DUKE POWER COMPANY

MCGUIRE NUCLEAR STATION

ATTACHMENT 7

WCAP-15315

REACTOR VESSEL CLOSURE HEAD/VESSEL FLANGE REQUIREMENTS EVALUATION FOR OPERATING PWR AND BWR PLANTS

Westinghouse Electric Company LLC

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NSBU-NRC-99-5954

November 4, 1999

The Secretary
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Rulemakings and Adjudications Staff

RE: Petition for Rulemaking

Westinghouse Electric Company LLC petitions the U.S. Nuclear Regulatory Commission to eliminate reactor vessel closure head flange requirements from 10 CFR Part 50, Appendix G. This petition requests that Table 1 in 10 CFR 50, Appendix G be modified such that table footnotes (2) and (6) be removed. The original Table 1 and a suggested modification to Table 1 are provided herein.

Enclosed is the petition and Westinghouse WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants" that sets forth the technical basis for the proposed modification, the petitioner's grounds for and interest in the action requested, and the specific issues and facts that support the petition.

HASepp

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Enclosure

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PETITION FOR RULE MAKING
Modification to Appendix G in 10 CFR 50
by
Westinghouse Electric Company LLC

Proposed Regulatory Text

NRC should modify Table 1 in 10 CFR Part 50, Appendix G removing requirements related to the reactor vessel closure head flange. The basis for this request is set forth in WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants".

Table 1 in 10 CFR 50, Appendix G currently shows the following:

Table 1 – Pressure and Temperature Requirements for the Reactor Pressure Vessel

Operating condition	Vessel pressure[1]	Requirements for pressure-temperature limits	Minimum temperature requirements
Hydrostatic pressure and leak tests (core is not critical):			
1.a Fuel in the vessel	≤20%	ASME Appendix G Limits	(2)
1.b Fuel in the vessel	>20%	ASME Appendix G Limits	(2) +90°F (6)
1.c No fuel in the vessel (Preservice Hydrotest Only)	ALL	(Not Applicable)	(3) +60°F
Normal operation (incl. Heat-up and cool-down), including anticipated operational occurrences:			
2.a Core not critical	≤20%	ASME Appendix G Limits	(2)
2.b Core not critical	>20%	ASME Appendix G Limits	(2) + 120°F (6)
2.c Core critical	≤20%	ASME Appendix G Limits + 40°F	Larger of [(4)] or [(2) + 40°F]
2.d Core critical	>20%	ASME Appendix G Limits + 40°F	Larger of [(4)] or [(2) + 40°F] 160F]
2.e Core critical for BWR (5)	≤20%	ASME Appendix G Limits + 40°F	(2) + 60°F

[1]Percent of the preservice system hydrostatic test pressure.
[2]The highest reference temperature of the material in the closure flange region that is highly stressed by the bolt preload.
[3]The highest reference temperature of the vessel.
[4]The minimum permissible temperature for the inservice system hydrostatic pressure test.
[5]For boiling water reactors (BWR) with water level within the normal range for power operation.
[6]Lower temperatures are permissible if they can be justified by showing that the margins of safety of the controlling region are equivalent to those required for the beltline when it is controlling.

As proposed, a revised Table 1 would read:

"Revised" Table 1 – Pressure and Temperature Requirements for the Reactor Pressure Vessel

Operating condition	Vessel pressure[1]	Requirements for pressure-temperature limits	Minimum temperature requirements
Hydrostatic pressure and leak tests (core is not critical):			
1.a Fuel in the vessel	≤20%	ASME Appendix G Limits	
1.b Fuel in the vessel	>20%	ASME Appendix G Limits	
1.c No fuel in the vessel (Preservice Hydrotest Only)	ALL	(Not Applicable)	([2]) +60°F
Normal operation (incl. Heat-up and cool-down), including anticipated operational occurrences:			
2.a Core not critical	≤20%	ASME Appendix G Limits	
2.b Core not critical	>20%	ASME Appendix G Limits	
2.c Core critical	≤20%	ASME Appendix G Limits + 40°F	([3])
2.d Core critical	>20%	ASME Appendix G Limits + 40°F	([3])
2.e Core critical for BWR ([5])	≤20%	ASME Appendix G Limits + 40°F	

[1]Percent of the preservice system hydrostatic test pressure.
[2]The highest reference temperature of the vessel.
[3]The minimum permissible temperature for the inservice system hydrostatic pressure test.
[4]For boiling water reactors (BWR) with water level within the normal range for power operation.

Background information in support of this petition is provided in the enclosed Westinghouse WCAP-15135, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants".

WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-15315

Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants

October 1999

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NOTICE

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1.0 INTRODUCTION

10 CFR Part 50, Appendix G contains requirements for pressure-temperature limits for the primary system, and requirements for the metal temperature of the closure head flange and vessel flange regions. The pressure-temperature limits are to be determined using the methodology of ASME Section XI, Appendix G, but the flange temperature requirements are specified in 10CFR50 Appendix G. This rule states that the metal temperature of the closure flange regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the pre-service hydrostatic test pressure, which is 621 psig for a typical PWR, and 300 psig for a typical BWR.

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

Improved knowledge of fracture toughness and other issues which affect the integrity of the reactor vessel have led to the recent change to allow the use of K_{Ic} in the development of pressure-temperature curves, as contained in ASME Code Case N640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1".

Figure 1-1 illustrates the problem created by the flange requirements for a typical PWR heatup curve. It is easy to see that the heatup curve using K_{Ic} provides for a much higher allowable pressure through the entire range of temperatures. For this plant, however, the benefit is negated at temperatures below RT_{NDT} +120°F because of the flange requirement of 10 CFR Part 50, Appendix G. The flange requirement of 10 CFR 50 was originally developed using the K_{Ia} fracture toughness, and this report will show that use of the newly accepted K_{Ic} fracture toughness for flange considerations leads to the conclusion that the flange requirement can be eliminated.

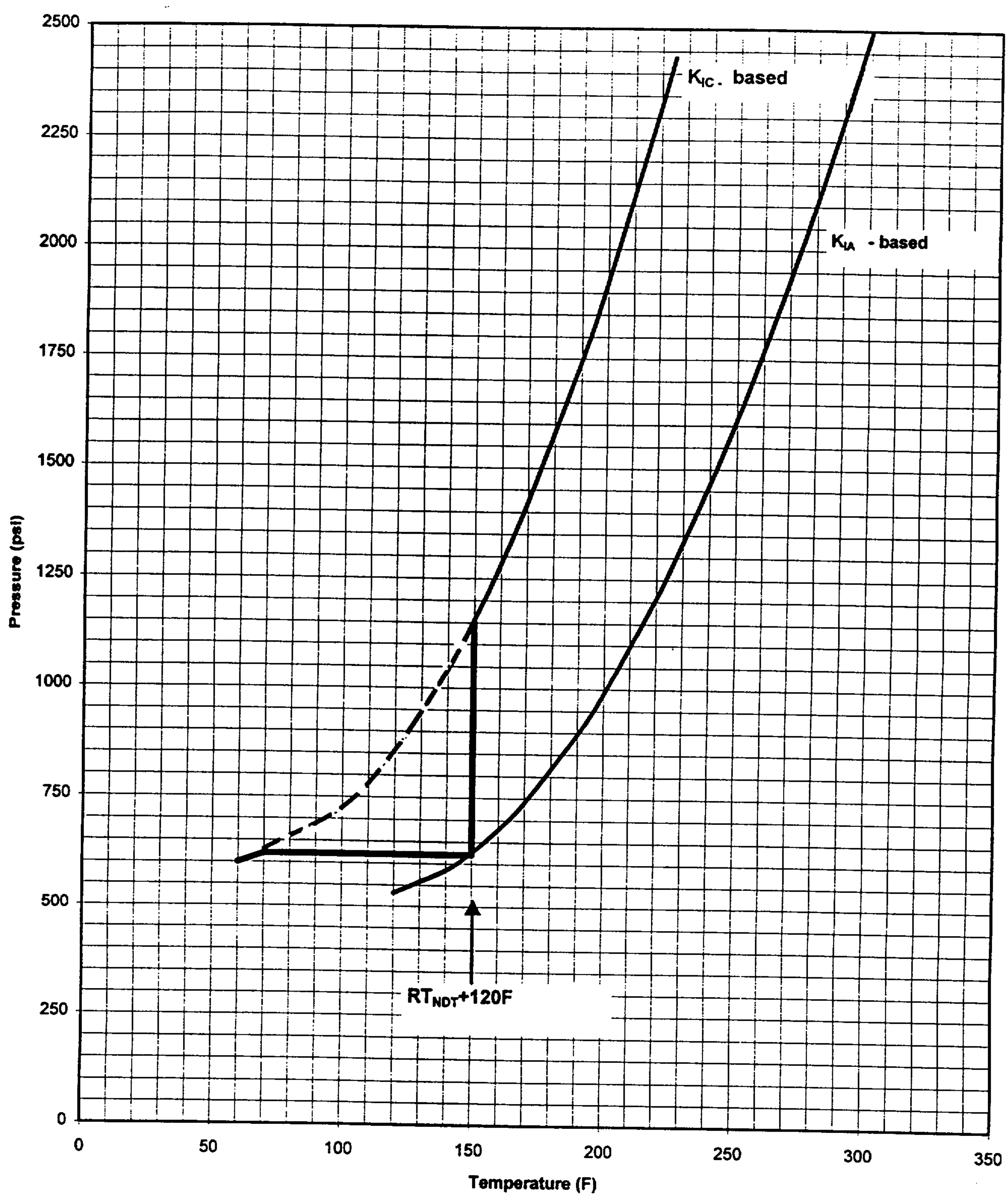


Figure 1-1 Illustration of the Impact of the Flange Requirement for a Typical PWR Plant

2.0 TECHNICAL APPROACH

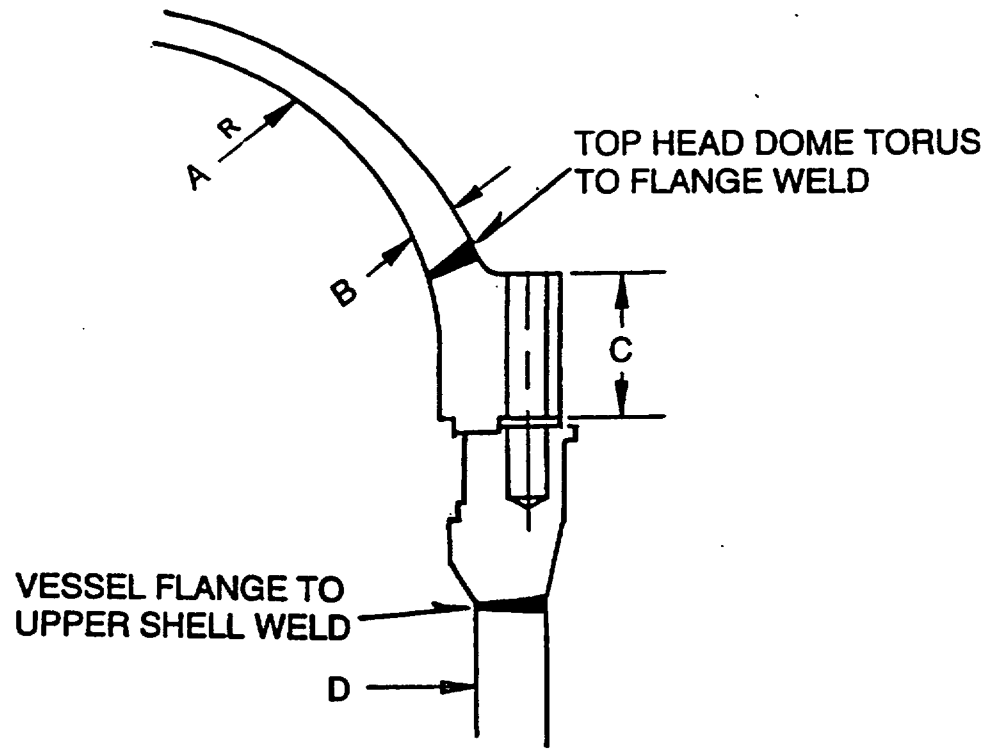
The evaluation to be presented here is intended to cover all operating light water reactor vessels. Fracture evaluations have been performed on the range of geometries which exist, and results will be tabulated and discussed.

