist of the replacement closure head, the pressurizer and the replacement steam generators.

The replacement closure head is considered in the cylindrical beltline P-T limit curves as described in Section 6 of this report. Furthermore, the replacement closure head has 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 pressurizer was constructed to the 1968 Edition through 1970 Summer Addenda Section III ASME Code and met all applicable requirements at the time of construction and is original to the plant. Furthermore, the pressurizer has 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 replacement steam generators were constructed to the 1989 Edition Section III ASME Code and met all applicable requirements at the time of construction. Furthermore, the replacement steam generators have not undergone neutron embrittlement that would affect P-T limits. Therefore, 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, December 19, 1995.

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

APPENDIX D CREDIBILITY EVALUATION OF THE WELD HEAT # 10137 SURVEILLANCE DATA

D.1 INTRODUCTION

Regulatory Guide 1.99, Revision 2 [Ref. D-1] describes general procedures acceptable to the NRC staff for calculating the effects of neutron radiation embrittlement of the low-alloy steels currently used for light-water-cooled reactor vessels. Position 2.1 of Regulatory Guide 1.99, Revision 2, describes the method for calculating the adjusted reference temperature of reactor vessel beltline materials using surveillance capsule data. The methods of Position 2.1 can only be applied when two or more credible surveillance data sets become available from the reactor in question.

The credibility of all surveillance program data applicable to the Arkansas Nuclear One Unit 2 beltline was assessed in WCAP-18166-NP [Ref. D-2]. However, the Arkansas Nuclear One Unit 2 extended beltline contains two welds with sister plant data, the Upper Shell Longitudinal Welds 1-203A, B, & C (Heat # BOLA) and the Upper to Intermediate Shell Girth Weld 8-203 (Heat # 10137). Note that no surveillance data is available for the other two Heats (# 6329637 and # FAGA) which were also used to make the Upper to Intermediate Shell Girth Weld 8-203.

The weld Heat # BOLA sister plant data is available from the J.M. Farley Unit 2 surveillance program. Since this surveillance data is analyzed by itself, the credibility conclusion documented in Appendix D of WCAP-16918, Revision 1 [Ref. D-3] is applicable to Arkansas Nuclear One Unit 2; thus, the credibility conclusion of the Heat # BOLA data need not be updated. The J.M. Farley Unit 2 surveillance weld data (Heat # BOLA) is non-credible in regard to the Arkansas Nuclear One Unit 2 reactor vessel materials.

The weld Heat # 10137 sister plant data is available from both the Calvert Cliffs Unit 2 and Millstone Unit 2 surveillance programs. The Millstone Unit 2 surveillance program includes two distinct welds, Heat # 10137 and Heat # 90136. In previous analyses, this weld surveillance data was treated as one combined weld and subsequently analyzed together. However, these two weld metal heats were not melted together into a tandem weld; they were individually deposited. It cannot be determined with full confidence how much of the overall surveillance weld is which weld metal heat and, furthermore, exactly which weld heat specimens are contained in which surveillance capsules in the Millstone Unit 2 program.

The Millstone Unit 2 (combined) surveillance weld data met the second and third credibility criteria of Regulatory Guide 1.99, Revision 2 [Ref. D-1]. Additionally, Table D-2 of WCAP-16012 [Ref. D-4] indicates that all of the measured weld ΔRT_{NDT} values were within the 1-sigma scatter band; therefore, suggesting that there is good agreement between the measured capsule data and the embrittlement correlations. If the two heats of weld material were evaluated individually, one would expect that the scatter in the data would decrease since the irradiated material would embrittle differently for the two separate welds with different, as-measured, copper and nickel contents. However, since the (combined) weld material already passes the Regulatory Guide 1.99, Revision 2 credibility analysis, a re-evaluation of the material (as two separate heats) is not expected to significantly change the overall results of the subsequent reactor vessel integrity analyses. Thus, the surveillance weld metal will be considered to be only Heat # 10137 for the evaluations contained herein. All currently determined input data for Position 2.1 chemistry factor determination (See Section 5) and surveillance data credibility assessment

documented in this Appendix will be used "as-is," as documented in the Millstone Unit 2 surveillance capsule analyses of record.

For conservatism, no reduction in the margin term of Regulatory Guide 1.99, Revision 2 [Ref. D-1] was taken to account for the additional uncertainties, despite the data remaining credible (see Section D.2). Note that this approach should also be followed when completing analyses per 10 CFR 50.61 [Ref. D-5]. Despite this additional conservatism, the Arkansas Nuclear One Unit 2 Upper to Intermediate Shell Girth Weld 8-203 (Heat # 10137) was not the limiting material for the Arkansas Nuclear One Unit 2 P-T limit curves.

D.2 EVALUATION

Per Appendix D of WCAP-17501-NP [Ref. D-6], the Calvert Cliffs Unit 2 surveillance weld data (Heat # 10137) was deemed credible, and per Appendix D of WCAP-16012 [Ref. D-4], the Millstone Unit 2 surveillance weld data (Heat # 10137) was also deemed credible. Thus, when analyzed individually, these surveillance welds pass all five of the Regulatory Guide 1.99, Revision 2 [Ref. D-1] credibility criterion. The only credibility criterion that must be updated as a result of analyzing the two surveillance welds together is Criterion 3. This evaluation is documented herein.

