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Watts Bar Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation

Westinghouse

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RECORD OF REVISION

Revision 0: Original Issue

Revision 1: This revision was created to correct the initial upper shelf energy values, and to update the end-of-life (EOL) upper shelf energy predictions using the new initial values. These changes are reflected in Table B-1 of Appendix B (Upper Shelf Energy Evaluation). No conclusions changed as a result of using the new initial values to update the EOL upper shelf energy predictions.

Revision 2: This revision was created to modify the executive summary. Paragraphs were added to describe the basis for the use of ASME Code Section XI rather than ASME Code Section III in the pressure-temperature limit curve development.

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EXECUTIVE SUMMARY

This report provides the methodology and results of the generation of heatup and cooldown pressure-temperature (P-T) limit curves for normal operation of the Watts Bar Unit 2 reactor vessel. The heatup and cooldown P-T limit curves were generated using the highest adjusted reference temperature (ART) value pertaining to Watts Bar Unit 2. The highest ART value was that of intermediate shell forging 05 at both 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations. The P-T curves made use of the K_{Ic} methodology detailed in the 1998 through 2000 Addenda Edition of the ASME Code, Section XI, Appendix G, and ASME Code Case N-641.

The applicable ASME Section III Edition and Addenda for the Watts Bar Unit 2 Reactor Pressure Vessel is the 1971 Edition with Addenda through Winter 1971. This Edition and Addenda did not contain specific information or requirements related to development of the Pressure-Temperature (P-T) limits required by 10 CFR 50, Appendix G. Appendix G of Section III was developed later. Since no guidance existed in the Code of Record for the vessel, the Pressure Temperature Limits Report (PTLR) has been developed in accordance with current methodologies contained in Westinghouse Topical Report WCAP-14040-A, Revision 4, which has been previously accepted by the NRC.

The P-T limit curves developed herein utilize the methodology contained in Appendix G of Section XI of the ASME Code. This is the NRC requirement for P-T limit curve development. Particularly, 10 CFR 50, Appendix G (Section IV.A.2.b), requires that P-T limits must be at least as conservative as limits obtained by following the methods of analysis and the margins of safety of Appendix G of Section XI of the ASME Code. Therefore, this NRC requirement is met by the P-T limit curves developed herein.

The methods of Appendix G of ASME Code Section XI are described in the NRC-approved Westinghouse methodology (WCAP-14040-A, Revision 4), which was used to develop the P-T limit curves. The NRC Safety Evaluation (SE), contained in the opening pages of WCAP-14040-A, Revision 4, concludes that the contents of WCAP-14040-A, Revision 4, are acceptable for referencing as PTLR methodology (See Section 4.0.b of the SE for the approval statement). WCAP-14040-A, Revision 4 contains guidance on the use of Code Case N-641, which is approved for use without any exemption request per NRC SE Section 4.0.b. In summary, the approach taken herein for P-T limit curve development is in accordance with the applicable NRC requirements. Further, given that ASME Section III, 1971 Edition with Addenda through Winter 1971 does not provide any criteria, this methodology does not conflict with the applicable construction Code requirements.

The P-T limit curves were generated for 7 EFPY using heatup rates of 60 and 100°F/hr and cooldown rates of 0, 20, 40, 60 and 100°F/hr. The curves were developed without margins for instrumentation errors. These curves can be found in Figures 5-1 and 5-2. Appendix A contains the thermal stress intensity factors for the maximum heatup and cooldown rates.

Also documented in this report are the upper shelf energy (USE), pressurized thermal shock (PTS), and emergency response guideline (ERG) limit evaluations. These evaluations are included in appendices B, C, and D, respectively.

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.

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} (IRT_{NDT}). The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and nickel) present in reactor vessel steels. The Nuclear Regulatory Commission (NRC) has published a method for predicting radiation embrittlement in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials" [Reference 1]. Regulatory Guide 1.99, Revision 2, is used for the calculation of Adjusted Reference Temperature (ART) values (IRT_{NDT} + ΔRT_{NDT} + margins for uncertainties) at the surface, 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 and the NRC approved methodology documented in WCAP-14040-A, Revision 4 [Reference 2], "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves." Specifically, the K_{IC} methodology of the 1998 through 2000 Addenda Edition of ASME Code, Section XI, Appendix G [Reference 3] was used.

The calculated ART values are documented in Tables 4-2 and 4-3 of this report. The design basis fluence projections are based on the values verified by Westinghouse in letter LTR-REA-08-105, Revision 2 [Reference 4].

The purpose of this report is to present the calculations and the development of the Watts Bar Unit 2 heatup and cooldown P-T limit curves for 7 EFPY. This report documents the calculated ART values and the development of the P-T limit curves for normal operation. The P-T curves herein were generated without instrumentation errors. The P-T curves include pressure-temperature limits for the vessel flange region per the requirements of 10 CFR Part 50, Appendix G [Reference 5].

2 FRACTURE TOUGHNESS PROPERTIES

The fracture-toughness properties of the ferritic materials in the Watts Bar Unit 2 reactor vessel are presented in Table 2-1. The unirradiated RT_{NDT} values for the closure head and vessel flange are documented in Table 2-2.