The geometry of the closure head region for all the vessels analyzed are shown in Figures 2-1 through 2-4. The geometries for the various PWR vessels are similar, and the same is true for the various BWR vessels. This is also reflected in the stresses, as will be discussed further in Section 4.

Stress analyses have been performed on all of these designs, and these stress results were used to perform fracture mechanics evaluations. The highest stress location in the closure head and vessel flange region is in the head, just above the bolting flange. This corresponds with the location of a weld in nearly all the designs. The highest stressed location is near the outside surface of the head in that region, and so the fracture evaluations have assumed a flaw at this location.

The goal of the evaluation is to compare the integrity of the closure head during the boltup process to the integrity during steady state operation. The question to be addressed is: With the higher K_{Ic} fracture toughness now known to be applicable, is there still a concern about the integrity of the closure head during boltup?

Table 2-1 Geometry Comparison		
Plant Type	Head Thickness	Vessel Diameter
Westinghouse 2 Loop	5.66"	132.4
3 Loop	5.75	155.5
4 Loop	7.0	178.9
B&W	6.63	168.4
Combustion Engineering	7.4	173.4
GE Design 1 (CE)	3.6	109.5
Design 2 (B&W)	4.0	122.4
Design 3 (CBI)	4.8	124.8



UPPER HEAD REGION

	2 LOOP	3 LOOP	4 LOOP
A	83.46	74.59	85.78
B	5.66	5.75	7.00
C	27.56	29.56	27.25
D	132.40	155.50	170.88

NOTE: ALL DIMENSIONS ARE IN INCHES

Figure 2-1 **Geometry of the Upper Head/Flange Region of a Typical Westinghouse Four Loop Plant Reactor Vessel**

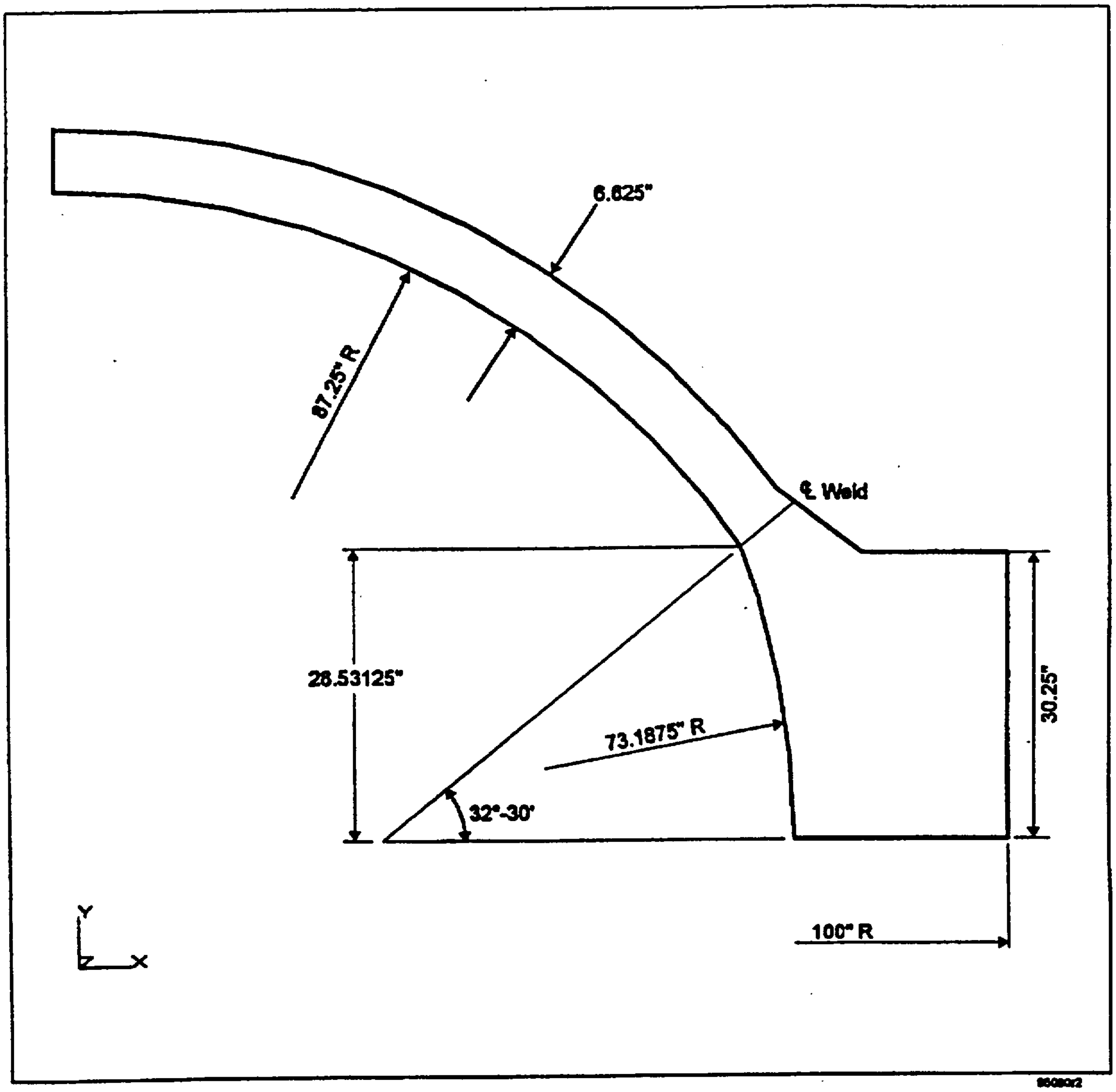


Figure 2-2 Geometry of Closure Head Region – Babcock and Wilcox Reactor Vessels

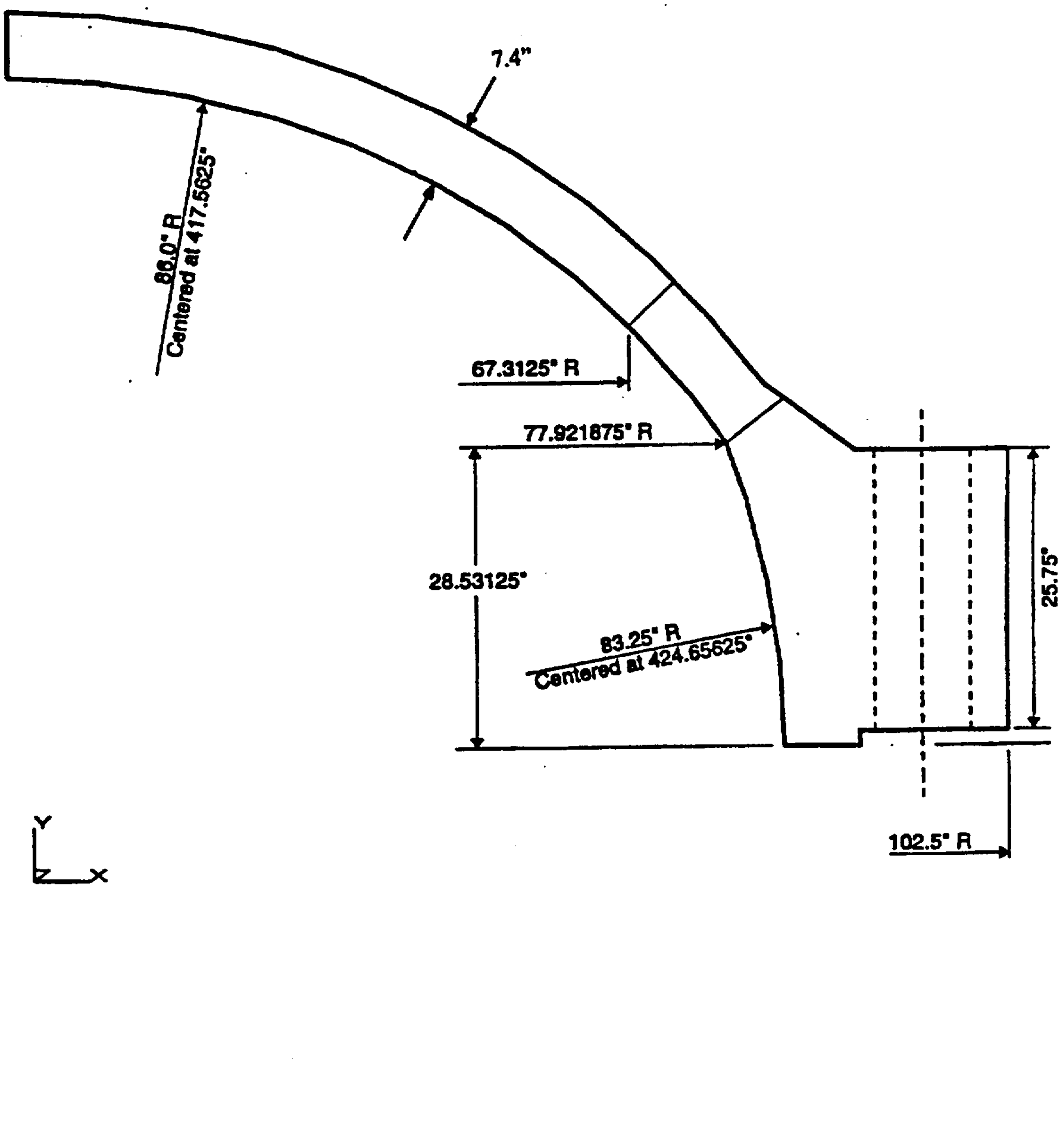
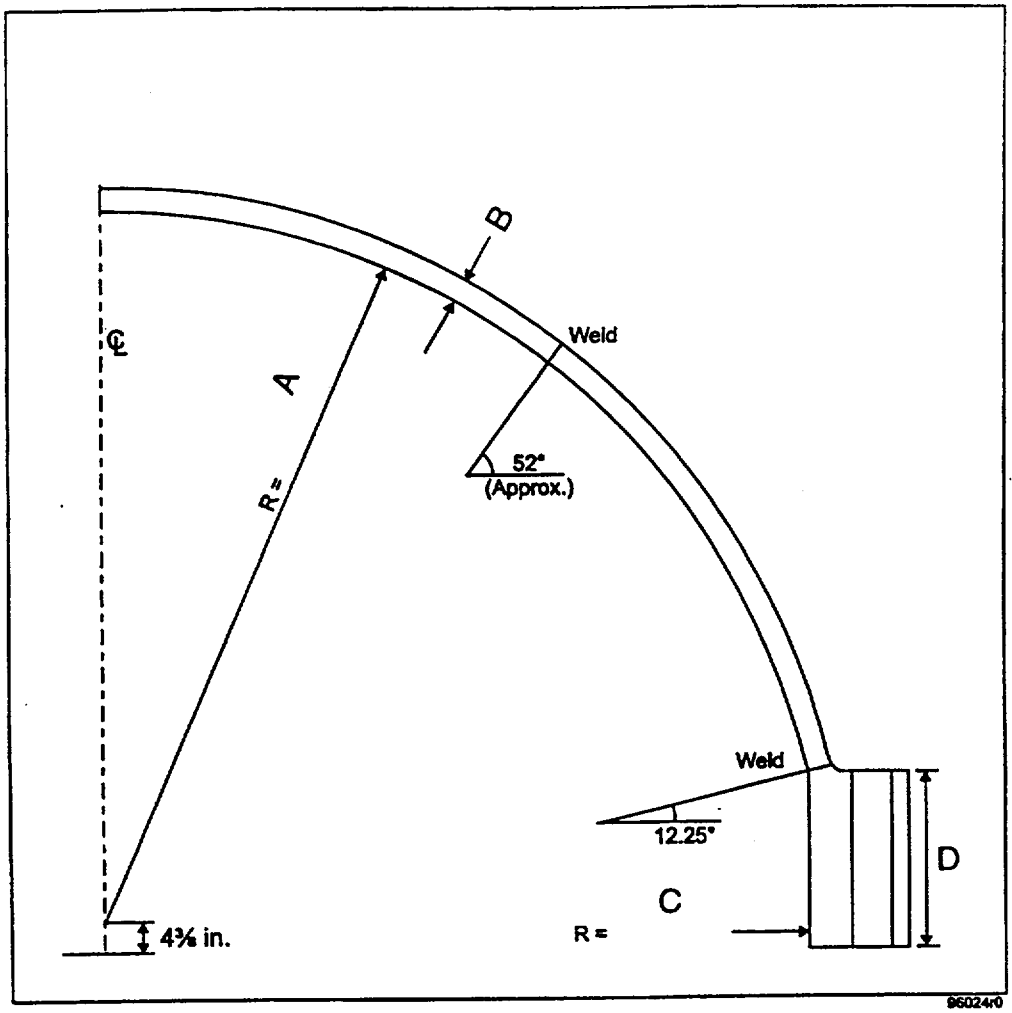


