WATTS BAR UNIT 1

LOW UPPER SHELF ENERGY EVALUATION

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Prepared by: J. M. Chicots J. M. Chicots

Reviewed by: M. J. Malone M. J. Malone

Westinghouse Electric Corporation
P. O. Box 355
Pittsburgh, PA 15230

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J. M. Chicots

M. J. Malone

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I. Introduction

Appendix G, "Fracture Toughness Requirements" to 10 CFR Part 50[1] requires that materials of the reactor vessel beltline region exhibit no less than 50 ft-lbs Charpy upper shelf energy during its license life, "unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of upper shelf energy will provide margins of safety against fracture equivalent to those required by the Appendix G of the ASME Code".

Westinghouse Owner's Group (WOG) recently performed an analysis to demonstrate that all participating WOG plant reactor vessels have a margin of safety equivalent to that required by Appendix G of the ASME Code. This was accomplished by performing generic bounding evaluations as per the proposed ASME Section XI, Appendix X criteria and requirements. The Tennessee Valley Authority's Watts Bar Unit 1 reactor vessel was included in this bounding analysis.

The bounding analysis utilized unirradiated J-R curve data (material resistance data) for WOG reactor vessel upper, intermediate and lower shell course materials. These curves were adjusted to reflect expected end-of-license values. As reactor vessels for different size plants have different geometries, representative geometry parameters were chosen for 2, 3 and 4-loop plants. $J_{applied}$ values were then calculated using Linear-Elastic Fracture Mechanics (LEFM) techniques for Level A and B conditions and Elastic-Plastic Fracture Mechanics (EPFM) techniques for Level C and D conditions.

Based upon the WOG generic bounding analysis, the Tennessee Valley Authority's Watts Bar Unit 1 reactor vessel met the ASME Section XI, Appendix X criteria.

II. Applicability of WOG Reactor Vessel Upper Shelf Energy Bounding Evaluation to Watts Bar Unit 1

A. Materials

As discussed in the introduction, 10 CFR Part 50 Appendix G requires minimum upper-shelf energy, as determined from Charpy V-notch tests. For the unirradiated condition, preservice, the minimum upper-shelf energy as determined from Charpy V-notch tests specimens in accordance with paragraph NB-2322.2 of the ASME Code is 75 ft-lbs unless it is demonstrated to the NRC by appropriate data and analyses that lower values of upper-shelf fracture energy will provide margins of safety against fracture equivalent to those required by Appendix G, ASME Code. The minimum value of upper-shelf energy of 75 ft-lbs was added to 10 CFR Part 50 Appendix G on July 17, 1973. The Westinghouse Electric Corporation Equipment Specification (E-Spec) Revision 4

imposed a minimum Charpy V-notch upper-shelf energy requirement for beltline region materials of 75 ft-lbs for all cases in May 1972, without distinction as to the predicted amount of irradiation damage. The Watts Bar Unit 1 reactor vessel was constructed to Revision 2 of the Westinghouse Electric Corporation E-Spec and therefore was constructed prior to the 75 ft-lb minimum requirement. The initial USE values for the Watts Bar Unit 1 reactor vessel are given in Table 1.

10 CFR Part 50 Appendix G required either implicitly or explicitly that during service life the upper-shelf energy of all reactor vessel materials must not be less than 50 ft-lbs. The initial or unirradiated upper-shelf energy is generally dependent upon the inclusion content, cleanness of the material and/or the directionality of forming the material. The decrease in upper-shelf energy during service life is associated with radiation. Residual copper has been identified as the most important chemical element in promoting the decrease in the upper-shelf energy during service life. The Westinghouse Electric Corporation Equipment Specification, Revision 3, dated July 1971, limited the copper content to 0.12 weight percent for base material (plates/forging) and to 0.10 for weldments. Prior to July 1971, Westinghouse Equipment Specifications did not specify a maximum copper content in the procurement of the reactor pressure vessel. Therefore, for reactor pressure vessels fabricated to Westinghouse Equipment Specifications Revisions 0, 1 and 2, the possibility exists that the upper shelf energy could be less than 50 ft-lbs during service life, such is the case for the Watts Bar Unit 1 intermediate shell forging. Revisions 3 and 4 to the Equipment Specification were meant to ensure compliance with 10CFR Part 50. Again, The Watts Bar Unit 1 reactor vessel was fabricated with Equipment Specification Revision 2.