The Regulatory Guide 1.99, Revision 2 methodology used to develop the heatup and cooldown P-T limit curves documented in this report is the same as that documented in WCAP-14040-A, Revision 4 [Reference 2]. The chemistry factors (CFs) were calculated using Regulatory Guide 1.99 Revision 2, Position 1.1. Position 1.1 uses the tables from the Regulatory Guide along with the best estimate copper and nickel weight percents, which are presented in Table 2-1. Table 2-3 summarizes the Position 1.1 CFs determined for the Watts Bar Unit 2 beltline materials.

Table 2-1 Summary of the Best Estimate Cu and Ni Weight Percent and Initial RT_{NDT} Values for the Watts Bar Unit 2 Reactor Vessel Materials

Material Description	Chemical Composition		Initial $RT_{NDT}^{(a)}$
Reactor Vessel Beltline Region Location	Cu wt%	Ni wt%	
Intermediate Shell Forging 05	0.05	0.78	14°F
Lower Shell Forging 04	0.05	0.81	5°F
Intermediate to Lower Shell Circumferential Weld Seam W05	0.05	0.70	-50°F

Note:

(a) The initial RT_{NDT} values are measured values, taken from WCAP-13830, Revision 1 [Reference 6].

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

Material Identification	Initial $RT_{NDT}^{(a)}$
Closure Head Flange	-40°F
Vessel Flange	-22°F

Note:

(a) The initial RT_{NDT} values are measured values, taken from WCAP-13830, Revision 1 [Reference 6].

Table 2-3 Summary of the Watts Bar Unit 2 Reactor Vessel Beltline Material Chemistry Factors

Beltline Materials	Chemistry Factor
Intermediate Shell Forging 05	31°F
Lower Shell Forging 04	31°F
Intermediate to Lower Shell Circumferential Weld Seam W05	68°F

3 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

3.1 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 [Reference 3]. The K_{Ic} curve is given by the following equation:

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

where,

K_{Ic} (ksi√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.

3.2 METHODOLOGY FOR PRESSURE-TEMPERATURE LIMIT CURVE DEVELOPMENT

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

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

where,

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

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

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

C = 2.0 for Level A and Level B service limits

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

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

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

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

$M_m = 1.85$ for $\sqrt{t} < 2$,
$M_m = 0.926\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$,
$M_m = 3.21$ for $\sqrt{t} > 3.464$

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

$M_m = 1.77$ for $\sqrt{t} < 2$,
$M_m = 0.893\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$,
$M_m = 3.09$ for $\sqrt{t} > 3.464$

and

p = internal pressure (ksi), Ri = vessel inner radius (in.), and t = vessel wall thickness (in.).

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

$$K_{Ib} = M_b * \text{Maximum Stress, where } M_b \text{ is two-thirds of } M_m \tag{4}$$

The maximum K_I produced by radial thermal gradient for the postulated inside surface defect of G-2120 is:

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

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

$$K_{It} = 0.753 \times 10^{-3} \text{ x HU x } t^{2.5} \tag{6}$$

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 inside surface defect using the relationship:

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

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

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

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

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

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

Note, that Equations 3, 7, and 8 were implemented in the OPERLIM computer code, which is the program used to generate the pressure-temperature (P-T) limit curves. The P-T curve methodology is the same as that described in WCAP-14040-A, Revision 4 "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" [Reference 2] Section 2.6 (equations 2.6.2-4 and 2.6.3-1).

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.

3.3 CLOSURE HEAD/VESSEL FLANGE REQUIREMENTS

10 CFR Part 50, Appendix G [Reference 5] addresses the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure flange regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (3107 psig for Watts Bar Unit 2), which is calculated to be 621 psig. The limiting unirradiated RT_{NDT} of -22°F occurs in the vessel flange of the Watts Bar Unit 2 reactor vessel, so the minimum allowable temperature of this region is 98°F at pressures greater than 621 psig (without instrument uncertainties). This limit is shown in Figures 5-1 and 5-2 wherever applicable.

4 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 [Reference 7]. If measured values of initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may be used if there are sufficient test results to establish a mean and standard deviation for the class.

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

The Westinghouse Radiation Engineering and Analysis Group evaluated the vessel fluence projections in LTR-REA-08-105, Revision 2 [Reference 4], and the results are presented in Table 4-1. The evaluation methods used in Reference 4 are consistent with the methods presented in WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" [Reference 2]. Table 4-1 also provides a summary of the vessel fluence projections at the 1/4T and 3/4T locations. Tables 4-2 and 4-3 contain the 1/4T and 3/4T calculated fluences and fluence factors, per Regulatory Guide 1.99, Revision 2, used to calculate the 7 EFPY ART values for all beltline materials in the Watts Bar Unit 2 reactor vessel.

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, and 8.5°F 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, and is 14°F (half the value) when credible surveillance capsule data is used. σ_Δ need not exceed 0.5 times the mean value of ΔRT_{NDT}.