Figure 2-3 Geometry of Closure Head Region: Combustion Engineering Reactor Vessels



	CE	B&W	CB&I
A	109.5	125.6	124.8
B	3.6	4.0	4.8
C	109.5	122.4	124.8
D	24.4	31.0	28.2

NOTE: ALL DIMENSIONS ARE IN INCHES

Figure 2-4 Geometry of Closure Head Region: General Electric Reactor Vessels

3.0 FRACTURE ANALYSIS METHODS AND MATERIAL PROPERTIES

The fracture evaluation was carried out using the approach suggested by Section XI Appendix G.[1] A semi-elliptic surface flaw was postulated to exist in the highest stress region, which is at the outside surface of the closure flange. The flaw depth was set at 25 percent of the wall thickness, and the shape was set at a length six times the depth.

3.1 STRESS INTENSITY FACTOR CALCULATIONS

One of the key elements of a fracture evaluation is the determination of the driving force or stress intensity factor (K_I). This was done using expressions available from the literature. In most cases, the stress intensity factor for the integrity calculations utilized a representation of the actual stress profile rather than a linearization. The stress profile was represented by a cubic polynomial:

$$\sigma(x) = A_0 + A_1 \frac{x}{t} + A_2\left(\frac{x}{t}\right)^2 + A_3\left(\frac{x}{t}\right)^3 \qquad (3\text{-}1)$$

where	x	=	is the coordinate distance into the wall, in.
	t	=	wall thickness, in.
	σ	=	stress perpendicular to the plane of the crack, ksi
	A_i	=	coefficients of the cubic fit

For the surface flaw with length six times its depth, the stress intensity factor expression of Raju and Newman [2] was used when a complete stress distribution was available. The stress intensity factor $K_I(\phi)$ can be calculated anywhere along the crack front. The point of maximum crack depth is represented by $\phi = 0$, and this location was found to also be the point of maximum K_I for the cases considered here. The following expression is used for calculating $K_I(\phi)$, where ϕ is the angular location around the crack. The units of $K_I(\phi)$ are $\text{ksi}\sqrt{\text{in}}$.

$$K_I(\phi) = \left[\frac{\pi a}{Q}\right]^{0.5} \sum_{j=1}^{4} G_j\,(a/c,\ a/t,\ t/R,\ \phi)\,A_j\,a^j \qquad (3\text{-}2)$$

The magnification factors $G_1(\phi)$, $G_2(\phi)$, $G_3(\phi)$ and $G_4(\phi)$ are obtained by the procedure outlined in reference [2]. The dimension "a" is the crack depth, and "c" is the crack length, while t is the wall thickness. In some cases only surface stress values were available, and in these cases the stresses were linearized through the thickness of the head, and the Raju-Newman expression was used.

3.2 FRACTURE TOUGHNESS

The other key element in a fracture evaluation is the fracture toughness of the material. The fracture toughness has been taken directly from the reference curves of Appendix A, Section XI.

In the transition temperature region, these curves can be represented by the following equations:

$$K_{Ic} = 33.2 + 20.734 \exp. [0.02 (T-RT_{NDT})] \tag{3-4}$$

$$K_{Ia} = 26.8 + 12.445 \exp. [0.0145 (T-RT_{NDT})] \tag{3-5}$$

where K_{Ic} and K_{Ia} are in $ksi\sqrt{in}$.

The upper shelf temperature regime requires utilization of a shelf toughness which is not specified in the ASME Code. A value of 200 $ksi\sqrt{in}$ has been used here for all the regions except the nozzle inner radius regions, since the upper shelf Charpy energy exceeds 50 ft-lb, even after irradiation. This value is consistent with general practice in such evaluations, as shown for example in reference [3], which provides the background and technical basis of Appendix A of Section XI.

The other key element in the determination of the fracture toughness is the value of RT_{NDT}, which is a parameter determined from Charpy V-notch and drop-weight tests.

The value of RT_{NDT} for the closure flange region of operating PWR plants was surveyed for 82 PWR plants world wide, and the average value of RT_{NDT} was found to be 9°F. The results ranged from –50°F to +60°F, with the 60°F cases representing the few cases where a test result was not available or the maximum allowed by the ordering requirement. For the head region of operating BWR plants, results ranged up to 40°F, which was the ordering requirement, while the average value of RT_{NDT} was found to be 10°F. Therefore, the value of 10°F was used for the illustrations to be discussed in Sections 4 and 5.

3.3 IRRADIATION EFFECTS

Neutron irradiation has been shown to produce embrittlement which reduces the toughness properties of reactor vessel steels. The decrease in the toughness properties can be assessed by determining the shift to higher temperatures of the reference nil-ductility transition temperature, RT_{NDT}.

The location of the closure flange region is such that the irradiation levels are very low and therefore the fracture toughness is not measurably affected.

4.0 FLANGE INTEGRITY

The first step in evaluation of the closure head/flange region is to examine the stresses. The stresses which are affected by the boltup event are the axial, or meridional stresses, which are perpendicular to the nominal plane of the closure head to flange weld. The stresses in this region during steady state operation are summarized in Table 4-1.

The table shows that the stresses in the various PWR designs are very similar during steady state operation, and stresses are not very high. The loadings are primarily membrane stress, and the bending stresses are somewhat lower. For the BWR designs, the membrane stress is very similar, as might be expected from use of the same design code. The bending stresses are higher for the BWR designs, due to the larger diameter and smaller thickness.

Table 4-2 provides a comparison of the stresses at boltup with those at steady state. It is easy to see that the stresses at boltup are mostly bending, with a very small membrane stress. As the vessel is pressurized, the membrane stresses increase.

The relative impact of these stresses can best be addressed through a fracture evaluation. A semi-elliptic surface flaw was postulated at the outer surface of the closure head flange, and the stress intensity factor, K, (or crack driving force) was calculated. The results are shown for the boltup condition in Figures 4-1 and 4-2. Figure 4-1 shows the results for the governing PWR design (B&W), while Figure 4-2 shows the results for the governing BWR design (B&W, 251 inches). In both cases it can be seen that the applied stress intensity factor at boltup reaches a maximum for a flaw about half way through the head thickness, and then decreases as the flaw extends into the lower stress region near the inside surface of the head. The maximum value of the stress intensity factor for each of the designs is tabulated in Table 4-3, and plots for each of the other design cases appear in the Appendix.

Also shown in Table 4-3 is the fracture toughness at boltup for typical PWR and BWR plants. The boltup temperature for a PWR is typically 60°F, while the boltup temperature for a BWR is typically 80°F. Since we know that the average value of RT_{NDT} is 10°F for all the plants, both the K_{Ic} and K_{Ia} values are easily calculated.

Study of Table 4-3 shows the difference in the integrity story using the two values of fracture toughness. Using the K_{Ia} toughness (which was the basis for the original flange requirements) it can be seen that the applied stress intensity factor exceeds the toughness for two cases, cases 2 and 6, for flaws about half way through the head thickness.

Using the K_{Ic} toughness, which has now been adopted by Section XI for P-T Curves, it can be seen that there is significant margin between the applied stress intensity factor and the fracture toughness at virtually all crack depths. Another objective of the requirements in Appendix G is to assure that fracture margins are maintained to protect against service induced cracking due to environmental effects. Since the governing flaw is on the outside surface (the inside is in compression) where there are no environmental effects, there is even greater assurance of fracture margin. Therefore it may be concluded that the integrity of the closure head/flange region is not a concern for any of the operating plants using the K_{Ic} toughness.

Furthermore, there are no known mechanisms of degradation for this region, other than fatigue. The calculated design fatigue usage for this region is less than 0.1, so it may be concluded that flaws are unlikely to initiate in this region.

