ENCLOSURE 2

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LOW UPPER-SHELF TOUGHNESS FRACTURE MECHANICS ANALYSIS OF REACTOR VESSEL OF POINT BEACH UNITS 1 AND 2 FOR EXTENDED LIFE THROUGH 53 EFFECTIVE FULL POWER YEARS

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Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessel of Point Beach Units 1 and 2 for Extended Life through 53 Effective Full Power Years

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EXECUTIVE SUMMARY

Nuclear Management Company is considering plant life extension, power uprate to 1678 MWt and removal of hafnium power suppression assemblies from the core for Point Beach Units 1 and 2. As a result of these changes, operating conditions including vessel temperatures and projected fluence values at 53 effective full power years (EFPY) of plant operation have changed. It must be ensured that these changes do not affect the plant adversely from a regulatory compliance point of view. One of the compliance issues is Appendix G to 10 CFR Part 50 where low upper-shelf toughness is addressed. An equivalent margins assessment has to be made for material toughness when the upper-shelf Charpy energy level falls below 50 ft-lb. This report addresses this particular compliance issue regarding low upper-shelf toughness only.

The Charpy upper-shelf value of reactor vessel beltline weld materials at Point Beach Units 1 and 2 may be less than 50 ft lb at 53 EFPY. In order to demonstrate that sufficient margins of safety against fracture remain to satisfy the requirements of Appendix G to 10 CFR Part 50, a low upper-shelf fracture mechanics analysis has been performed. The limiting welds in the beltline region have been evaluated for ASME Levels A, B, C, and D Service Loadings based on the evaluation acceptance criteria of the ASME Code, Section XI, Appendix K.

The analysis presented in this report demonstrates that the limiting reactor vessel beltline weld at Point Beach Units 1 and 2 satisfies the ASME Code requirements of Appendix K for ductile flaw extensions and tensile stability using projected low upper-shelf Charpy impact energy levels for the weld material at 53 EFPY.

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1.0 Introduction

Nuclear Management Company is considering plant life extension, power uprate to 1678 MWt and removal of hafnium power suppression assemblies from the core for Point Beach Units 1 and 2. This document assesses the effect of these proposed changes on the upper-shelf fracture toughness of the reactor vessels. The B&W Owners Group (B&WOG) fracture toughness model was used in the low upper-shelf toughness fracture mechanics analyses of the reactor vessels of the B&WOG Reactor Vessel Working Group (RVWG) which includes the Point Beach Units 1 and 2 reactor vessels. The low upper-shelf toughness analysis for all reactor vessels of the B&WOG RVWG for Levels A & B Service Loadings was documented in BAW-2192PA [1]. An additional fracture mechanics analysis for Levels C & D Service Loadings was carried out for all these reactor vessels and documented in BAW-2178PA [2]. Both these reports have been accepted by the NRC. As a result of a subsequent power uprate, an additional low upper-shelf toughness analysis covering end-of-license and end-of-license renewal fluence values was performed for Point Beach Units 1 and 2 [3]. For the current planned changes, the effect on the reactor vessel materials upper-shelf toughness is assessed in this report.

Welds in the beltline region of all B&W Owners Group Reactor Vessel Working Group plants, including the Point Beach Units 1 and 2 vessels, have been analyzed [1, 2] for 32 effective full power years (EFPY) of operation to demonstrate that these low upper-shelf energy materials would continue to satisfy federal requirements for license renewal. In Reference 3, the Point Beach vessels were analyzed up to their forecasted end-of-license extension periods at a partially uprated power level of 1650MWt with hafnium power suppression assemblies, and both vessels were shown to be acceptable. The purpose of the present analysis is to perform a similar low upper-shelf toughness evaluation of the reactor vessel welds at the Point Beach plants for projected neutron fluences at 53 EFPY.

The present analysis addresses ASME Levels A, B, C, and D Service Loadings. For Levels A and B Service Loadings, the low upper-shelf toughness analysis is performed according to the acceptance criteria and evaluation procedures contained in Appendix K to Section XI of the ASME Code [4]. The evaluation also utilizes the acceptance criteria and evaluation procedures prescribed in Appendix K for Levels C and D Service Loadings. Levels C and D Service Loadings are evaluated using the one-dimensional, finite element, thermal and stress models and linear elastic fracture mechanics methodology of Framatome ANP's PCRIT computer code to determine stress intensity factors for a worst case pressurized thermal shock transient.

2.0 Changes in Operating Condition Parameters

As a result of the planned updates to the Point Beach Units 1 and 2, there are increases in the projected end of life fluences for both the units. There are also changes in the plants' operating temperatures. These inputs were provided by the Nuclear Management Company and included as Appendix A and summarized in this section.