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

Table 4-1 Fluence Values for the Watts Bar Unit 2 Reactor Vessel Beltline Materials

Beltline Materials	7 EFPY Fluence (n/cm^2, E > 1.0 MeV)		
	Inner Wetted Surface	1/4T Location (x=2.116 in.)	3/4T Location (x=6.349 in.)
Intermediate Shell Forging 05	6.93E+18	4.17E+18	1.51E+18
Lower Shell Forging 04	6.93E+18	4.17E+18	1.51E+18
Intermediate to Lower Shell Circumferential Weld Seam W05	6.93E+18	4.17E+18	1.51E+18

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

Reactor Vessel Location	CF (°F)	1/4T f (n/cm^2, E > 1.0 MeV)	1/4T FF	ΔRT_{NDT} (°F)	IRT_{NDT}(a) (°F)	σ_i(a) (°F)	σ_Δ (°F)	M (°F)	ART (°F)
Intermediate Shell Forging 05	31	4.17E+18	0.757	23.5	14	0	11.7	23.5	61
Lower Shell Forging 04	31	4.17E+18	0.757	23.5	5	0	11.7	23.5	52
Intermediate to Lower Shell Circumferential Weld Seam W05	68	4.17E+18	0.757	51.5	-50	0	25.7	51.5	53

Note:
(a) The initial RT_{NDT} values are measured values; therefore, σ_i = 0°F.

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

Reactor Vessel Location	CF (°F)	3/4T f (n/cm^2, E > 1.0 MeV)	3/4T FF	ΔRT_{NDT} (°F)	IRT_{NDT}(a) (°F)	σ_i(a) (°F)	σ_Δ (°F)	M (°F)	ART (°F)
Intermediate Shell Forging 05	31	1.51E+18	0.504	15.6	14	0	7.8	15.6	45
Lower Shell Forging 04	31	1.51E+18	0.504	15.6	5	0	7.8	15.6	36
Intermediate to Lower Shell Circumferential Weld Seam W05	68	1.51E+18	0.504	34.3	-50	0	17.1	34.3	19

Note:
(a) The initial RT_{NDT} values are measured values; therefore, σ_i = 0°F.

Contained in Table 4-4 is a summary of the limiting ART values used in the generation of the Watts Bar Unit 2 reactor vessel P-T limit curves. The limiting material for both the 1/4T location and the 3/4T location is Intermediate Shell Forging 05.

Table 4-4 Summary of the Limiting ART Values Used in the Generation of the Watts Bar Unit 2 Heatup/Cooldown Curves

EFPY	Limiting ART (°F)	
	1/4T	3/4T
7	61	45

5 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 beltline region using the methods discussed in Sections 3 and 4 of this report. This approved methodology is also presented in WCAP-14040-A, Revision 4.

Figure 5-1 presents the limiting heatup curves without margins for possible instrumentation errors using heatup rates of 60 and 100°F/hr applicable for 7 EFPY with the “Flange-Notch” requirement and using the “Axial-flaw” methodology. This curve was generated using 1998 through 2000 Addenda ASME Code Section XI, Appendix G. Figure 5-2 presents the limiting cooldown curve without margins for possible instrumentation errors using cooldown rates of 0, 20, 40, 60 and 100°F/hr applicable for 7 EFPY with the “Flange-Notch” requirement. Again, this curve was generated using 1998 through 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 5-1 and 5-2. This is in addition to other criteria, which must be met before the reactor is made critical, as discussed below 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 5-1 (heatup curve only). The straight-line portion of the criticality limit is at the minimum permissible temperature for the 2485 psig inservice hydrostatic test as required by Appendix G to 10 CFR Part 50. The governing equation for the hydrostatic test is defined in 1998 through 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,

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

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 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 4 of this report. For the heatup and cooldown curves without margins for instrumentation errors, the minimum temperatures for the inservice hydrostatic leak tests for the Watts Bar Unit 2 reactor vessel at 7 EFPY is 122°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 5-1 and 5-2 define all of the above limits for ensuring prevention of non-ductile failure for the Watts Bar Unit 2 reactor vessel for 7 EFPY with the "Flange-Notch" requirement, without instrumentation uncertainties. The data points used for developing the heatup and cooldown pressure-temperature limit curves shown in Figures 5-1 and 5-2 are presented in Tables 5-1 and 5-2.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Intermediate Shell Forging 05
LIMITING ART VALUES AT 7 EFPY: 1/4T, 61°F
3/4T, 45°F