As stated above, 10 CFR Part 50 requires that the reactor pressure vessel "beltline" materials upper-shelf energy be no less than above 50 ft-lbs during service life. The "beltline" is defined as the irradiated region of the reactor vessel that directly surrounds the effective height of the active core and adjacent regions that are predicted to experience sufficient neutron damage to warrant consideration in the selection of surveillance material. Therefore, the upper shell course, intermediate shell course, lower shell course and all associated weldments of the reactor pressure vessel are considered to be in the beltline and were assessed as part of the WOG bounding analysis report. The decrease in upper-shelf energy during the service life of the reactor pressure vessel was determined using the methodology given in Regulatory Guide 1.99, Revision 2. The Regulatory Guide identifies two methods for predicting the decrease in upper-shelf energy; by plotting the reduced plant surveillance data on Figure 2 of the guide and fitting the data with a line drawn parallel to the existing lines as the upper bound of all data; or when surveillance data are not available

assume that the upper-shelf energy decreases as a function of fluence and copper content as indicated in Figure 2 of the guide. Both methodologies were used to calculate the decrease in upper-shelf energy for the "bounding analysis".

The unirradiated upper shelf energy values were calculated for the upper shell course, intermediate shell course, lower shell course, and surveillance welds using the guidelines specified in ASTM E-185[3] and Branch Technical Position MTEB 5-2 of the Standard Review Plan[4] which meet the requirements in 10 CFR Part 50, Appendices G and H. The unirradiated charpy V-notch data used to determine the initial upper shelf energy values was obtained from material certification and surveillance capsule program reports. This information is presented in Table 1 for Watts Bar Unit 1 reactor vessel.

The end-of-license (EOL) upper shelf energy values were calculated for the beltline region materials using the Regulatory Guide 1.99, Rev. 2[5] methodology. The EOL upper shelf energy for Watts Bar Unit 1 was determined by calculating the decrease in upper shelf energy as a function of copper content and EOL fluence using Figure 2 of Reference 5. For the upper shell course, intermediate shell course, lower shell course, and surveillance welds, a 1/4-T EOL fluence value was used as defined in 10 CFR Part 50, Appendix G.

B. Mechanical Properties

As the analysis is intended to bound all participating plants, the minimum mechanical properties at an operational temperature of 600°F are used. These values are assumed to be conservative as they represent the minimum yield strength, ultimate strength and Youngs Modulus allowed by the ASME Code[6] for vessel materials. They are taken directly from Reference 6:

- Yield Strength (σ_Y) = 37.8 ksi for SA-302, Grade A and 43.8 ksi for all other materials
- Ultimate Strength (σ_U) = 75 ksi for SA-302, Grade A and 80 ksi for all other materials
- Youngs Modulus (E) = 26.4 Mpsi

C. Stress-Strain Curve

A representative stress-strain curve was developed for use in this analysis. This curve was generated using typical stress-strain data for carbon steel as found in Reference 7. This curve was adjusted to have the code minimum values as listed herein. The linear-elastic portion was developed to have a slope equal to the code minimum Youngs Modulus. The plastic portion strain values were reduced so as to have the yield stress point equal to the code minimum value. This representative curve is shown in Figure 1.

D. Criteria Synopsis

ASME Section XI, Appendix X contains acceptance criteria for three levels of service load conditions:

- Level A and B conditions corresponding to Normal and Upset operational conditions.
- Level C conditions corresponding to Emergency operational conditions
- Level D conditions corresponding to Faulted operational conditions

The Appendix X criteria for these service load conditions are given in the following sections.

1. Level A and B Conditions

For a postulated semi-elliptical surface flaw with flaw depth to wall thickness ratio a/t = 0.25 and with an aspect ratio, surface length to flaw depth of 6 to 1, and oriented along the weld of concern, two criteria must be satisfied as described below. If the base metal is governing, the postulated flaw must be axially oriented. Smaller flaw sizes may be used on an individual case basis if a smaller size of the above postulated flaw can be justified. The expected accumulation pressure to be discussed below is the maximum pressure which satisfies the requirement of ASME Section III, NB-7311(b). The two criteria are:

1. The crack driving force must be shown to be less than the material toughness as given below:

$$J_{applied} < J_{0.1}$$

where $J_{applied}$ is the J-integral value calculated for the postulated flaw under pressure and thermal loading where the assumed pressure is 1.15 times expected accumulation pressure, and with thermal loading using the plant specified heatup and cooldown conditions. The parameter $J_{0.1}$ is the J-integral characteristic of the material resistance to ductile tearing ($J_{material}$), as usually denoted by a J-R curve, at a crack extension of 0.1 inch.