The analysis for current licensed rated power conditions (1540 MWt) gives a maximum cold leg temperature of 544.5°F. As a result of the power uprate to 1678 MWt, the maximum cold leg temperature is reduced to 541.4°F. The projected reactor vessel fluence values at 53 EFPY are provided in Table 2-1. For this analysis, three cases, termed Evaluation Conditions, are studied – uprated power conditions without hafnium assemblies, current power conditions without hafnium assemblies, and current power conditions with hafnium assemblies. Fluence values for these three cases are reported only for the controlling welds identified through review of the results reported in References 1, 2 and 3. Locations of the reactor vessel welds for Point Beach Units 1 and 2 are illustrated in Figures 2-1 and 2-2 respectively.

Table 2-1 Evaluation Conditions

Plant	Weld Location [1]	Weld Number [1]	Cu (wt%) [5]	Ni (wt%) [5]	Fluence (n/cm^2) at 53 EFPY		
					EVALUATION CONDITION 1 Uprated Power Conditions Without Hafnium Assemblies Cold Leg Temp: 541.4°F	EVALUATION CONDITION 2 Current Power Conditions Without Hafnium Assemblies Cold Leg Temp: 544.5°F	EVALUATION CONDITION 3 Current Power Conditions With Hafnium Assemblies Cold Leg Temp: 544.5°F
PB-1	Lower Shell Long.	SA-847	0.23	0.52	3.37E+19	3.12E+19	2.67E+19
	Inter. Shell/Lower Shell Circ.	SA-1101	0.23	0.59	4.91E+19	4.52E+19	3.82E+19
PB-2	Inter. Shell/Lower Shell Circ.	SA-1484	0.26	0.60	5.09E+19	4.65E+19	3.79E+19