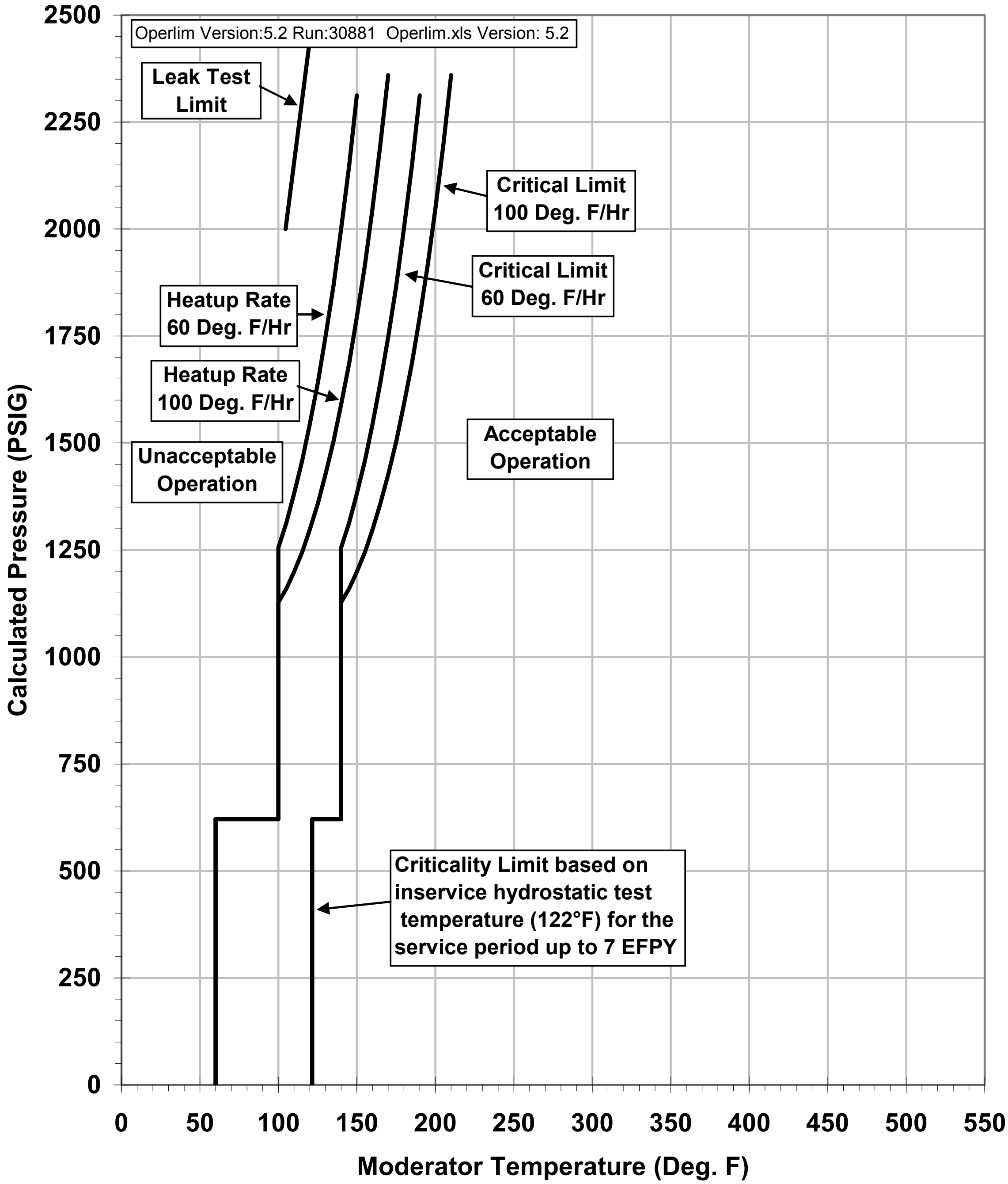

Figure 5-1 Watts Bar Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rates of 60 and 100°F/hr) Applicable for 7 EFPY (without Margins for Instrumentation Errors) Using 1998 through 2000 Addenda App. G Methodology (w/K_{Ic})

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Intermediate Shell Forging 05
LIMITING ART VALUES AT 7 EFPY: 1/4T, 61°F
3/4T, 45°F