Table 4-1 Axial Stress Comparison Steady State Operation @ 2250 psi (PWR), 1000 psi (BWR)

Plant	OD Stress (ksi)	Membrane Stress (ksi)	Bending Stress (ksi)
W 4 Loop	22.8	10.0	12.8
W 3 Loop	20.9	11.6	9.3
CE	46.4	12.8	33.6
B&W	55.7	19.0	36.7
GE BWR Design 1 (CE)	49.6	18.0	31.6
GE BWR Design 2 (B&W)	53.0	15.5	37.5
GE BWR Design 3 (CBI)	52.5	14.3	38.2

Table 4-2 Stress Comparison Boltup vs. Steady State

Plant	Boltup Membrane (ksi)	Boltup Bending (ksi)	SS Membrane (ksi)	SS Bending (ksi)
W 4 Loop	1.1	14.2	10.0	12.8
W 3 Loop	2.1	14.5	11.6	9.3
CE	0.8	22.8	12.8	33.6
B&W	4.3	27.6	19.0	36.7
GE BWR Design 1 (CE)	0.8	26.3	18.0	31.6
GE BWR Design 2 (B&W)	0.5	48.5	15.5	37.5
GE BWR Design 3 (CBI)	0.5	35.5	14.3	38.2

Table 4-3 Flange Integrity Results at Boltup

Design	Maximum K, in $ksi\sqrt{in}$ (Flaw Depth/Thickness)	Fracture Toughness at Boltup*	
		K_{Ia} ($ksi\sqrt{in}$)	K_{Ic} ($ksi\sqrt{in}$)
1. CE	41 (.42)	52.7	89.6
2. B&W	56 (.60)	52.7	89.6
3. W Four Loop	31 (.44)	52.7	89.6
4. W Three Loop	32 (.44)	52.7	89.6
5. GE BWR (CBI 251)	56 (.42)	61.4	117.3
6. GE BWR (B&W 251)	69 (.40)	61.4	117.3
7. GE BWR (CE 218)	37 (.42)	61.4	117.3

*Boltup is typically at 60°F for PWRs, and 80°F for BWRs.

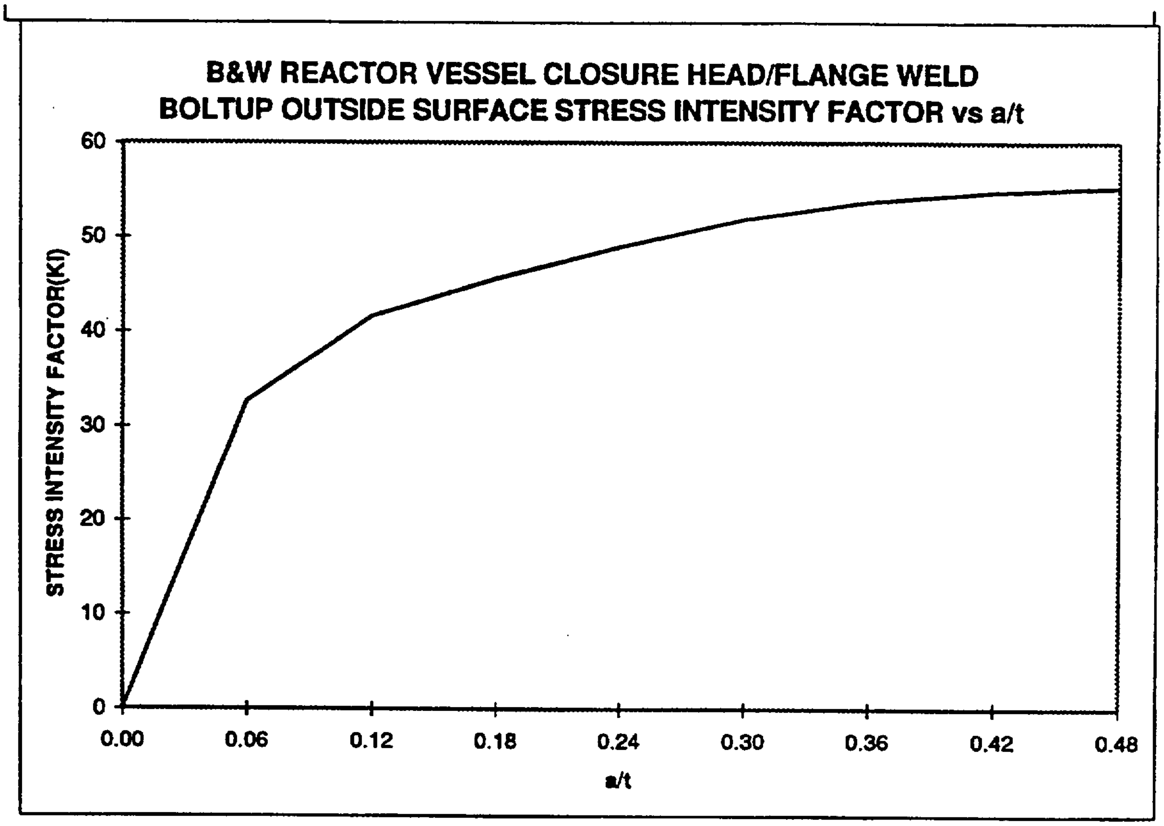


Figure 4-1. Crack Driving Force as a Function of Flaw Size: Outside Surface Flaw in the Closure Head to Flange Region Weld for the Governing PWR Design

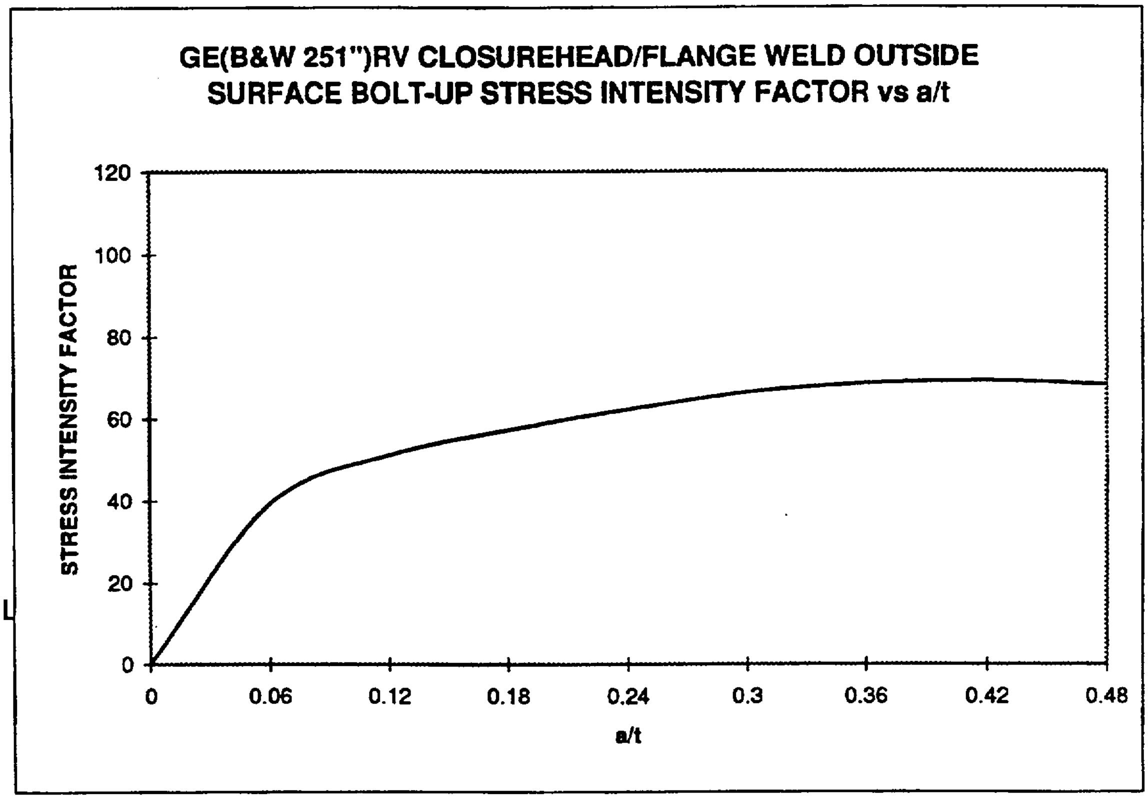


Figure 4-2. Crack Driving Force as a Function of Flaw Size: Outside Surface Flaw in the Closure Head to Flange Region Weld, for the Governing BWR Design

5.0 ARE FLANGE REQUIREMENTS NECESSARY?

Using the K_{Ic} curve can support the elimination of the flange requirement. This can be illustrated by examining the stress intensity factor change for a postulated flaw as the vessel is pressurized after boltup, progressing up to steady state operation.

The stresses at the region of interest are shown in Table 4-1, for steady state operation. Included here are the stresses at the outside surface, which is the highest stress location for this region, as well as the membrane and bending stresses. Table 4-2 shows a comparison of the boltup and steady state stresses for the same plant designs. The results are similar for the designs shown, which bracket all plants in service. No comparisons are available for two loop Westinghouse plants, but they are conservatively covered by the four loop Westinghouse plant results.

As the vessel is pressurized, the stresses in the closure flange region gradually change from mostly bending stresses to a combination of bending and membrane stresses. The stress intensity factor, or driving force, increases for a postulated flaw at the outside surface, as the vessel is pressurized.

As mentioned in Section 4, the boltup temperature for a PWR is nominally 60°F, while that for a BWR is nominally 80°F. From Section 3, the average value of RT_{NDT} for the closure head material is 10°F for all the designs, so boltup is typically at RT_{NDT} + 50 for PWRs, and RT_{NDT} + 70°F for BWRs.

A direct comparison between the original basis for the boltup requirement and the new K_{Ic} approach is provided in Table 5-1. This table provides calculated boltup requirements for all the designs, using a safety factor of 2, and a reference flaw depth of a/t = 0.10, which was used by Randall as the basis for the original requirement [4]. The boltup requirements using K_{Ia} are shown in the right-most column, and the governing case would have a boltup requirement of RT_{NDT} + 118°F, which closely matches the requirement of RT_{NDT} +120°F now in 10CFR50 Appendix G.

Now consider the equivalent result using K_{Ic}, which is just to the left of the column just discussed. The boltup requirement using the same margin now ranges from RT_{NDT} to RT_{NDT} + 41F for PWR plants, and from RT_{NDT} to RT_{NDT} + 56 for BWR plants. Since the average value of RT_{NDT} is 10°F for all the plants, the boltup requirements can be easily translated into actual temperatures. For PWRs the requirement for boltup ranges from 10°F to 51°F, and the actual boltup temperature is 60°F. For BWRs the requirement ranges from 10°F to 66°F, and the actual boltup temperature is 80°F. It is therefore clear that no additional boltup requirements are necessary, and therefore the requirement can be eliminated from 10CFR50 Appendix G.