Figure 2-1 Reactor Vessel of Point Beach Unit 1

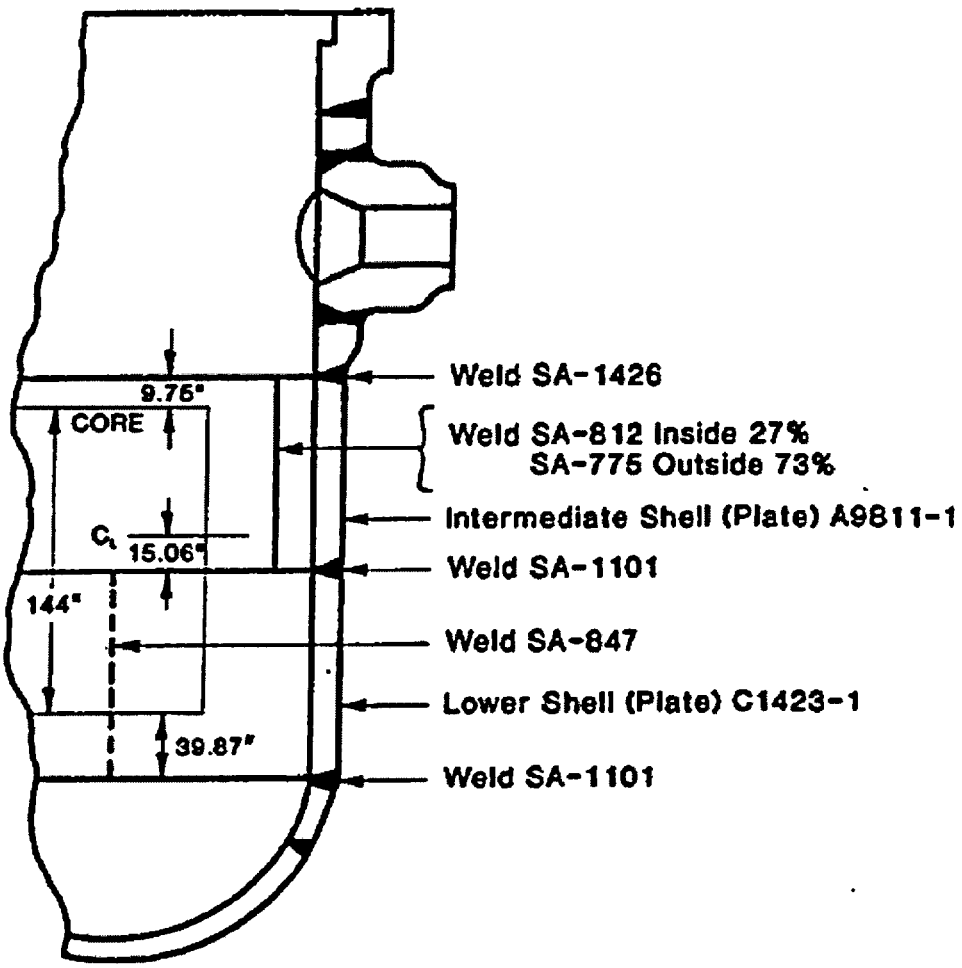

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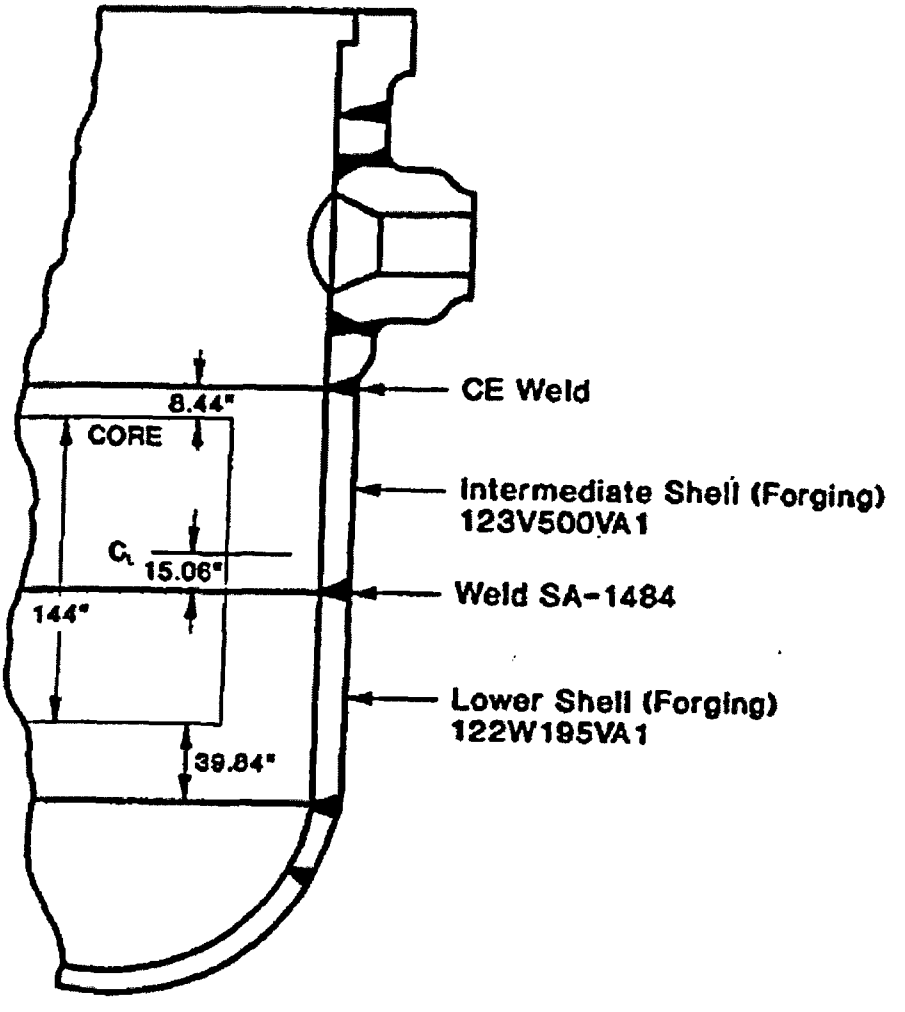
Figure 2-2 Reactor Vessel of Point Beach Unit 2



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3.0 Material Properties and Reactor Vessel Design Data

An upper-shelf fracture toughness material model is discussed below, as well as mechanical properties for the weld material and reactor vessel design data.

3.1 J-Integral Resistance Model for Mn-Mo-Ni/Linde 80 Welds

A model for the *J*-integral resistance versus crack extension curve (*J-R* curve) required to analyze low upper-shelf energy materials has been derived specifically for Mn-Mo-Ni/Linde 80 weld materials. A previous analysis of the reactor vessels of B&W Owners Group RVWG [1] described the development of this toughness model from a large data base of fracture specimens. A lower bound ($-2S_e$) *J-R* curve is obtained by multiplying *J*-integrals from the mean *J-R* curve by 0.699 [1]. It was shown in a previous low upper-shelf toughness analysis performed for B&W Owners Group plants [6] that a typical lower bound *J-R* curve is a conservative representation of toughness values for reactor vessel beltline materials, as required by Appendix K [4] for Levels A, B, and C Service Loadings. The best estimate representation of toughness required for Level D Service Loadings is provided by the mean *J-R* curve [7].

3.2 Reactor Vessel Design Data

Pertinent design data for upper-shelf flaw evaluations in the beltline region of the reactor vessel are provided below for Point Beach Units 1 and 2.