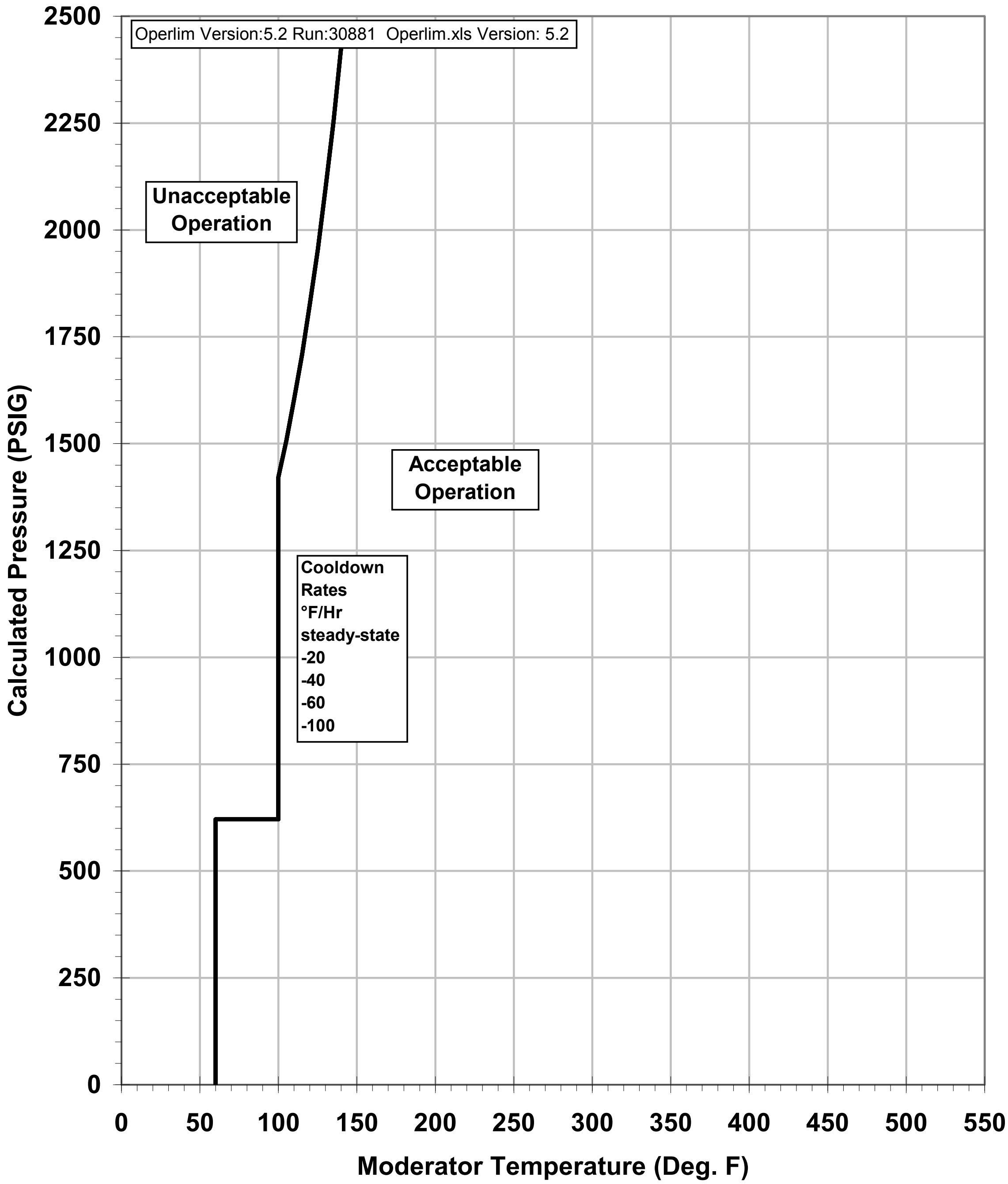

Figure 5-2 Watts Bar Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for 7 EFPY (without Margins for Instrumentation Errors) Using 1998 through 2000 Addenda App. G Methodology (w/K_{Ic})

Table 5-1 7 EFPY Heatup Curve Data Points Using 1998 through 2000 Addenda App. G Methodology (w/K_{IC}, w/Flange Notch and w/o Uncertainties for Instrumentation Errors)

Leak Test Limit		60°F/hr Heatup		60°F/hr Criticality		100°F/hr Heatup		100°F/hr Criticality	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
105	2000	60	0	122	0	60	0	122	0
105	2000	60	621	122	621	60	621	122	621
122	2485	65	621	122	621	65	621	122	621
122	2485	70	621	122	621	70	621	122	621
		75	621	122	621	75	621	122	621
		80	621	125	621	80	621	125	621
		85	621	130	621	85	621	130	621
		90	621	135	621	90	621	135	621
		95	621	140	621	95	621	140	621
		100	621	140	1256	100	621	140	1128
		100	621	145	1314	100	621	145	1160
		100	1256	150	1381	100	1128	150	1199
		105	1314	155	1458	105	1160	155	1245
		110	1381	160	1544	110	1199	160	1298
		115	1458	165	1640	115	1245	165	1358
		120	1544	170	1748	120	1298	170	1426
		125	1640	175	1868	125	1358	175	1503
		130	1748	180	2001	130	1426	180	1590
		135	1868	185	2149	135	1503	185	1687
		140	2001	190	2312	140	1590	190	1795
		145	2149			145	1687	195	1915
		150	2312			150	1795	200	2048
						155	1915	205	2196
						160	2048	210	2360
						165	2196		
						170	2360		

Table 5-2 7 EFPY Cooldown Curve Data Points Using 1998 through 2000 Addenda App. G Methodology (w/K_{IC}, w/Flange Notch and w/o Uncertainties for Instrumentation Errors)

Steady State		20°F/hr.		40°F/hr.		60°F/hr.		100°F/hr.	
T(°F)	P (psig)	T(°F)	P (psig)	T(°F)	P (psig)	T(°F)	P (psig)	T(°F)	P (psig)
60	0	60	0	60	0	60	0	60	0
60	621	60	621	60	621	60	621	60	621
65	621	65	621	65	621	65	621	65	621
70	621	70	621	70	621	70	621	70	621
75	621	75	621	75	621	75	621	75	621
80	621	80	621	80	621	80	621	80	621
85	621	85	621	85	621	85	621	85	621
90	621	90	621	90	621	90	621	90	621
95	621	95	621	95	621	95	621	95	621
100	621	100	621	100	621	100	621	100	621
100	1422	100	1422	100	1422	100	1422	100	1422
105	1508	105	1508	105	1508	105	1508	105	1508
110	1603	110	1603	110	1603	110	1603	110	1603
115	1709	115	1709	115	1709	115	1709	115	1709
120	1825	120	1825	120	1825	120	1825	120	1825
125	1954	125	1954	125	1954	125	1954	125	1954
130	2096	130	2096	130	2096	130	2096	130	2096
135	2253	135	2253	135	2253	135	2253	135	2253
140	2427	140	2427	140	2427	140	2427	140	2427

6 REFERENCES

1. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," U. S. Nuclear Regulatory Commission, May 1988.
2. WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," J. D. Andrachek, et al., May 2004.
3. Appendix G to the 1998 through the 2000 Addenda Edition of the ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Division 1, "Fracture Toughness Criteria for Protection Against Failure."
4. Westinghouse Letter LTR-REA-08-105, Revision 2, "Pressure Vessel Design Basis Fluence for Watts Bar Unit 2," M. A. Hunter, dated March 18, 2009.
5. Code of Federal Regulations, 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Washington, D. C., Federal Register, Volume 60, No. 243, dated December 19, 1995.
6. WCAP-13830, Revision 1, "Heatup and Cooldown Limit Curves for Normal Operation for Watts Bar Unit 2," J. M. Chicots, et al., February 1995.
7. ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Division 1, Subsection NB, Section NB-2300, "Fracture Toughness Requirements for Material."