Criterion 3: When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Guide 1.99, Revision 2 [Ref. D-1] normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail this criterion for use in shift calculations, they may be credible for determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E185-82 [Ref. D-7].

The functional form of the least-squares method as described in Regulatory Guide 1.99, Revision 2 will be utilized to determine a best-fit line for this data and to determine if the scatter of these ΔRT_{NDT} values about this line is less than 28°F for the weld.

Following is the calculation of the best-fit line as described in Reference D-1. In addition, the recommended NRC methods for determining credibility will be followed. The NRC methods were presented to industry at a meeting held by the NRC on February 12 and 13, 1998 [Ref. D-8]. At this meeting the NRC presented five cases. Of the five cases, Case 5 ("Surveillance Data from Other Sources Only") most closely represents the situation for the Arkansas Nuclear One Unit 2 reactor vessel Upper to Intermediate Shell Girth Weld 8-203 (Heat # 10137) as described below:

Heat # 10137 (Case 5) – This weld heat pertains to the Upper to Intermediate Shell Girth Weld 8-203 in the Arkansas Nuclear One Unit 2 reactor vessel. This weld heat is not contained in the Arkansas Nuclear One Unit 2 surveillance program. However, it is contained in the Calvert Cliffs Unit 2 and Millstone Unit 2 surveillance programs. NRC Case 5 per Reference D-8 is entitled "Surveillance Data from Other Sources Only" and most closely represents the situation for Arkansas Nuclear One Unit 2 weld Heat # 10137.

Credibility Assessment Case 5: Weld Heat # 10137 (Calvert Cliffs Unit 2 Data Only)

Following the NRC Case 5 guidelines, the Calvert Cliffs Unit 2 and Millstone Unit 2 surveillance weld metal (Heat # 10137) will be evaluated for credibility. Weld Heat # 10137 pertains to Arkansas Nuclear One Unit 2 reactor vessel Upper to Intermediate Shell Girth Weld 8-203, but is not contained in the Arkansas Nuclear One Unit 2 surveillance program.

In accordance with the NRC Case 5 guidelines, the data from only Calvert Cliffs Unit 2 will be analyzed first, since the irradiation environment for Calvert Cliffs Unit 2 is judged closer to that of Arkansas Nuclear One Unit 2 as evidenced by the temperature adjustments documented in Table 4-2. This assessment was performed in Appendix D of WCAP-17501-NP [Ref. D-6] and concluded that the surveillance data for Heat # 10137 from Calvert Cliffs Unit 2 only was credible. Therefore, in accordance with Case 5, the combined data from both Calvert Cliffs Unit 2 and Millstone Unit 2 will now be assessed to determine the credibility conclusion for all applicable data for weld Heat # 10137.

Credibility Assessment Case 5: Weld Heat # 10137 (All data)

In accordance with the NRC Case 5 guidelines, the data from Calvert Cliffs Unit 2 and Millstone Unit 2 will now be analyzed together. Data is adjusted to the mean chemical composition and operating temperature of the surveillance capsules. This is performed in Table D-1.

Table D-1 Mean Chemical Composition and Operating Temperature for Calvert Cliffs Unit 2 and Millstone Unit 2

Material	Capsule	Cu Wt. %[(a)]	Ni Wt. %[(a)]	Inlet Temperature during Period of Irradiation (°F)[(b)]
Weld Metal Heat # 10137 (Calvert Cliffs Unit 2 Data)	263°	0.21	0.06	550
	97°			549
	104°			548
Weld Metal Heat # 10137 (Millstone Unit 2 Data)	97°	0.21	0.06	544.3
	104°			547.6
	83°			548.0
MEAN		0.21	0.06	547.8

Note:

(a) Chemistry data obtained from Table 3-1.
(b) Temperature data obtained from Table 4-2.

Since the mean chemical composition of the surveillance capsule data is identical to the actual chemical composition data for each capsule, no chemistry adjustment is necessary. However, since the Calvert Cliffs Unit 2 and Millstone Unit 2 surveillance capsule operating temperatures are not identical to the mean operating temperature, the surveillance capsule data will be adjusted to the mean operating temperature. The capsule-specific temperature adjustments are as shown in Table D-2.

Table D-2 Operating Temperature Adjustments for the Calvert Cliffs Unit 2 and Millstone Unit 2 Surveillance Capsule Data

Material	Capsule	Inlet Temperature during Period of Irradiation (°F)	Mean Operating Temperature (°F)	Temperature Adjustment (°F)
Weld Metal Heat # 10137 (Calvert Cliffs Unit 2 Data)	263°	550	547.8	2.2
	97°	549		1.2
	104°	548		0.2
Weld Metal Heat # 10137 (Millstone Unit 2 Data)	97°	544.3		-3.5
	104°	547.6		-0.2
	83°	548.0		0.2

Using the chemical composition and operating temperature adjustments described and calculated above, an interim chemistry factor is calculated for weld Heat # 10137 using the Calvert Cliffs Unit 2 and Millstone Unit 2 data. This calculation is shown on the following page in Table D-3.