2. The flaw must be stable under ductile crack growth as given below:

$$\frac{dJ_{applied}}{da} < \frac{dJ_{material}}{da}$$

at $J_{applied} = J_{material}$

where $J_{applied}$ is calculated for the postulated flaw under

pressure and thermal loading for all service Level A and B conditions where the assumed pressure is 1.25 times expected accumulation pressure, with thermal loading as is defined in Section 3.0.

The J-integral resistance versus crack growth curve used should reflect a conservative bound representative of the vessel material under evaluation.

2. Level C Conditions

When the upper shelf Charpy energy of any weld material is less than 50 ft-lb, postulate interior semi-elliptic surface flaws with their major axis oriented along the weld of concern and the flaw plane oriented in the radial direction. Postulate both interior axial and circumferential flaws and use the toughness properties for the corresponding orientation. Consider postulated surface flaws with depths up to one tenth the base metal wall thickness, plus the clad, but with total depth not to exceed 1.0 inch and with aspect ratios of 6 to 1 surface length to flaw depth. Smaller flaw sizes may be used on an individual case basis if a smaller size can be justified. For these evaluations, two criteria must be satisfied, as described below:

1. The crack driving force must be shown to be less than the material toughness as given below:

$$J_{applied} < J_{0.1}$$

where $J_{applied}$ is the J-integral value calculated for the postulated flaw in the beltline region of the reactor vessel under the governing level C condition. $J_{0.1}$ is the J-integral characteristic of the material resistance to ductile tearing ($J_{material}$), as usually denoted by a J-R curve test, at a crack extension of 0.1 inch.

2. The flaw must also be stable under ductile crack growth as given below:

$$\frac{dJ_{applied}}{da} < \frac{dJ_{material}}{da}$$

at $J_{applied} = J_{material}$

where $J_{applied}$ is calculated for the postulated flaw under the governing level C condition. The J-integral resistance versus crack growth curve shall be a conservative representation of the vessel material under evaluation.

3. Level D Conditions

When the upper shelf Charpy energy of any weld material is less than 50 ft-lb, postulate interior semi-elliptic surface flaws with their major axis oriented along the weld of concern and the flaw plane oriented in the radial direction with aspect ratio of 6 to 1. Postulate both interior axial and circumferential flaws and use the toughness properties for the corresponding orientation. Consider postulated surface flaws with depths up to one tenth the base metal wall thickness, plus the clad, but with total depth not to exceed 1.0 inch and with aspect ratios of 6 to 1 surface length to depth. Smaller flaw sizes may be used on an individual case basis if a smaller size can be justified. For these evaluations, the following criterion must be met.

The postulated flaw must be stable under ductile crack growth as given below:

$$\frac{dJ_{applied}}{da} < \frac{dJ_{material}}{da}$$

$$\text{at } J_{applied} = J_{material}$$

where $J_{applied}$ is calculated for the postulated flaw under the governing level D condition. The material property to be used for this assessment is the best estimate J-R curve.

4. Safety Margins

Margins of safety have been incorporated in the analysis in a number of ways. First, a flaw having depth of 1/4 the wall thickness (1/4 t) is postulated to exist. Second, conservatism is introduced to level A and B transients by incorporating a safety factor on pressure. Finally, the probability of occurrence of level C and D condition transients is relatively low, so the assumption that this type of transient occurs represents a margin of safety.

E. J-R Curve Representative Values

The objective of this section is to determine limiting values for $J_{material}$ at 0.1" crack extension and $(dJ/da)_{material}$ for each reactor vessel base material (no weld metal fell below 50 ft-lbs.) and each size plant (2, 3 or 4 loop). These two material parameters furnish the limiting values for $J_{applied}$ to meet the ASME Section XI, Appendix X requirements.