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

The following pages contain the thermal stress intensity factors (K_{It}) for the maximum heatup and cooldown rates. The vessel radii to the 1/4T and 3/4T locations are as follows:

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

Table A-1 K_{It} Values for 100°F/hr Heatup Curve (w/o Margins for Instrument Errors)

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

Table A-2 K_{It} Values for 100°F/hr Cooldown Curve (w/o Margins for Instrument Errors)

Water Temp. (°F)	Vessel Temperature @ 1/4T Location for 100°F/hr Cooldown (°F)	100°F/hr Cooldown 1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)
210	236.000	16.31
205	230.917	16.24
200	225.833	16.18
195	220.750	16.11
190	215.666	16.05
185	210.582	15.98
180	205.497	15.91
175	200.413	15.85
170	195.329	15.78
165	190.244	15.71
160	185.160	15.65
155	180.075	15.58
150	174.990	15.51
145	169.906	15.45
140	164.821	15.38
135	159.737	15.32
130	154.653	15.25
125	149.568	15.18
120	144.484	15.12
115	139.400	15.05
110	134.316	14.99
105	129.232	14.92
100	124.148	14.86
95	119.064	14.79
90	113.981	14.72
85	108.897	14.66
80	103.814	14.59
75	98.731	14.53
70	93.647	14.47
65	88.565	14.40
60	83.483	14.34

APPENDIX B UPPER SHELF ENERGY EVALUATION

Per Regulatory Guide 1.99, Revision 2 [Reference B-1], the Charpy upper shelf energy (USE) is assumed to decrease as a function of fluence and copper content as indicated in Figure 2 of the Guide (Figure B-1 of this report) when surveillance data is not used. Linear interpolation is permitted.

The 32 EFPY end-of-life (EOL) USE of the vessel materials can be predicted using the corresponding 1/4T fluence projection, the copper content of the beltline materials, and Figure 2 in Regulatory Guide 1.99, Revision 2. The maximum vessel clad/base metal interface fluence value was used to determine the corresponding 1/4T fluence value at 32 EFPY.

The Watts Bar Unit 2 reactor vessel beltline region thickness is 8.465 inches. Per LTR-REA-08-105, Revision 2 [Reference B-2], the maximum vessel clad/base metal interface fluence value is 3.17E+19 n/cm^2 (E > 1.0 MeV). Calculation of the 1/4T vessel surface fluence values at 32 EFPY for the beltline materials is shown as follows:

Maximum Vessel Fluence @ 32 EFPY = 3.17E+19 n/cm^2 (E > 1.0 MeV)

1/4T Fluence @ 32 EFPY = (3.17E+19 n/cm^2) * $e^{(-0.24 * (8.465 / 4))}$
= 1.91E+19 n/cm^2 (E > 1.0 MeV)

Table B-1 provides the predicted EOL USE values for 32 EFPY.