Design Pressure, P_d	= 2485 psig [2] (use 2500 psig)
Inside radius, R_i	= 66 in. [2]
Vessel thickness, t	= 6.5 in. [2]
Nominal cladding thickness, t_c	= 0.1875 in. [2]

3.3 Mechanical Properties for Weld Material

Mechanical properties for the base and weld materials are presented in Tables 3-1 through 3-3. The reactor vessel base metal at Point Beach Unit 1 is SA-302, Grade B low alloy steel, and at Point Beach Unit 2 is SA-508, Grade 2, Class 1 low alloy steel [8]. Base metal properties are found in the ASME Code [9]. Weld metal tensile properties are taken from appropriate surveillance capsule data of each weld material. The ASME transition region fracture toughness curve for K_{Ic}, used to define the beginning of the upper-shelf toughness region, is indexed by the initial RT_{NDT} of the weld material. Also, Poisson's ratio, ν, is taken to be 0.3.

3.3.1 Axial Weld SA-847

Table 3-1 Mechanical Properties for SA-847 Weld of Point Beach Unit 1

Temp.	E	Yield Strength (σ_y)		Ultimate Strength (σ_u)*		α
Material:	Base Metal	Base Metal	Weld SA-847	Base Metal	Weld SA-847	Base Metal
Source: [Ref.]	Code [9]	Code [9]	Actual [10]	Code [9]	Actual [10]	Code [9]
(°F)	(ksi)	(ksi)	(ksi)	(ksi)	(ksi)	(in/in/°F)
100	29200	50.00	95.00	80	99.8	7.06E-06
200	28500	47.50	89.60	80	99.8	7.25E-06
300	28000	46.10	86.01	80	99.8	7.43E-06
335	27790	45.74	85.10	80	97.6	7.48E-06
400	27400	45.10	84.77	80	99.8	7.58E-06
500	27000	44.50	84.26	80	99.8	7.70E-06
541.4	26751.6	44.16	84.04	80	99.8	7.75E-06
544.5	26733	44.14	84.03	80	99.8	7.76E-06
550	26700	44.11	84.00	80	99.8	7.77E-06
600	26400	43.80	83.74	80	99.8	7.83E-06

* Note: The ultimate strength values of the base and weld metals given here are not used in calculations

Initial RT_{NDT} = -5.0°F [5]
Margin = 48.3°F [5]

3.3.2 Circumferential Weld SA-1011

Table 3-2 Mechanical Properties for SA-1101 Weld of Point Beach Unit 1

Temp.	E	Yield Strength (σ_y)		Ultimate Strength (σ_u)*		α
Material:	Base Metal	Base Metal	Weld SA-1101	Base Metal	Weld SA-1101	Base Metal
Source: [Ref.]	Code [9]	Code [9]	Actual [11]	Code [9]	Actual [11]	Code [9]
(°F)	(ksi)	(ksi)	(ksi)	(ksi)	(ksi)	(in/in/°F)
100	29200	50.00	93.66	80	105.10	7.06E-06
200	28500	47.50	92.20	80	104.90	7.25E-06
300	28000	46.10	90.74	80	104.70	7.43E-06
400	27400	45.10	89.29	80	104.50	7.58E-06
500	27000	44.50	87.83	80	104.30	7.70E-06
541.4	26751.6	44.14	87.23	80	104.21	7.76E-06
544.5	26733	44.14	87.18	80	104.21	7.76E-06
550	26700	44.11	87.10	80	104.20	7.77E-06
600	26400	43.80	86.37	80	104.10	7.83E-06

* Note: The ultimate strength values of the base and weld metals given here are not used in calculations

Initial RT_{NDT} = 10.0°F [5]
Margin = 56.0°F [5]

3.3.3 Circumferential Weld SA-1484

Table 3-3 Mechanical Properties for SA-1484 Weld of Point Beach Unit 2

Temp.	E	Yield Strength (σ_y)		Ultimate Strength (σ_u)*		α
Material:	Base Metal	Base Metal	Weld SA-1484	Base Metal	Weld SA-1484	Base Metal
Source: [Ref.]	Code [9]	Code [9]	Actual [12]	Code [9]	Actual [12]	Code [9]
(°F)	(ksi)	(ksi)	(ksi)	(ksi)	(ksi)	(in/in/°F)
100	27800	50.00	82.10	80	96.90	6.50E-06
200	27100	47.50	79.57	80	92.98	6.67E-06
300	26700	46.10	78.00	80	90.40	6.87E-06
400	26100	45.10	77.17	80	89.41	7.07E-06
450	25900	44.76	76.80	80	89.60	7.15E-06
500	25700	44.50	76.42	80	90.29	7.25E-06
541.4	25460	44.16	76.15	80	91.25	7.32E-06
544.5	25444	44.14	76.13	80	91.34	7.33E-06
580	25264	43.94	76.00	80	92.50	7.39E-06
600	25200	43.80	75.80	80	93.28	7.42E-06

* Note: The ultimate strength values of the base and weld metals given here are not used in calculations

Initial RT_{NDT} = -5.0°F [5]
Margin = 68.5°F [5]

4.0 Analytical Methodology

Upper-shelf toughness is evaluated through use of fracture mechanics analytical methods that utilize the acceptance criteria and evaluation procedures of Section XI, Appendix K [4], where applicable.