Correlations for J-R curve values with temperature, USE values and crack extension are contained in Reference 8. The model relations are as follows:

$$\ln J = \ln C1 + C2 \ln(\Delta a) + C3(\Delta a)^{C4}$$

where:

$\ln C1 = a_1 + a_2 \ln(CVN) + a_3 T$
CVN = Charpy energy (ft-lbs)
T = Temperature (°F)
$C2 = d_1 + d_2 \ln C1$
$C3 = d_4 + d_5 \ln C1$
$C4 < 0$

The values for a_1, a_2, a_3, d_1, d_2, d_4, d_5 and C4 are taken from Appendix B of Reference 8. These correlations were developed using an extensive amount of test data and advanced pattern recognition tools. These correlations are material independent; however, as indicated in Reference 8, different correlations are utilized for base material and weld fluxes.

Based on this model, J-R values were calculated for each size plant. The values obtained were considered to be limiting as the lowest end-of-license (EOL) USE for each size plant was utilized. These J-R values were then adjusted to reflect a 2σ (standard deviation) margin for conservatism. The lowest projected end-of-license USE value for the participating 4-loop plants was from the Watts Bar Unit 1 intermediate shell forging. The temperature value used in the correlation was 390.5°F. This value represents the greatest temperature at the crack tip for a 1/4t flaw. As will be discussed in the following section, Level A and B conditions are limiting. The 1/4t flaw temperature value is assumed for this flaw size in the criteria for level A and B conditions. The temperature value of 390.5° is based on the Level A and B cooldown transient. Both $J_{material}$ and $(dJ/da)_{material}$ may be calculated using this methodology.

Actual J-R data are available for SA-302 grade B material having an initial USE of 50 ft-lbs. These J-R data are given in Reference 9 and are considered to be a lower bound for all J-R data. Data given in Reference 10 also shows the effect of manganese sulfide inclusions in the steel, however the J-R curves of Reference 10 exhibit a much higher resistance to ductile tearing than the values given in Reference 9. In order to perform the most conservative bounding analysis the J-R values were recalculated using the information from Reference 9. Based on Reference 9 the lowest value for $J_{material}$ at 0.1" crack extension for SA-302B material with USE of 50 ft-lbs is 694 in-lb/in^2. This $J_{material}$ value may be adjusted for EOL using the Reference 8 correlation. Based on this model, the percentage decrease in $J_{material}$ for 0.1" crack extension per unit decrease in USE can be calculated. It was determined that the percent decrease in $J_{material}$ per unit decrease in USE never exceeded 3%. The lowest EOL USE for all plants considered in this bounding analysis is 42 ft-lbs for SA-302B base metal material (3 loop

plant). Reducing 694 in-lbs/in^2 by 24% (3% drop per ft-lb x 8 ft-lbs) yields a $J_{material}$ value of 527 in-lbs/in^2. Similarly, another point on the J-R curve given in Reference 9 can be utilized to calculate $(dJ/da)_{material}$. The limiting J-R curve values for the 2, 3 and 4 loop plants are summarized in Table 2.