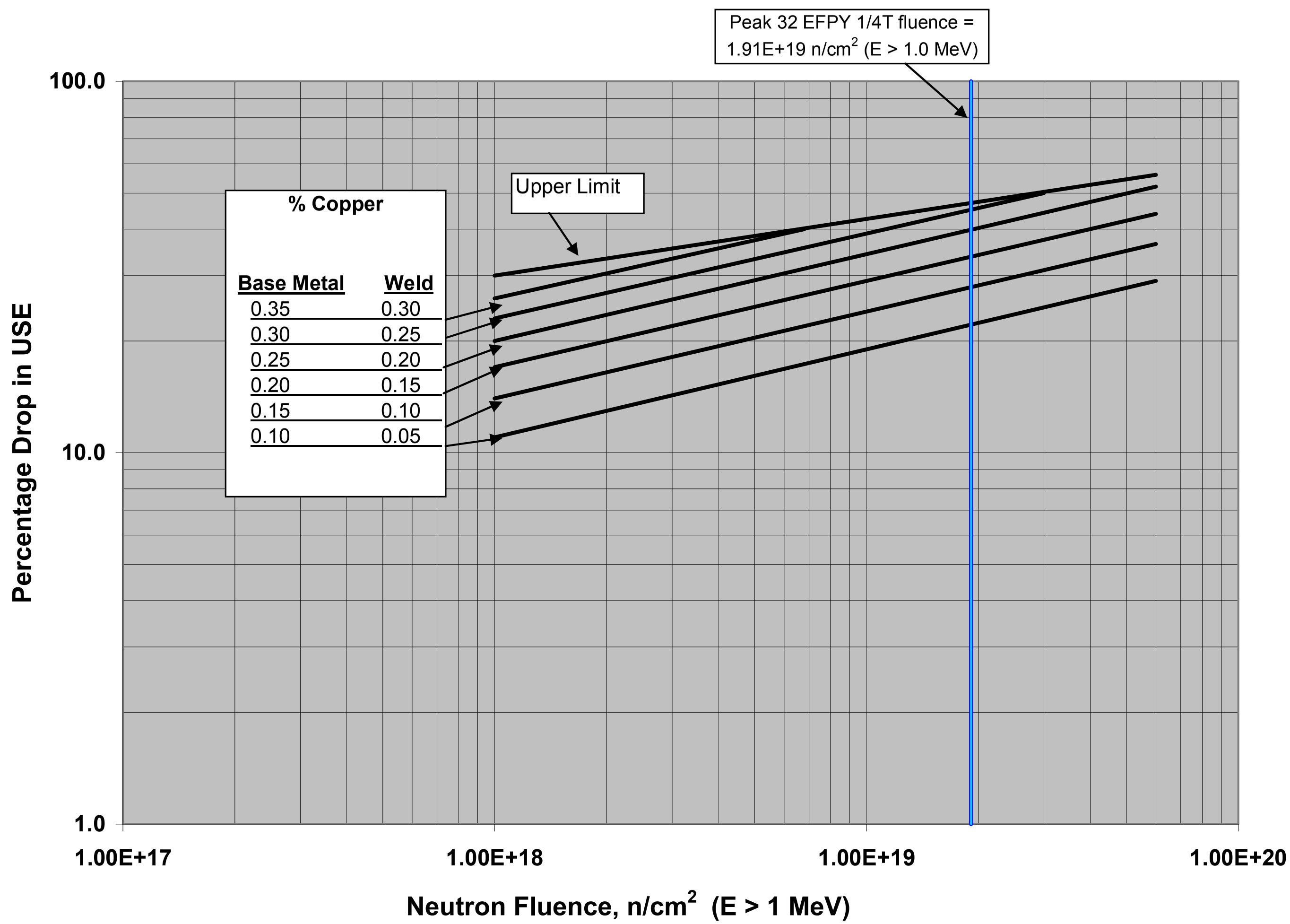

Figure B-1 Regulatory Guide 1.99, Revision 2 Predicted Decrease in Upper Shelf Energy as a Function of Copper and Fluence

Table B-1 Predicted Position 1.2 Upper Shelf Energy Values at 32 EFPY

Material	Weight % Cu	1/4T EOL Fluence (n/cm^2, E > 1.0 MeV)	Unirradiated USE[(a)] (ft-lb)	Projected USE Decrease[(d)]	Projected EOL USE (ft-lb)
Intermediate Shell Forging 05	0.05	1.91E+19	(138) 90[(b)]	23[(e)]	69
Lower Shell Forging 04	0.05	1.91E+19	(162) 105[(b)]	23[(e)]	81
Intermediate to Lower Shell Circumferential Weld Seam W05	0.05	1.91E+19	127[(c)]	23	98

Notes:

(a) Information source is CMTR-RV-WBT [Reference B-3]. Reference B-3 reports energy values in units of kgm/cm^2. Per WCAP-9455, Revision 3 [Reference B-4], specimen cross sections are 0.394 in x 0.315 in. The conversion factors used are as follows:

$1\ kg/cm^2 = 14.223\ lb/in^2$

1 m = 3.280833 ft

(b) According to Reference B-3, the specimens were tested in the strong (tangential) direction. The strong direction initial USE values are listed in parentheses. However, in accordance with the recommendations of NUREG-0800, Revision 1 [Reference B-5], the strong direction values were reduced to 65% in order to approximate the weak (axial) direction values. These values are listed outside the parentheses, and are used in the EOL USE projections.

(c) The circumferential weld testing is considered non-directional. Therefore, no percent reduction was performed to determine the initial USE value to be used in the EOL USE projections.

(d) Projected USE decreases were calculated in accordance with Regulatory Guide 1.99, Revision 2, Position 1.2.

(e) These projected USE decreases were conservatively taken from the base metal 0.10% copper line in Figure 2 of Regulatory Guide 1.99, Revision 2.

USE Conclusion

All of the beltline materials in the Watts Bar Unit 2 reactor vessel are projected to remain above the USE screening criterion value of 50 ft-lb (per 10 CFR 50, Appendix G [Reference B-6]) at 32 EFPY.

B.1 REFERENCES

B-1 Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.

B-2 Westinghouse Letter LTR-REA-08-105, Revision 2, “Pressure Vessel Design Basis Fluence for Watts Bar Unit 2,” M. A. Hunter, dated March 18, 2009.

B-3 CMTR-RV-WBT, Revision 0, “WBT Reactor Vessel Certified Material Test Reports.”

B-4 WCAP-9455, Revision 3, “Tennessee Valley Authority Watts Bar Unit No. 2 Reactor Vessel Radiation Surveillance Program,” B. A. Rosier and B. N. Burgos, September 2009.

B-5 NUREG-0800, Revision 1, Section 5.3.2, Branch Technical Position MTEB 5-2, “Fracture Toughness Requirements,” U. S. Nuclear Regulatory Commission, Washington, D. C., July 1981.