4.1 Procedure for Evaluating Levels A and B Service Loadings

The applied *J*-integral is calculated per Appendix K, paragraph K-4210 [4], using an effective flaw depth to account for small scale yielding at the crack tip, and evaluated per K-4220 for upper-shelf toughness and per K-4310 for flaw stability.

4.2 Procedure for Evaluating Levels C and D Service Loadings

Levels C and D Service Loadings are evaluated using the one-dimensional, finite element, thermal and stress models and linear elastic fracture mechanics methodology of the PCRIT computer code to determine stress intensity factors. The beltline region welds identified in Section 3.3 are analyzed for all Level C and D transients. Two Level D transients are specified for the Point Beach Units. The original equipment specification includes a Steam Line Break (SLB) transient and a Reactor Coolant Line Break (LOCA) transient. The Point Beach FSAR contains a Steam Line Break (two loops in service) without Offsite Power transient [13].

The transients considered appear in Figure 5.1. Transients are assumed to hold steady at the end of their definitions, and are held constant until the thermal gradient through the shell has developed fully and begins to dissipate.

The evaluation is performed as follows:

(1) For each transient described above, utilize PCRIT to calculate stress intensity factors for a semi-elliptical flaw of depth $^1/_{10}$ of the base metal wall thickness, as a function of time, due to internal pressure and radial thermal gradients with a factor of safety of 1.0 on loading. The applied stress intensity factor, K_I, calculated by PCRIT for each of these transients is compared to the K_{Jc} limit of the weld. The transient that most closely approaches the K_{Jc} limit is chosen as the limiting transient, and the critical time in the limiting transient occurs at the point where K_I most closely approaches the upper-shelf toughness curve.

(2) At the critical transient time, develop a crack driving force diagram with the applied *J*-integral and *J-R* curves plotted as a function of flaw extension. The adequacy of the upper-shelf toughness is evaluated by comparing the applied *J*-integral with the *J-R* curve at a flaw extension of 0.10 in. Flaw stability is assessed by examining the slopes of the applied *J*-integral and *J-R* curves at the points of intersection.

(3) Verify that the extent of stable flaw extension is no greater than 75% of the vessel wall thickness by determining when the applied *J*-integral curve intersects the mean *J-R* curve.

(4) Verify that the remaining ligament is not subject to tensile instability. The internal pressure p shall be less than P_l, where P_l is the internal pressure at tensile instability of the remaining ligament. Equations for P_l are given below for the axial and circumferential flaws [14]. These equations first appear in the 2001 Edition of the ASME Section XI code that is cited.

(a) For an axial flaw,

$$P_l = 1.07\sigma_O\left[\frac{1-(A_c/A)}{(R_i/t)+(A_c/A)}\right] \qquad \text{[eqn. 1]}$$

where

$$\sigma_O = \frac{\sigma_y + \sigma_u}{2} \qquad \text{[eqn. 2]}$$

$$A = t(\ell + t) \qquad \text{[eqn. 3]}$$

$$A_c = \frac{\pi a \ell}{4} \qquad \text{[eqn. 4]}$$

and

ℓ = surface length of crack, six times the depth, a
R_m = mean radius of vessel

This equation for P_l includes the effect of pressure on the flaw face.

(b) For a circumferential flaw,

$$P_l = 1.07\sigma_O\left[\frac{1-(A_c/A)}{(R_i^2/(2R_m t))+(A_c/A)}\right] \qquad \text{[eqn. 5]}$$

where σ_O, A, and A_c are given by equations 2, 3 and 4, respectively.

This equation for P_l includes the effect of pressure on the flaw face. This equation is valid for internal pressures not exceeding the pressure at tensile instability caused by the applied hoop stress acting over the nominal wall thickness of the vessel. This validity limit on pressure for the circumferential flaw equation for P_l is

$$P_l \le 1.07\sigma_O\left[\frac{t}{R_i}\right] \qquad \text{[eqn. 6]}$$

4.3 Temperature Range for Upper-Shelf Fracture Toughness Evaluations

Upper-shelf fracture toughness is determined through use of Charpy V-notch impact energy versus temperature plots by noting the temperature above which the Charpy energy remains on a plateau, maintaining a relatively high constant energy level. Similarly, fracture toughness can be addressed in three different regions on the temperature scale, i.e. a lower-shelf toughness region, a transition region, and an upper-shelf toughness region. Fracture toughness of reactor vessel steel and associated weld metals are conservatively predicted by the ASME initiation toughness curve, K_{Ic}, in the lower-shelf and transition regions. In the upper-shelf region, the upper-shelf toughness curve, K_{Jc}, is derived from the upper-shelf *J*-integral resistance model described in Section 3.1. The upper-shelf toughness then becomes a function of fluence, copper content, temperature, and fracture specimen size. When upper-shelf toughness is plotted versus temperature, a plateau-like curve develops that decreases slightly with increasing temperature. Since the present analysis addresses the low upper-shelf toughness issue, only the upper-shelf temperature range, which begins at the intersection of K_{Ic} and the upper-shelf toughness curves, K_{Jc}, is considered.