Table 5-1 Comparison of Various Plant Designs Boltup Requirements

Plant	K (a/t = .1)	K with SF = 2	$T - RT_{NTD}$ (°F) using K_{Ic} (a/t = .10)	$T - RT_{NTD}$ (°F) using K_{Ia} (a/t = .10)
CE	30.0	60.0	13	68
B&W	39.4	79.8	41	100
W 4 Loop	19.7	39.4	0	1
W 3 Loop	19.4	38.8	0	0
GE (CBI 251")	38.7	77.4	38	97
GE (B&W 251")	48.0	96.0	56	118
GE (CE 218")	25.1	50.2	0	43

*All units in $ksi\sqrt{in}$

6.0 SAFETY IMPLICATIONS OF THE FLANGE REQUIREMENT

There are important safety implications which are associated with the flange requirement, as illustrated by Figure 6-1. The safety concern is the narrow operating window at low temperatures forced by the flange requirement. The flange requirement sets a pressure limit of 621 psi for a PWR (20 percent of hydrotest pressure). Thus, no matter how good the toughness of the vessel, the P-T limit curve may be superceded by the flange requirement for temperatures below RT_{NDT} + 120°F. This requirement was originally imposed to ensure the integrity of the flange region during boltup, but Section 4 has shown that this is no longer a concern.

The flange requirement can cause severe operational limitations when instrument uncertainties are added to the lower limit (621 psi), for the Low Temperature Overpressure Protection system of PWRs. The minimum pressure required to cool the seals of the main coolant pumps is 325 psi, so the operating window sometimes becomes very small, as shown schematically in Figure 6-1. If the operator allows the pressure to drop below the pump seal limit, the seals could fail, causing the equivalent of a small break LOCA, a significant safety problem. Elimination of the flange requirement will significantly widen the operating window for most PWRs.

An example will be provided to illustrate this situation for an operating PWR plant, Byron Unit 1. This is a forging-limited vessel at 12 EFPY, with a low leakage core, and low copper weld material in the core region. The vessel has excellent fracture toughness, which means that the flange notch is very prominent, as shown in the vessel heatup curve of Figure 6-2. As illustrated before in Figure 6-1, Byron has the LTOP setpoints significantly below the flange requirement of 621 psi, because of a relatively large instrument uncertainty. The setpoints of the two power operated relief valves are staggered by about 16 psi to prevent a simultaneous activation. The two PORVs have different instrument uncertainties, and for conservatism the higher uncertainty is used. A similar situation exists for cooldown, as shown in Figure 6-3.

Elimination of the flange requirement for Byron Unit 1 would mean that the PORV curve could become level at 604/587 psig, which are the leading/trailing setpoints to protect the PORV downstream piping, through the temperature range of the 350°F down to boltup at 60°F. The operating window between the leading PORV and the pump seal limit rises from 121 psig (446-325) to 262 psig (587-325). This change will make a significant improvement in plant safety by reducing the probability of a small LOCA, and easing the burden on the operators.

This is only one example of the impact of the flange requirement. Every operating PWR plant will have a different situation, but the operational safety level will certainly be generally improved by the elimination of this unnecessary requirement.

Elimination of the flange requirement has no impact on BWRs. The saturation temperature corresponding to the 300 psig operating pressure (20% of the pre-service hydrostatic test pressure) is 420°F. This is well in excess of the RT_{ndt} + 120°F requirement. Therefore the flange temperature requirements are satisfied regardless of whether they exist or not. Therefore, elimination of the flange temperature requirement has no impact on BWR flange integrity.

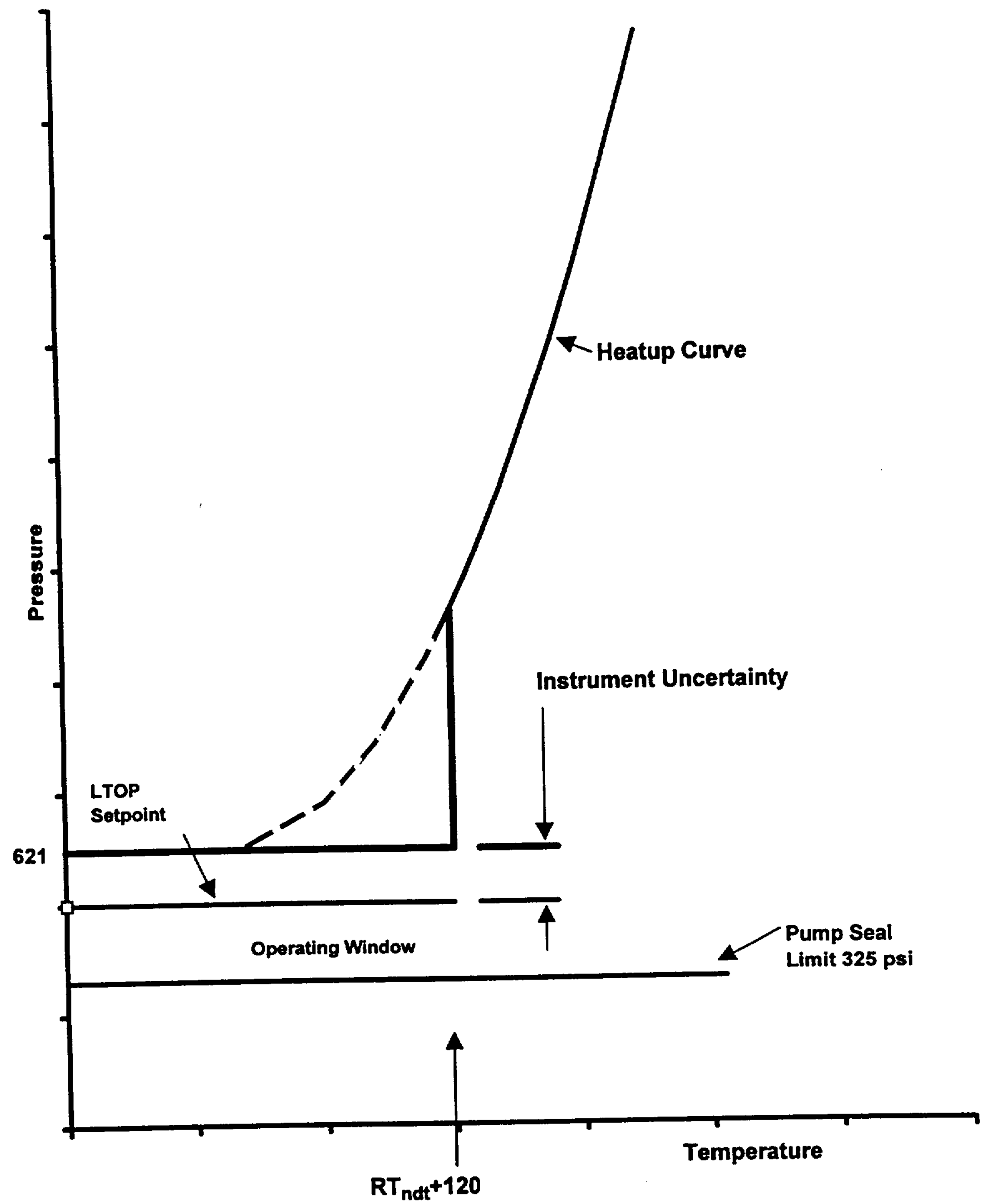


Figure 6-1 Illustration of the Flange Requirement and its Effect on the Operating Window for a Typical Heatup Curve

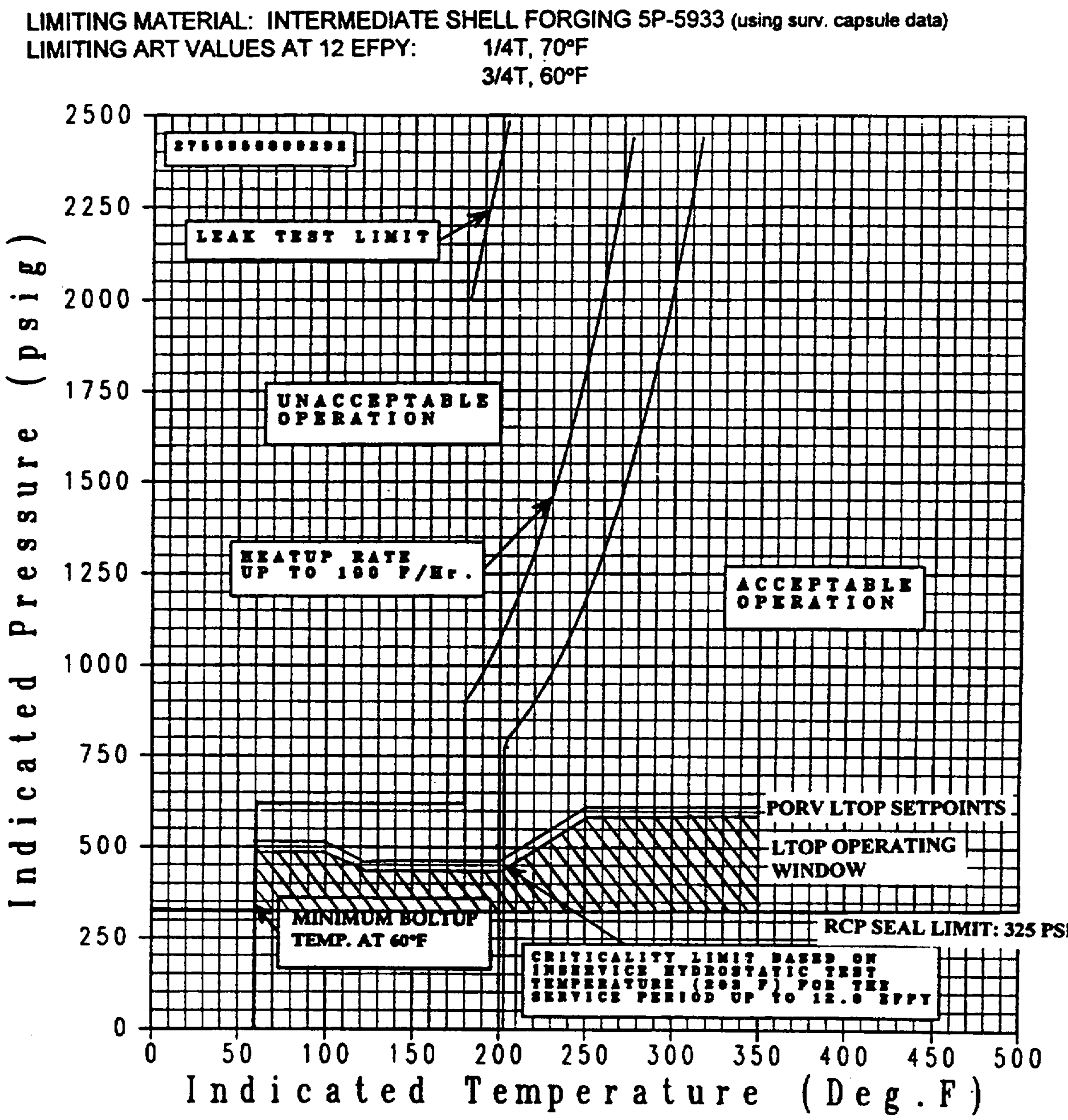


Figure 6-2 Illustration of the Actual Operating Window for Heatup of Byron Unit 1, a Low Copper Plant at 12 EFPY