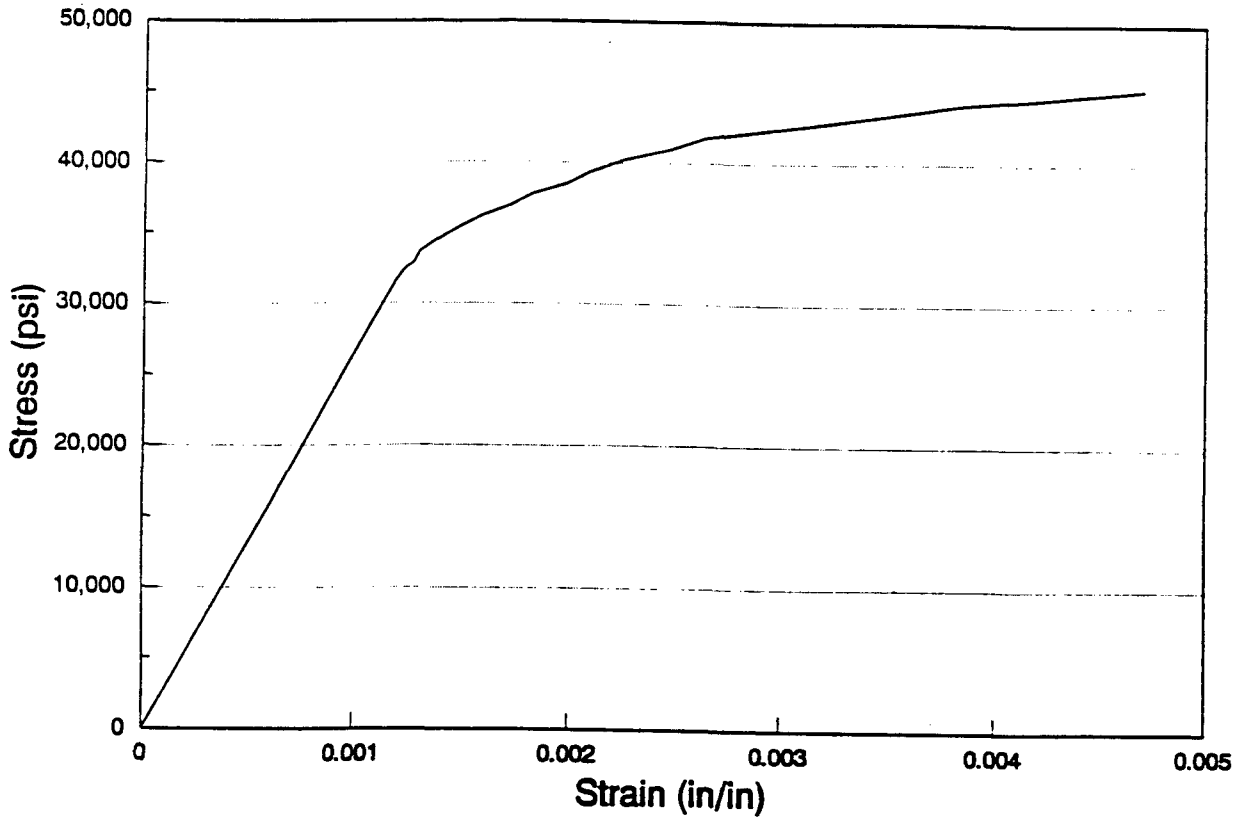
TABLE 1
Watts Bar Unit 1 Beltline Region Material Properties

Description of Beltline Region Material	Base Material Specification	Initial USE (ft-lbs)	EOL USE (ft-lbs)
Inter. Shell Forging	SA-508, Cl 2	62	44
Lower Shell Forging	SA-508, Cl 2	111	87
Vessel Ring Forging	SA-508, Cl 2	96	85
Surveillance Weld	Weld Flux, Grau Lo (LW320)	134	105

TABLE 2
Limiting Values of J-R Data for Reactor Vessel Base Metal Materials

Case	J @ Δ a = 0.1	$dJ_{material}/da$
2 Loop Plants	702	2925
3 Loop Plants	585 (527)*	2140 (599)*
4 Loop Plants	614	2330
Notes: * Calculations based on J-R curves in Reference 9		

FIGURE 1
Representative Stress-Strain Curve



F. Analysis

This section describes the analyses performed to determine the applied J-integral value required for an ASME Section XI, Appendix X evaluation. The inputs needed to perform this analysis are: Material stress-strain curve, mechanical properties, geometry and appropriate transients. The stress-strain curve and mechanical properties are described in the preceding sections. The geometry and transient inputs along with the analyses are described below.

1. Geometry

The two geometry parameters required as inputs to the analysis are reactor vessel inner radius and thickness. Representative values of these parameters were chosen for 2, 3 and 4 loop plants. The geometry values are given in Table 3.

Table 3
Geometry of the Cases Analyzed

Case	Thickness (in.)	Inner Radius (in.)
2 loop	6.500	66.00
3 loop	7.875	78.50
4 loop	8.500	86.50
Watts Bar Unit 1	8.465	86.50

2. Analysis for Level A and B Conditions

The stresses due to Level A and B conditions are significantly lower than the material yield strength. Consequently $J_{applied}$ can be calculated using linear-elastic fracture mechanics (LEFM) techniques with a plastic-zone correction. Guidelines for performing Level A and B analyses are contained in Reference 11. This contains a procedure for calculating $J_{applied}$ which was developed by the ASME Code Committee that generated the ASME Section XI, Appendix X requirements. This approach was utilized to determine the applied J values for each of the cases listed in Table 3.