4.4 Effect of Cladding Material

The PCRIT code utilized in the flaw evaluations for Levels C and D Service Loadings does not consider stresses in the cladding when calculating stress intensity factors for thermal loads. To account for this cladding effect, an additional stress intensity factor, K_{Icclad}, is calculated separately and added to the total stress intensity factor computed by PCRIT.

The contribution of cladding stresses to stress intensity factor was examined previously [2]. In this low upper-shelf toughness analysis performed for B&W Owners Group Reactor Vessel Working Group plants, the Zion-1 WF-70 weld using thermal loads from the Turkey Point SLB was determined to be the bounding case. The Zion-1 vessel was as thick as or thicker than any other vessel. The thicknesses of the reactor vessels for the both Point Beach units are 6.5" whereas the Zion vessel is 8.44". The nominal cladding thickness is 3/16" for both vessels. From a thermal stress perspective, it is conservative to consider the thicker vessel. For the Zion vessel, the maximum value of K_{Icclad}, at any time during the transient and for any flaw depth, was determined to be 9.0 ksi$\sqrt{\text{in}}$. This bounding value is therefore used as the stress intensity factor for K_{Icclad} in this Point Beach low upper-shelf toughness analysis.

5.0 Applied Loads

The Levels A and B Service Loadings required by Appendix K are an accumulation pressure (internal pressure load) and a cooldown rate (thermal load). Since Levels C and D Service Loadings are not specified by the Code, Levels C and D pressurized thermal shock events are reviewed and a worst case transient is selected for use in flaw evaluations.

5.1 Levels A and B Service Loadings

Per paragraph K-1300 of Appendix K [4], the accumulation pressure used for flaw evaluations should not exceed 1.1 times the design pressure. Using 2.5 ksi as the design pressure, the accumulation pressure is 2.75 ksi. The cooldown rate is also taken to be the maximum required by Appendix K, 100°F/hour.

5.2 Levels C and D Service Loadings

As discussed in Section 4.2, the SLB and LOCA transients are evaluated using the computer code PCRIT. Pressure and temperature time histories for the two transients considered are shown in Figure 5-1.

Figure 5-1 Level D transients – Reactor Coolant Temperature and Pressure vs. Time

6.0 Evaluation for Levels A and B Service Loadings

The material mean and lower bounding *J-R* values for Evaluation Conditions 1, 2 and 3 detailed in Table 2-1 are given in Tables 6-1 through 6-3, respectively. Initial flaw depths equal to $^1/_4$ of the vessel wall thickness are analyzed for Levels A and B Service Loadings following the procedure outlined in Section 4.1 and evaluated for acceptance based on values for the *J*-integral resistance of the materials from Section 3.3. The results of the evaluation are presented in Table 6-4 through 6-6, where it is seen that the minimum ratio of material *J*-integral resistance ($J_{0.1}$) to applied *J*-integral (J_1) is 1.87 for the SA-847 axial weld for Evaluation Condition 1, uprated power conditions without hafnium power suppression assemblies. This ratio is higher than the minimum acceptable value of 1.0. Also included in Table 6-4 through 6-6 is the applied *J*-integral at ($J_{0.1}$) with a safety factor on pressure of 1.25.

The flaw evaluation for the controlling weld (SA-847) and controlling Evaluation Condition (1) is repeated by calculating applied *J*-integrals for various amounts of flaw extension with safety factors (on pressure) of 1.15 and 1.25. The results, along with mean and lower bound *J-R* curves, are plotted in Figure 6-1. The requirement for ductile and stable crack growth is also demonstrated by Figure 6-1 since the slope of the applied *J*-integral curve for a safety factor of 1.25 is considerably less than the slope of the lower bound *J-R* curve at the point where the two curves intersect.