B-6 Code of Federal Regulations, 10 CFR 50, Appendix G, “Fracture Toughness Requirements,” U.S. Nuclear Regulatory Commission, Washington D.C., Federal Register, Volume 60, No. 243, dated December 19, 1995.

APPENDIX C PRESSURIZED THERMAL SHOCK EVALUATION

The PTS Rule, 10 CFR 50.61 [Reference C-1], requires that for each pressurized water nuclear power reactor for which an operating license has been issued, the licensee shall have projected values of RT_{PTS} accepted by the NRC for each reactor vessel beltline material at the end-of-life (EOL) fluence of the plant. This assessment must specify the basis for the projected value of RT_{PTS} for each vessel beltline material, including the assumptions regarding core-loading patterns, and must specify the copper and nickel contents and the fluence value used in the calculation. This assessment must be updated whenever there is a significant change in projected values of RT_{PTS}, or upon request for a change in the expiration date for operation of the facility. Changes to RT_{PTS} values are considered significant if either the previous value or the current value, or both values, exceed the screening criterion prior to the expiration of the operating license, including any renewed term, if applicable, for the plant.

Table C-1 contains the RT_{PTS} calculations for each of the beltline region reactor vessel materials for Watts Bar Unit 2 at 32 EFPY.

Table C-1 RT_{PTS} Calculations for the Watts Bar Unit 2 Beltline Materials at 32 EFPY

Material	CF (°F)	32 EFPY Fluence (n/cm^2, E > 1.0 MeV)	FF(a)	IRT_{NDT} (°F)	ΔRT_{NDT}(b) (°F)	σ_U(c) (°F)	σ_Δ(d) (°F)	M(e) (°F)	RT_{PTS}(f) (°F)
Intermediate Shell Forging 05	31	3.17E+19	1.30	14	40.4	0	17	34	88
Lower Shell Forging 04	31	3.17E+19	1.30	5	40.4	0	17	34	79
Intermediate to Lower Shell Circumferential Weld Seam W05	68	3.17E+19	1.30	-50	88.7	0	28	56	95

Notes:

(a) FF = fluence factor = $f^{(0.28 - 0.1 \log(f))}$.

(b) $\Delta RT_{NDT} = \Delta RT_{PTS} = CF * FF$.

(c) As indicated in Table 2-1 of this report, the IRT_{NDT} values are measured; hence, according to 10 CFR 50.61, $\sigma_U = 0$°F.

(d) Per the guidance of 10 CFR 50.61, the base metal $\sigma_\Delta = 17$°F and the weld metal $\sigma_\Delta = 28$°F when surveillance data is not utilized. However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$.

(e) $M = \text{Margin} = 2 *(\sigma_U^2 + \sigma_\Delta^2)^{1/2}$.

(f) $RT_{PTS} = IRT_{NDT} + \Delta RT_{PTS} + \text{Margin}$.

PTS Conclusions for RT_{PTS} Values at EOL (32 EFPY)

All of the beltline materials in the Watts Bar Unit 2 reactor vessel are below the RT_{PTS} screening criteria values of 270°F, for axially oriented welds and plates / forgings, and 300°F, for circumferentially oriented welds, (Per 10 CFR 50.61) at 32 EFPY.

C.1 REFERENCES

C-1 Code of Federal Regulations, 10 CFR Part 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.

APPENDIX D EMERGENCY RESPONSE GUIDELINE LIMITS EVALUATION

The Emergency Response Guideline (ERG) limits [Reference D-1] were developed in order to establish guidance for operator action in the event of an emergency situation, such as a PTS event. Generic categories of limits were developed for the guidelines based on the limiting inside surface RT_{NDT}. These generic categories were conservatively generated for the Westinghouse Owners Group (WOG) to be applicable to all Westinghouse plants.

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

The ERG category is determined by the magnitude of the limiting RT_{NDT} value, which is calculated the same way as the RT_{PTS} values were calculated in Appendix C of this report. The material with the highest RT_{NDT} defines the limiting material, which for Watts Bar Unit 2 is the Intermediate to Lower Shell Circumferential Weld Seam W05 (see Table C-1). Table D-1 identifies ERG category limits and the limiting material RT_{NDT} value at 32 EFPY.

Table D-1 Evaluation of Watts Bar Unit 2 ERG Limit Category

ERG Pressure-Temperature Limits	
Applicable RT_{NDT} Value[(a)]	**ERG P-T Limit Category**
RT_{NDT} < 200°F	Category I
200°F < RT_{NDT} < 250°F	Category II
250°F < RT_{NDT} < 300°F	Category III b
Limiting RT_{NDT} Values at 32 EFPY	
Material	**RT_{NDT} Value**
Intermediate to Lower Shell Circumferential Weld Seam W05	95°F

Note:

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

Emergency Response Guideline Limits Conclusion

Reviewing the ERG limit categories with the limiting RT_{NDT} value provided in Table D-1 would place Watts Bar Unit 2 in Category I through 32 EFPY.

D.1 REFERENCES

D-1 Background Information for Westinghouse Owner's Group Emergency Response Guidelines, Critical Safety Function Status Tree, F-0.4 Integrity, HP/LP-Revision 2, April 30, 2005.