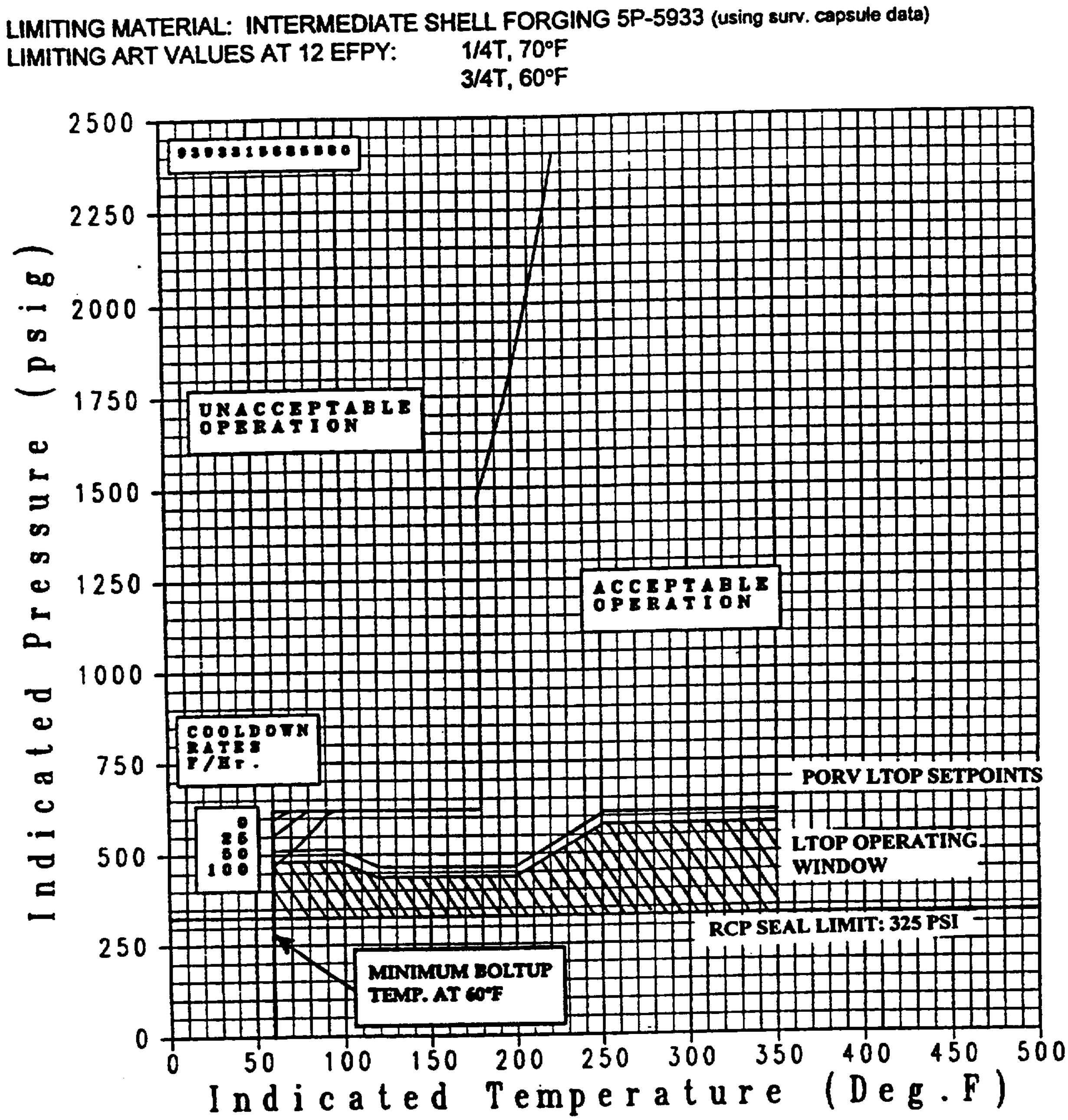


Figure 6-3 Illustration of the Actual Operating Window for Cooldown of Byron Unit 1, a Low Copper Plant at 12 EFPY

7.0 REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, 1996 Addenda, ASME, New York.

2. Newman, J. C. Jr. and Raju, I. S., "Stress Intensity Factors for Internal Surface Cracks in Cylindrical Pressure Vessels," Trans. ASME, Journal of Pressure Vessel Technology, Vol. 102, 1980, pp 342-346.

3. Marston, T. U., ed., "Flaw Evaluation Procedures: ASME Section XI," Electric Power Research Institute Report EPRI-NP-719 SR, August 1978.

4. Randall, N., Abstract of Comments and Staff Response to Proposed Revision to 10 CFR Part 50, Appendices G and H, Published for Comment in the Federal Register, November 14, 1980.

Appendix 1

Stress Intensity Factor Curves for the Boltup Condition

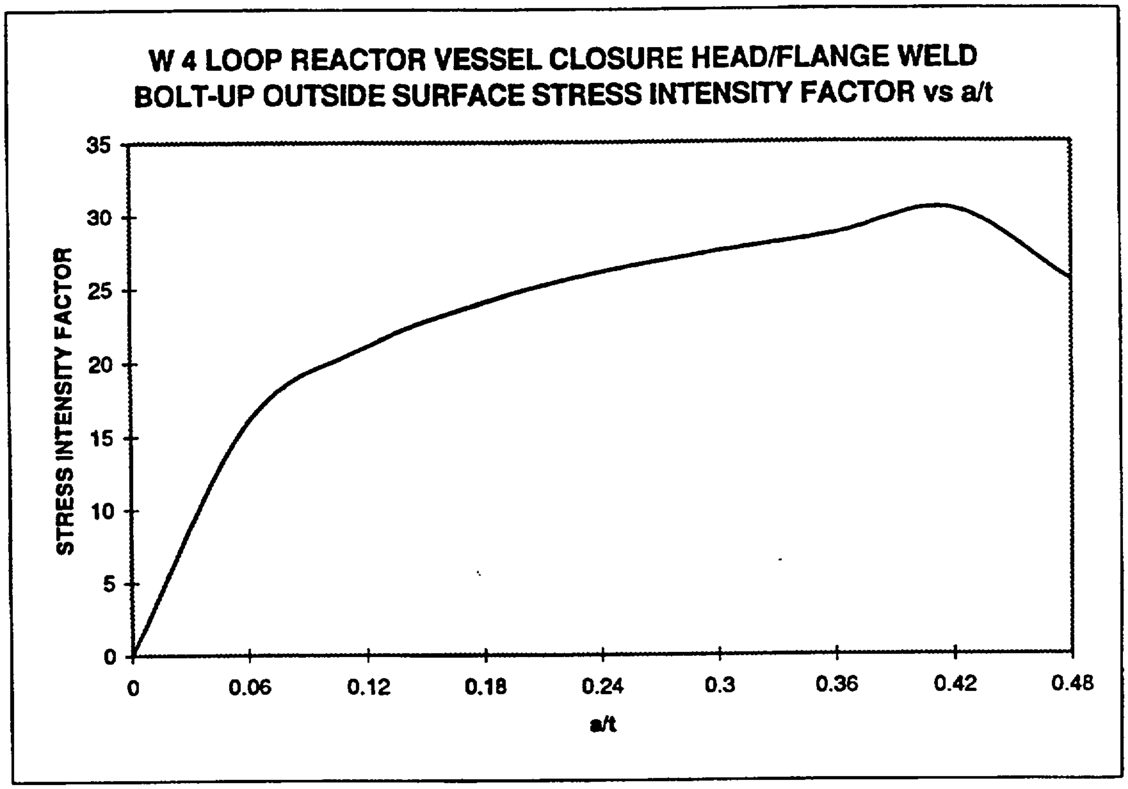


Figure A-1. **Crack Driving Force as a Function of Flaw Size: Outside Surface Flaw in the Closure Head to Flange Region Weld for the Westinghouse Four Loop Plant Design**

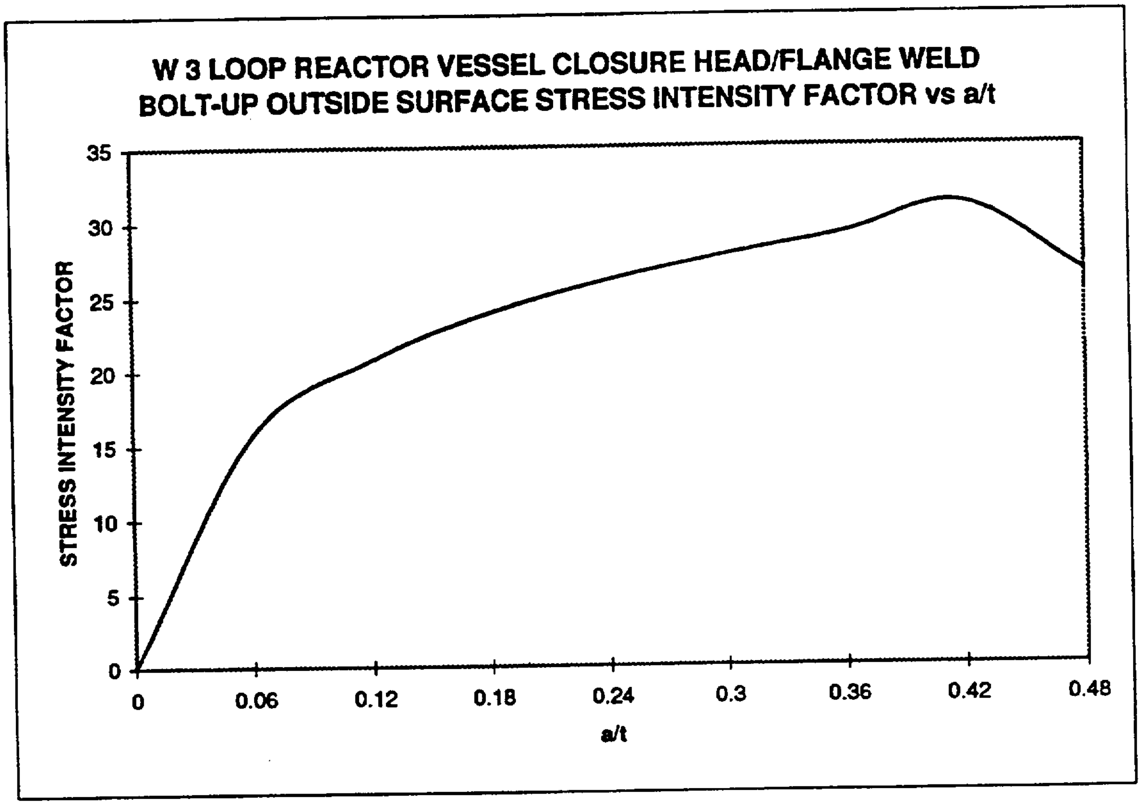


Figure A-2. **Crack Driving Force as a Function of Flaw Size: Outside Surface Flaw in the Closure Head to Flange Region Weld for the Westinghouse Three Loop Plant Design**