Table D-3 Calculation of Weld Heat # 10137 Interim Chemistry Factor for the Credibility Evaluation Using Calvert Cliffs Unit 2 and Millstone Unit 2 Surveillance Capsule Data

Material	Capsule	Capsule f[(a)] (x 10^{19} n/cm², E > 1.0 MeV)	FF[(b)]	ΔRT_{NDT}[(c)] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Weld Metal Heat # 10137 (Calvert Cliffs Unit 2 Data)	263°	0.825	0.9460	74.9 (72.7)	70.86	0.895
	97°	1.95	1.1825	84.1 (82.9)	99.45	1.398
	104°	2.44	1.2401	69.9 (69.7)	86.68	1.538
Weld Metal Heat # 10137 (Millstone Unit 2 Data)	97°	0.324	0.6902	62.4 (65.93)	43.09	0.476
	104°	0.949	0.9853	51.9 (52.12)	51.16	0.971
	83°	1.740	1.1523	56.3 (56.09)	64.86	1.328
				SUM:	416.10	6.606
	$CF_{Heat\ \#\ 10137} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (416.10) \div (6.606) =$ **63.0°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. Each ΔRT_{NDT} value has been adjusted according to the temperature adjustments summarized in Table D-2. The ΔRT_{NDT} values for each surveillance weld data point are not adjusted by the ratio procedure, because the mean chemical composition is identical to each capsule chemical composition (pre-adjusted values are listed in parentheses and were taken from Table 4-2).

The scatter of ΔRT_{NDT} values about the functional form of a best-fit line drawn as described in Regulatory Guide 1.99, Revision 2, Position 2.1 [Ref. D-1] is presented in Table D-4.

Table D-4 Best-Fit Evaluation for Surveillance Weld Metal Heat # 10137 Using Calvert Cliffs Unit 2 and Millstone Unit 2 Data

Material	Capsule	CF ($Slope_{best-fit}$) (°F)	Capsule f (x 10^{19} n/cm², E > 1.0 MeV)	FF	Measured ΔRT_{NDT} (°F)	Predicted ΔRT_{NDT} (°F)	Residual ΔRT_{NDT} (°F)	<28°F (Weld)
Weld Metal Heat # 10137 (Calvert Cliffs Unit 2 Data)	263°	63.0	0.825	0.9460	74.9	59.6	15.3	Yes
	97°	63.0	1.95	1.1825	84.1	74.5	9.6	Yes
	104°	63.0	2.44	1.2401	69.9	78.1	8.2	Yes
Weld Metal Heat # 10137 (Millstone Unit 2 Data)	97°	63.0	0.324	0.6902	62.4	43.5	18.9	Yes
	104°	63.0	0.949	0.9853	51.9	62.1	10.2	Yes
	83°	63.0	1.740	1.1523	56.3	72.6	16.3	Yes

The scatter of ΔRT_{NDT} values about the best-fit line, drawn as described in Regulatory Guide 1.99, Revision 2, Position 2.1 [Ref. D-1], should be less than 28°F for weld metal. Table D-4 indicates that 100% (six out of six) of the surveillance data points fall within the +/- 1σ of 28°F scatter band for surveillance weld materials. Therefore, the surveillance weld material (Heat # 10137) is deemed "credible" per the third criterion when all available data is considered.

In conclusion, the combined surveillance data from Calvert Cliffs Unit 2 and Millstone Unit 2 for weld Heat # 10137 may be applied to the Arkansas Nuclear One Unit 2 reactor vessel weld. The Position 2.1 chemistry factor calculation, as applicable to the Arkansas Nuclear One Unit 2 reactor vessel weld, is contained in Section 5. This Position 2.1 CF value could be used with a reduced margin term in the ART calculations contained in Section 7. However, consistent with the discussion in Section D.1 of this Appendix, the ART values calculated with the Position 2.1 CF value for weld Heat # 10137 utilize a full margin term for conservatism.

D.3 REFERENCES

D-1 U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.

D-2 Westinghouse Report WCAP-18166-NP, Revision 0, "Analysis of Capsule 284° from the Entergy Operations, Inc. Arkansas Nuclear One Unit 2 Reactor Vessel Radiation Surveillance Program," September 2016.

D-3 Westinghouse Report WCAP-16918-NP, Revision 1, "Analysis of Capsule V from the Southern Nuclear Operating Company Joseph M. Farley Unit 2 Radiation Surveillance Program," April 2008.

D-4 Westinghouse Report WCAP-16012, Revision 0, "Analysis of Capsule W-83 from the Dominion Nuclear Connecticut Millstone Unit 2 Reactor Vessel Radiation Surveillance Program," February 2003.

D-5 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.

D-6 Westinghouse Report WCAP-17501-NP, Revision 0, "Analysis of Capsule 104° from the Calvert Cliffs Unit No. 2 Reactor Vessel Radiation Surveillance Program," February 2012.

D-7 ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," ASTM, July 1982.

D-8 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]*