The procedure was developed specifically for this application. Consequently, it is applicable for a semi-elliptical flaw with an aspect ratio of 6 to 1. Axial and circumferential flaws may be calculated per this procedure. The methodology is as follows:

First, the stress intensity factors (K_I) due to pressure and thermal loadings are calculated. The stress intensity factor is a measure of the driving force of crack extension. K_I is a function of the size of the crack, the applied stress, and the geometry of the structure.

For an axial flaw with a length to depth aspect ratio of 6 to 1, the stress intensity factor, K_{Ip}, due to internal pressure is given by [11]:

$$K_{Ip} = (SF)\ p\ [1 + (R_i/t)]\ (\pi a)^{0.5}\ F_1$$

$$F_1 = 0.982 + 1.006\ (a/t)^2$$

where, "a" is the flaw depth, R_i is the inner radius of the vessel, t is the wall thickness, p is the internal pressure, and

(SF) is the safety factor on pressure. This equation for K_{Ip} is valid for $0.20 \leq a/t \leq 0.50$, and includes the effect of pressure acting on the flaw faces.

For a circumferential flaw with a length to depth aspect ratio of 6 to 1, the stress intensity factor due to internal pressure is given by:

$$K_{Ip} = (SF)\ p\ [1 + (R_i/(2t))]\ (\pi a)^{0.5}\ F_2$$

$$F_2 = 0.885 + 0.233\ (a/t) + 0.345\ (a/t)^2$$

This equation for K_{Ip} is valid for $0.20 \leq a/t \leq 0.50$, and includes the effect of pressure acting on the flaw faces.

For both axial and circumferential flaws with aspect ratio of 6 to 1, the stress intensity factor due to radial thermal gradients is given by:

$$K_{It} = ((CR)/1000)\ t^{2.5}\ F_3$$

$$\text{Case 1: } F_3 = 0.584 + 2.647\ (a/t) - 6.294\ (a/t)^2 + 2.990\ (a/t)^3$$

$$\text{Case 2: } F_3 = 0.690 + 3.127\ (a/t) - 7.435\ (a/t)^2 + 3.532\ (a/t)^3$$

where, t is in inches, K_{It} is in ksi$\sqrt{\text{in.}}$ and, (CR) is the cooldown rate in °F/hour. This equation for K_{It} is valid for $0.20 \leq a/t \leq 0.50$ and $0 \leq (CR) \leq 100$°F/hour. The through-wall temperature distribution used to develop this equation is the same through-wall distribution assumed in Appendix G of Section III and in Section XI. The thermal stress distribution includes the temperature dependence of material properties.

The Case 1 equation is the original approved approximation from ASME Section XI, which is used in calculating stress intensity factors. The Case 2 equation is currently being reviewed by the ASME Section XI Code Committee and upon approval will replace the Case 1 approximation. Both cases were evaluated and included in this analysis for completeness.

Using these stress intensity factors, the J-integral or $J_{applied}$ can be calculated. The calculation of the J-integral due to the applied loads should account for the full elastic-plastic behavior of the stress-strain curve for the material. For a reactor vessel with a low upper shelf Charpy energy level, the J-integral can be calculated using the stress intensity factor with the plastic-zone correction for plain-strain. This procedure is as follows.

The stress intensity factors due to internal pressure, K_{Ip}, and radial thermal gradients, K_{It}, are first calculated using the actual postulated flaw depth "a". The effective flaw depth for

small-scale yielding, a_{eff}, is then calculated by using:

$$a_{eff} = a + (1/(6\pi)) \; [(K_{Ip} + K_{It})/\sigma_y]^2$$

where a is in inches, K_{Ip} and K_{It} are in ksi$\sqrt{in.}$ and σ_y is the yield strength for the material in ksi. The stress intensity factors for small-scale yielding due to internal pressure, $K_{Ip}(a_{eff})$, and due to radial thermal gradients, $K_{It}(a_{eff})$, are then calculated by substituting a_{eff} in place of "a" into the appropriate stress intensity factor equations given above.

The J-integral due to the applied loads for small-scale yielding is given by:

$$J = 1000 \; [K_{Ip}(a_{eff}) + K_{It}(a_{eff})]^2/E'$$

where

$$E' = E/(1 - \upsilon^2)$$

and J is in-lb/in^2, E is Youngs Modulus in ksi, and υ is Poisson's ratio.