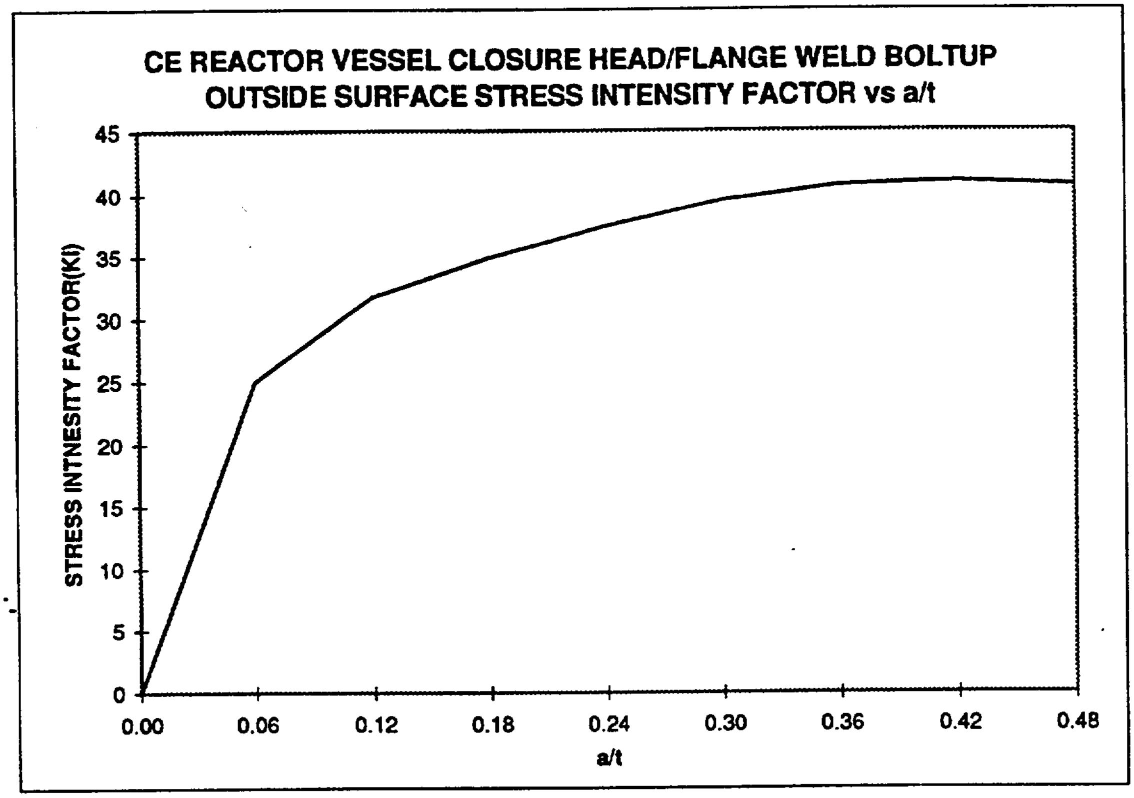


Figure A-3. Crack Driving Force as a Function of Flaw Size: Outside Surface Flaw in the Closure Head to Flange Region Weld for the Combustion Engineering Design

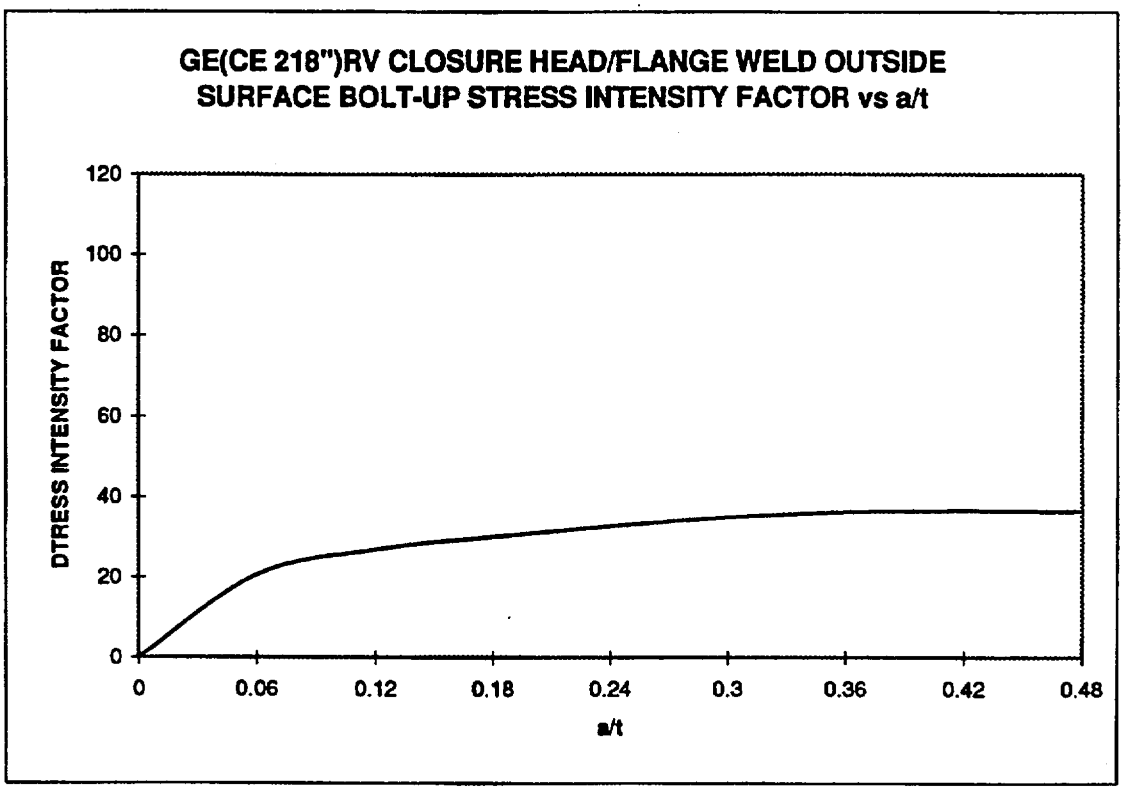


Figure A-4. Crack Driving Force as a Function of Flaw Size: Outside Surface Flaw in the Closure Head to Flange Region Weld for the General Electric – CE Fabricated Design

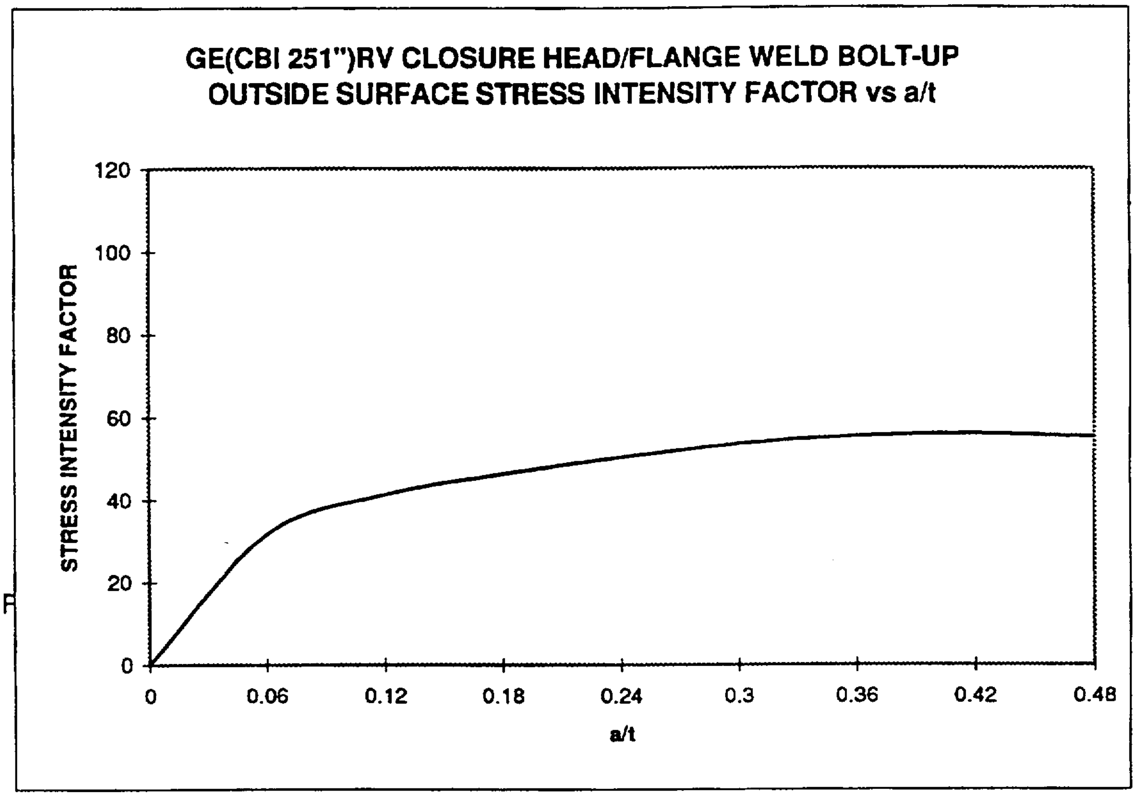


Figure A-5. **Crack Driving Force as a Function of Flaw Size: Outside Surface Flaw in the Closure Head to Flange Region Weld for the General Electric – CBI Fabricated Design**

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ATTACHMENT 8

TECHNICAL SPECIFICATION AND NUREG-1431 MARKED-UP PAGES AS INCLUDED IN MAY 27, 1997 SUBMITTAL

TABLE 3.3-3 (Continued)

TABLE NOTATION

A.1 (a) # ~~Trip function may be blocked in this MODE below~~ above the P-11 (Pressurizer Pressure Interlock) Setpoint.

A.1 (c) ## Trip function automatically blocked above P-11 and may be blocked below P-11 when ~~Safety Injection~~ Main Steam Isolation on low steam pressure is not blocked. *

A.1 ~~** These values left blank pending NRC approval of three loop operation.~~

A.15 ~~Note 1: Turbine driven auxiliary feedwater pump will not start on a blackout signal coincident with a safety injection signal.~~

ACTION STATEMENTS

A.1 One train inoperable

A.27 Restore train to OPERABLE status in 6 hours, or

ACTION ~~14~~ C ~~With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement,~~ be in at least HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours; however, one ~~channel~~ train may be bypassed for up to 4 hours for surveillance testing ~~per Specification 4.3.2.1~~, provided the other ~~channel~~ train is OPERABLE.

A.1 move to Action C Note

A.34

ACTION ~~15~~ D ~~With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required OPERATIONAL TEST provided~~ the inoperable channel is placed in the tripped condition within 6 hours.

A.1 One channel inoperable

M.11 OR Be in MODE 3 in 12 hours and Be in MODE 4 in 18 hours.