Table 6-1 Material *J*-Integral Resistance for Levels A and B Service Loadings – Evaluation Condition 1 – Uprated Power Conditions Without Hafnium Assemblies

Plant	Cold Leg Temp. (°F)	Controlling Weld			Fluence $\times 10^{18}$ (n/cm^2)		*J-R* at Δa = 0.1 in.	
		Material ID	Weld Orientation	Cu Content (wt%)	at I.S.	at t/4	Mean (lb/in)	Lower Bound at -2Se (lb/in)
PB-1	541.4	SA-847	L	0.23	33.70	22.28	885	618
PB-1	541.4	SA-1101	C	0.23	49.10	32.46	870	608
PB-2	541.4	SA-1484	C	0.26	50.90	33.64	826	578

Table 6-2 Material *J*-Integral Resistance for Levels A and B Service Loadings – Evaluation Condition 2 – Current Power Conditions Without Hafnium Assemblies

Plant	Cold Leg Temp. (°F)	Controlling Weld			Fluence $\times 10^{18}$ (n/cm^2)		*J-R* at Δa = 0.1 in.	
		Material ID	Weld Orientation	Cu Content (wt%)	at I.S.	at t/4	Mean (lb/in)	Lower Bound at -2Se (lb/in)
PB-1	544.5	SA-847	L	0.23	31.15	20.59	885	618
PB-1	544.5	SA-1101	C	0.23	45.20	29.88	870	608
PB-2	544.5	SA-1484	C	0.26	46.45	30.70	827	578

Table 6-3 Material *J*-Integral Resistance for Levels A and B Service Loadings – Evaluation Condition 3 – Current Power Conditions With Hafnium Assemblies

Plant	Cold Leg Temp. (°F)	Controlling Weld			Fluence $\times 10^{18}$ (n/cm^2)		*J-R* at Δa = 0.1 in.	
		Material ID	Weld Orientation	Cu Content (wt%)	at I.S.	at t/4	Mean (lb/in)	Lower Bound at -2Se (lb/in)
PB-1	544.5	SA-847	L	0.23	26.65	17.62	891	623
PB-1	544.5	SA-1101	C	0.23	38.20	25.25	877	613
PB-2	544.5	SA-1484	C	0.26	37.85	25.02	836	585

Table 6-4 Flaw Evaluation for Levels A and B Service Loadings – Evaluation Condition 1 – Uprated Power Conditions Without Hafnium Assemblies

Plant	Weld Number	Weld Orientation	Lower Bounding $J_{0.1}$ at t/4 (lb/in)	SF = 1.15		SF = 1.25	
				J_1 (lb/in)	$J_{0.1}/J_1$	J_1 (lb/in)	$J_{0.1}/J_1$
PB-1	SA-847	L	618	331	1.87	388	1.59
PB-1	SA-1101	C	608	98	6.20	113	5.38
PB-2	SA-1484	C	578	104	5.56	119	4.86

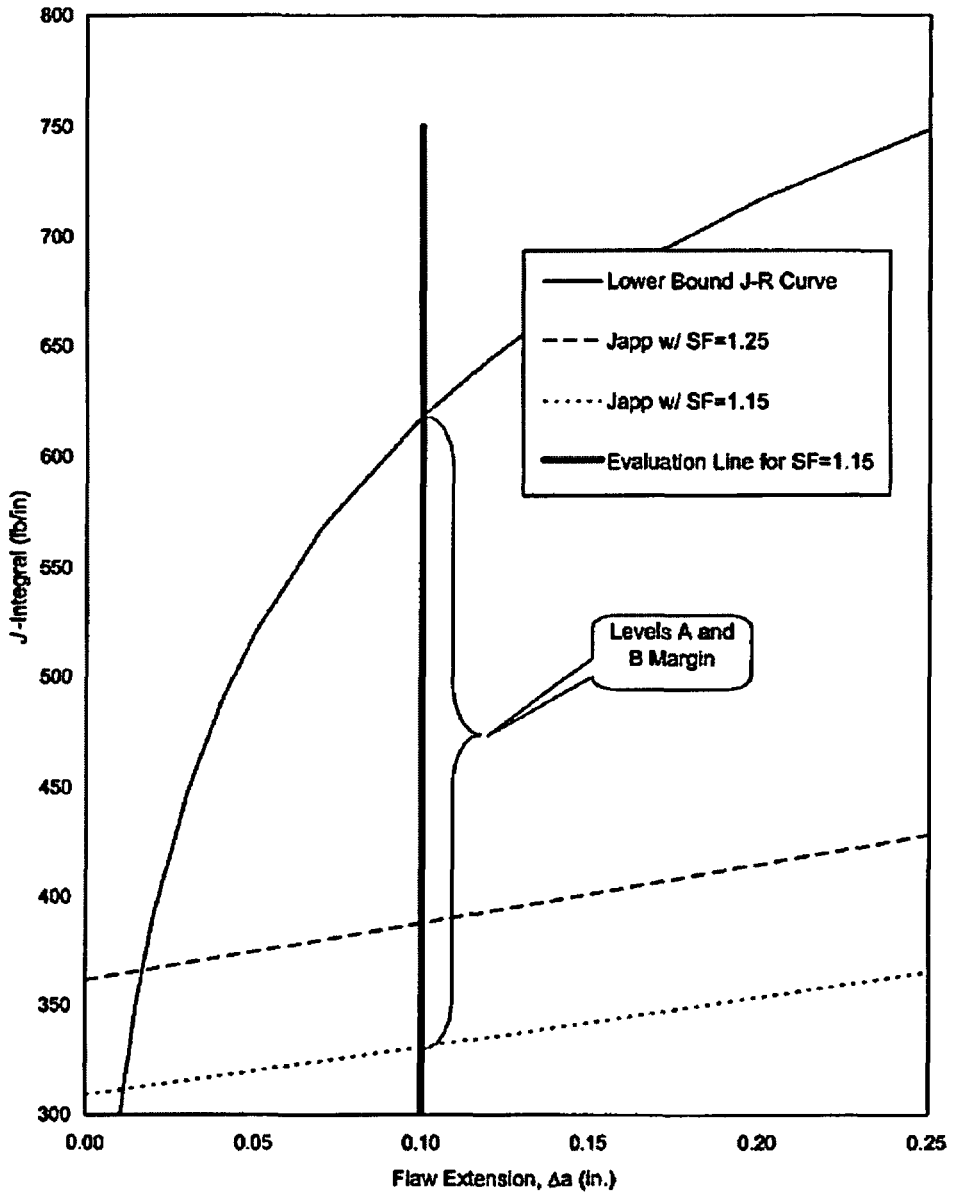
Table 6-5 Flaw Evaluation for Levels A and B Service Loadings – Evaluation Condition 2 – Current Power Conditions Without Hafnium Assemblies