The criteria as described in the Criteria Synopsis section mandate a plant-specific heatup or cooldown rate be utilized in the analysis for the thermal loading while a constant pressure of 1.15 or 1.25 times the maximum accumulation pressure is assumed. A cooldown rate of 100°F per hour was assumed in the analysis which is the maximum allowed by the plant technical specifications. A cooldown, as opposed to a heatup is utilized because a cooldown causes tensile stress on the inside surface, whereas the heatup causes compressive stresses. Additionally, the inside surface is where degradation due to irradiation is the greatest. The pressure loading will also cause stresses to be tensile, consequently the cooldown is the governing transient.

The maximum accumulation pressure which satisfies ASME Section III, NB-7311(b) is 2734 psi for all PWR systems manufactured by Westinghouse. This is used in conjunction with the safety factors as defined previously.

In each case, the axial flaw yielded the greatest values. The results obtained using this evaluation with the appropriate inputs for the 2, 3 and 4 loop cases described in Table 3 are given in Table 4, along with the J-R curve material values from Table 2. Based on Table 4, all participating plants meet the Level A & B Appendix X criteria.

Additionally, a test case was evaluated using LEFM techniques to validate the Reference 11 approach. The through-wall stress distribution was first determined using the appropriate material and geometry inputs. It was calculated using the WECAN[12]

computer code. A two dimensional finite element model was generated to model the reactor vessel beltline region using the inputs as defined in the previous sections.

Subsequently, the stress intensity factor (K_I) as described previously was calculated using the peak stress distribution for a range of postulated flaw depths. Since the stresses are in the elastic range, $J_{applied}$ could then be calculated directly from the following relation:

$$J_{applied} = \frac{K_I^2}{E}$$

Based on this evaluation, it was concluded that the approach identified in Reference 11 produced conservative results.

3. Analysis for Level C and D Conditions

The stress levels achieved by imposing Level C and D transients can exceed the material yield strength. Consequently, an elastic-plastic fracture mechanics analysis is required for these conditions.

The first step in performing the reactor vessel integrity assessment is the selection of the limiting, or bounding transient to represent the emergency and faulted conditions. An assessment was conducted to determine the limiting Level C and D transients. Level C and D transients that may potentially impact the reactor vessel are as follows per Reference 13:

Level C Transients

Small Loss-of-Coolant Accident
Small Steam Line Break
Complete Loss of Flow
Small Feedwater Line Break

Level D Transients

Reactor Coolant Pipe Break (Large LOCA)
Large Steam Line Break
Large Feedwater Line Break
Reactor Coolant Locked Rotor
Control Rod Ejection
Steam Generator Tube Rupture
Simultaneous Steam Line Feedwater Line Break

The criteria for choosing the limiting transient were peak stress as well as the overall magnitude of total through-wall stress. Based on a review of the above transient results, it was judged that the small steam line break was the limiting Level C

transient and large LOCA and large steam line break were limiting Level D transients. Elastic-plastic stress analyses were performed for a two-dimensional finite element model of a typical 4-loop Westinghouse reactor vessel using the WECAN computer code for all of these transients.

Using the small and large steam line break stress distributions, $J_{applied}$ and dJ/da were calculated using the PCFAD[14] computer code. PCFAD is a fracture mechanics computer code for use in performing safety analysis for flawed structures against failure due to the application of a postulated load. The procedure used here has been referred to as the failure assessment diagram approach for prediction of instability loads. The procedure uses a diagram with the stress intensity factor/fracture toughness ratio as the ordinate and the applied stress/net section plastic collapse stress ratio as the abscissa. For a particular stress level, flaw size and geometry, the coordinates can be readily calculated. $J_{applied}$, and subsequently dJ/da can be determined using these coordinate values.

The PCFAD results for each of the cases described in Table 3 using the stresses determined by the finite element analysis are given in Tables 5 and 6, along with the limiting material values. It should be noted that only the stability conditions are evaluated for level D conditions. Based on Tables 5 and 6, all participating plants meet the level C and D criteria.

Table 4
Level A and B Conditions

Plant Type	Applied				Material		Met Criteria ?
	$J_{0.1}$ (in-lbs/in^2)		$dJ_{applied}/da$		$J_{0.1}$ (in-lbs/in^2)	$dJ_{material}/da$	
	Case 1	Case 2	Case 1	Case 2			
2 Loop	370	384	310	318	702	2925	Yes
3 Loop	472	500	320	321	585 (527) *	2140 (599) *	Yes
4 Loop	554	590	330	345	614	2330	Yes
* Calculations based on J-R curves in [9].							