L.22 Note: One channel may be bypassed for up to 4 hours for surveillance testing.

ACTION 15a With the number of OPERABLE channels less than the Total Number of Channels, operation may proceed until performance of the next required OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 6 hours. With more than one channel inoperable, enter Specification 3.8.1.1.

moved to ITS 3.3.5

~~Note: One channel may be bypassed for up to 4 hours for surveillance testing.~~

L.22

ACTION ~~15b~~ P ~~With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required OPERATIONAL TEST provided~~ the inoperable channel is placed in the tripped condition within 1 hour.

A.1 One channel inoperable

Or Be in MODE 3 in 7 hours and Be in MODE 4 in 13 hours

A.1 One containment pressure channel inoperable

M.20 in 6 hours

ACTION ~~16~~ E ~~With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided~~ the inoperable channel is placed in the bypassed condition ~~and the Minimum Channels OPERABLE requirement is met~~. One additional channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1.

A.1 move to Action E Note

M.11 OR Be in MODE 3 in 12 hours AND Be in MODE 4 in 18 hours

moved to ITS 3.3.6

ACTION 17 With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge supply and exhaust valves are maintained closed.

* License amendment request 10/97

TABLE 3.3-3 (Continued)

TABLE NOTATION

A.1 (a) # Above Trip function may be blocked in this MODE below the P-11 (Pressurizer Pressure Interlock) Setpoint.

A.1 (c) ## Trip function automatically blocked above P-11 and may be blocked below P-11 when ~~Safety Injection~~ Main Steam Isolation on low steam pressure is not blocked. | *

A.1 ** ~~These values left blank pending NRC approval of three loop operation.~~

LA.15 ~~Note 1: Turbine driven auxiliary feedwater pump will not start on a blackout signal coincident with a safety injection signal.~~

ACTION STATEMENTS One train inoperable A.1 Restore train to OPERABLE status in 6 hours, or A.27

A.1 MOVE TO ACTION C NOTE

ACTION ~~14~~ C ~~With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement,~~ be in at least HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours; ~~however,~~ one ~~channel~~ train may be bypassed for up to 4 hours for surveillance testing ~~per Specification 4.3.2.1~~, provided the other ~~channel~~ train is OPERABLE. L.22

A.34 M.11 OR Be in MODE 3 in 12 hours and Be in MODE 4 in 18 hours

ACTION ~~15~~ D A.1 One channel inoperable ~~With the number of OPERABLE channels one less than the Total Number of Channels,~~ operation may proceed ~~until performance of the next required OPERATIONAL TEST provided~~ the inoperable channel is placed in the tripped condition within 6 hours. Note: One channel may be bypassed for up to 4 hours for surveillance testing.

MOVED TO ITS 3.3.5

~~ACTION 15a With the number of OPERABLE channels less than the Total Number of Channels, operation may proceed until performance of the next required OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 6 hours. With more than one channel inoperable, enter Specification 3.8.1.1.~~

L.22 Note: One channel may be bypassed for up to 4 hours for surveillance testing.

OR Be in MODE 3 in 7 hours and be in MODE 4 in 13 hours

ACTION ~~15b~~ P One channel inoperable A.1 ~~With the number of OPERABLE channels one less than the Total Number of Channels,~~ operation may proceed ~~until performance of the next required OPERATIONAL TEST provided~~ the inoperable channel is placed in the tripped condition within 1 hour. |

M.11 OR Be in MODE 3 in 12 hours AND Be in MODE 4 in 18 hours

ACTION ~~16~~ E One containment pressure channel inoperable A.1 ~~With the number of OPERABLE channels one less than the Total Number of Channels,~~ operation may proceed ~~provided~~ the inoperable channel is placed in the bypassed condition in 6 hours M.20 ~~and the Minimum Channels OPERABLE requirement is met~~. One additional channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1. A.1 MOVE TO ACTION E NOTE

MOVED TO ITS 3.3.6

~~ACTION 17 With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge supply and exhaust valves are maintained closed.~~

* License amendment request 10/97

Table 3.3.2-1 (page 4 of 8)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT (a) 3
4. Steam Line Isolation (continued)						
c. Containment Pressure – High 3 High	1,2 (j), 3 (j) b	1 [4] (e)	E 5 D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 8 SR 3.3.2.10 9	3.0 ≤ [6.61] psig	2.9 ≤ [6.35] psig 1
d. Steam Line Pressure						
(1) Low	1,2 (j), b 3 (k)(j) a	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 8 SR 3.3.2.10 9	755 1 ≥ [635] (a) 3 psig	775 ≥ [675] (a) 3 psig
(2) Negative Rate – High	3 (g)(i) b c	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 8 SR 3.3.2.10 9	120 d ≤ [121.6] (h) psi/sec	100 d ≤ [110] (h) psi/sec 1
e. High Steam Flow in Two Steam Lines	1,2 (i), 3 (i)	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	(e)	(f)
Coincident with T_{avg} – Low Low	1,2 (i), 3 (d)(i)	1 per loop	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ [550.6]°F	≥ [553]°F

7

(continued)

(a) Reviewer's Note: Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.

a (b) Above the P-11 (Pressurizer Pressure) interlock.

(c) Time constants used in the lead/lag controller are $t_1 \geq$ [50] seconds and $t_2 \leq$ [5] seconds.

3

(d) Above the P-12 (T_{avg} – Low Low) interlock.

(e) Less than or equal to a function defined as ΔP corresponding to [44]% full steam flow below [20]% load, ΔP increasing linearly from [44]% full steam flow at [20]% load to [114]% full steam flow at [100]% load, and ΔP corresponding to [114]% full steam flow above 100% load.

(f) Less than or equal to a function defined as ΔP corresponding to [40]% full steam flow between [0]% and [20]% load and then a ΔP increasing linearly from [40]% steam flow at [20]% load to [110]% full steam flow at [100]% load.

c (g) Below the P-11 (Pressurizer Pressure) interlock.

d (h) Time constant utilized in the rate/lag controller is ≤ [50] seconds. ≥ 50 1

b (i) Except when all MSIVs are closed and [de-activated].

5

Trip function automatically blocked above P-11 (Pressurizer Pressure) interlock and may be blocked below P-11 when Safety Injection Steam Line Pressure - Low is not blocked.

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ATTACHMENT 9

INDUSTRY EXEMPTIONS AND AMENDMENTS RELATED TO CODE CASE N-640 AND P/T LIMITS

INDUSTRY EXEMPTIONS AND AMENDMENTS RELATED TO CODE CASE N-640 AND P/T LIMITS

Plant Name	Application Date	Exemption Date	Amendment Date
Beaver Valley 2	6/17/00	9/6/00	9/6/00
Clinton	8/25/00	10/30/00	10/31/00
Dresden	2/23/00	8/25/00	9/19/00
Hatch	6/1/00	8/29/00	8/29/00
Oconee	5/11/99	7/23/99	10/1/99
Shearon Harris	4/12/00	7/26/00	7/28/00

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ATTACHMENT 10

NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

Duke Energy Corporation (Duke) has made the determination that this amendment request involves No Significant Hazards Considerations by applying the standards established by NRC regulations in 10CFR50.92.

Standard 1: Would operation of the facility in accordance with the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

No.

The proposed changes to the reactor coolant system (RCS) pressure/temperature (P/T) limits are developed utilizing the methodology of ASME XI, Appendix G, in conjunction with the methodology of Code Case N-640. Usage of these methodologies provides compliance with the underlying intent of 10CFR50 Appendix G and provides operational limits that ensure failure of the reactor vessel will not occur. The changes do not modify the RCS pressure boundary, nor make any physical changes to the facility. The probability of any design basis accident (DBA) is not affected by these changes, nor are the consequences of any DBA affected by these changes. The P/T limits, and low temperature overpressure protection (LTOP) limits and setpoints are not considered to be initiators or contributors to any accident analysis addressed in the McGuire UFSAR.

The proposed changes do not adversely affect the integrity of the RCS such that its function in the control of radiological consequences is affected. The power operated relief valve (PORV) LTOP setpoint is established to protect RCS pressure boundary (UFSAR 5.2.2). The changes do not alter any assumption previously made in the radiological consequence evaluations nor affect the mitigation of the radiological consequences of an accident previously evaluated. Therefore, the probability or consequences of an accident previously evaluated will not be increased by the proposed changes.

Standard 2: Would operation of the facility in accordance with the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

No.

The proposed changes to the RCS P/T limits and LTOP required actions do not affect the assumed accident performance of any structure, system or component previously evaluated. The proposed changes do not introduce any new modes of system operation or failure mechanisms. Consequently, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Standard 3: Would operation of the facility in accordance with the proposed amendment involve a significant reduction in a margin of safety?

No.

The proposed changes are developed utilizing the methodology of ASME XI, Appendix G, in conjunction with Code Case N-640 methodology. Usage of these methodologies provides compliance with the underlying intent of 10CFR50 Appendix G and provides operational limits that ensure failure of the reactor vessel will not occur. Although the Code Case constitutes relaxation from the current requirements of 10CFR50 Appendix G, the alternative methodology allowed by the Code is based on industry experience gained since the inception of the 10CFR50 Appendix G requirements for which some of the requirements have now been determined to be excessively conservative. The more appropriate assumptions and provisions allowed by the Code Case maintain a margin of safety that is consistent with the intent of 10CFR50 Appendix G, i.e., with regard to the margin originally contemplated by 10CFR50 Appendix G for determination of RCS P/T limits. Therefore, there will be no significant reduction in a margin of safety as a result of the proposed changes.

Duke has concluded, based on this information, there are No Significant Hazards Considerations involved in this amendment request.

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ATTACHMENT 11

ENVIRONMENTAL IMPACT ASSESSMENT

ENVIRONMENTAL IMPACT ASSESSMENT

Pursuant to 10CFR51.22(b), an evaluation of the proposed amendment has been performed to determine whether or not it meets the criteria for categorical exclusion set forth in 10CFR51.22 (c)9 of the regulations. **The proposed amendment meets the criteria for categorical exclusion if it does not involve the following:**

1) A significant hazards consideration:

This conclusion is supported by the No Significant Hazards Consideration Evaluation that is contained in Attachment 10.

2) A significant change in the types or significant increase in the amounts of any effluents that may be released offsite:

The proposed changes provide operational limits that ensure failure of the reactor vessel will not occur. The changes do not modify the RCS pressure boundary, nor make any physical changes to the facility. The proposed changes do not adversely affect the integrity of the RCS such that its function in the control of radiological consequences is affected. Therefore, no change in the types or significant increase in the amounts of any effluents that may be released offsite will be involved with the proposed changes.

3) A significant increase in the individual or cumulative occupational radiation exposure:

In addition to the above, the proposed changes do not involve any new modes of system operation or failure mechanisms. Therefore, no significantly increase the individual or cumulative occupation radiation exposure will be involved with the proposed changes.

In conclusion, this amendment request meets the criteria set forth in 10CFR51.22(c)(9) of the regulations for categorical exclusion from an environmental assessment/impact statement.