Table 5
Level C Conditions

Case	Applied		Material		Met Criteria ?
	$J_{0.1}$ (in-lbs/in^2)	$dJ_{applied}/da$	$J_{0.1}$ (in-lbs/in^2)	$dJ_{material}/da$	
2 Loop	319	360	702	2925	*Yes*
3 Loop	318	320	585 (527) *	2140 (599) *	*Yes*
4 Loop	314	320	614	2330	*Yes*
* Calculations based on J-R curves in [9].					

Table 6
Level D Conditions

Case	Applied	Material	Met Criteria ?
	$dJ_{applied}/da$	$dJ_{material}/da$	
2 Loop	400	2925	*Yes*
3 Loop	400	2140 (599) *	*Yes*
4 Loop	380	2330	*Yes*
* Calculations based on J-R curves in [9].			

III. Results and Conclusions

The WOG bounding analysis was conducted to demonstrate that participating WOG plants' reactor vessels maintain a margin on USE equivalent to that of ASME Section III, Appendix G through license life. This was accomplished by demonstrating that the reactor vessel materials meet ASME Section XI, Appendix X criteria.

J-integral values were calculated for A, B, C and D level conditions using representative geometries of 2, 3, and 4-loop plants. Material J values representing EOL conditions were calculated based on available methodology. Comparison cases were evaluated for each material and each representative geometry.

Applied J-values for Level A and B loading conditions along with the limiting material properties are tabulated in Table 4. Level C and D condition results are tabulated in Tables 5 and 6. Based on the information contained in these tables, all participating WOG plants, including Watts Bar Unit 1 meet the ASME Section XI, Appendix X criteria.

IV. References

1. Code of Federal Register 10 CFR Part 50, Appendix G "Fracture Toughness Requirements."

2. S. Tandon, et al., WCAP-13587, Revision 0, "Reactor Vessel Upper Shelf Energy Bounding Evaluation for Westinghouse Pressurized Water Reactors", February 1993.

3. American Society For Testing And Materials (ASTM), Designation: E-185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels, E 706(IF)".

4. Branch Technical Position - MTEB 5-2, "Fracture Toughness Requirements", included in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition", USNRC.

5. Regulatory Guide 1.99, Proposed Revision 2, "Radiation Damage to Reactor Vessel Materials", U.S. Nuclear Regulatory Commission, February, 1986.

6. ASME Boiler and Pressure Vessel Code, Section III - Division I Appendices, June 1983.

7. Reddy G. B., Ayres D. J. "High-Temperature Elastic-Plastic and Creep Properties for SA533 Grade B and SA508 Materials," Combustion Engineering Research Project 2055-8, Paper NP-2763, December 1982.

8. Eason E. D., et al, NUREG/CR-5729, MCS 910401, "Multivariable Modeling of Pressure Vessel and Piping J-R Data," May 1991.

9. Hiser A. L., Terrell J. B., NUREG/CR-5265, MEA-2320, "Size Effects on J-R Curves for A 302-B Plate," dated January 1989.

10. Begley J. A., WCAP-13554, "Effects of Section Size and Cleanliness on the Upper Shelf and Transition Range Toughness of Three Nuclear Pressure Vessel Steels," August 1992.

11. "Background to Evaluation Procedures in ASME Section XI for Assessment Vessels with Low Upper Shelf Charpy Energy Levels," Aug. 7, 1992.

12. Westinghouse Electric Computer ANalysis (WECAN) Code, File 87-1J7-WESAD-R1, dated December 1987.

13. Systems Standard 1.3.F, Nuclear Steam Supply System, Reactor Coolant System Design Transients for Standard Plants.

14. Bloom J. M., and Lee D. R., "Users Guide for the Failure Assessment Diagram Computer Code FAD, "Babcock and Wilcox, Rev. 4, April 1990.

ENCLOSURE 2

GENERIC LETTER 92-01
REQUEST FOR ADDITIONAL INFORMATION

COMMITMENT LIST

The Final Safety Analysis Report is being amended in Amendment 78 to correct the initial USE values.