Plant	Weld Number	Weld Orientation	Lower Bounding $J_{0.1}$ at t/4 (lb/in)	SF = 1.15		SF = 1.25	
				J_1 (lb/in)	$J_{0.1}/J_1$	J_1 (lb/in)	$J_{0.1}/J_1$
PB-1	SA-847	L	618	331	1.87	388	1.59
PB-1	SA-1101	C	608	98	6.20	113	5.38
PB-2	SA-1484	C	578	104	5.56	119	4.86

Table 6-6 Flaw Evaluation for Levels A and B Service Loadings – Evaluation Condition 3 – Current Power Conditions With Hafnium Assemblies

Plant	Weld Number	Weld Orientation	Lower Bounding $J_{0.1}$ at t/4 (lb/in)	SF = 1.15		SF = 1.25	
				J_1 (lb/in)	$J_{0.1}/J_1$	J_1 (lb/in)	$J_{0.1}/J_1$
PB-1	SA-847	L	623	331	1.88	388	1.61
PB-1	SA-1101	C	613	98	6.26	113	5.42
PB-2	SA-1484	C	585	104	5.63	119	4.92

Figure 6-1 J-Integral vs. Flaw Extension for Levels A & B Service Loadings - Evaluation Condition 1 - Uprated Power Conditions Without Hafnium Assemblies - Weld SA-847



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7.0 Evaluation for Levels C and D Service Loadings

A flaw depth of $^1/_{10}$ of the base metal wall thickness, plus the cladding thickness, is used to evaluate the Level D Service Loadings. The stress intensity factor K_I calculated by the PCRIT code is the sum of thermal, residual stress, deadweight, and pressure terms. PCRIT is run for each Level D transient. RT_{NDT} is also calculated by PCRIT. Transition region toughness is obtained from the ASME Section XI equation for crack initiation [15].

$$K_{Ic} = 33.2 + 2.806 \exp[0.02(T - RT_{NDT} + 100°F)] \qquad \text{[eqn. 7]}$$

where:

K_{Ic} = transition region toughness, ksi√in
T = crack tip temperature, °F

Upper-shelf toughness is derived from the J-integral resistance model of Section 3.1 for a flaw depth of $^1/_{10}$ of the wall thickness, a crack extension of 0.10 in., and fluence, as follows:

$$K_{Jc} = \sqrt{\frac{J_{0.1}E}{1000(1-\nu^2)}} \qquad \text{[eqn. 8]}$$

where

K_{Jc} = upper-shelf region toughness, ksi√in
$J_{0.1}$ = J-integral resistance at Δa = 0.1 in.

Figure 7-1 through 7-3 shows the variation of applied stress intensity factor, K_I, transition range toughness, K_{Ic}, and upper-shelf toughness, K_{Jc} with temperature for the Evaluation Condition 1 described in Table 2-1 for the three welds. The markers on the K_I curve indicate points in time at which PCRIT solutions are available. For all the three welds that were analyzed, the LOCA transient is limiting since it most closely approaches the K_{Jc} limit of each weld. All subsequent analysis will pertain to this transient. In the upper-shelf toughness range, the K_I curve is closest to the lower bound K_{Jc} curve at a particular time point into the transient for each weld, as listed below:

Weld	Time (min)
SA-847	2.40
SA-1011	1.50
SA-1484	1.30

For each weld, the time specified above is selected as the critical time in the transient at which to perform the flaw evaluation for Level D Service Loadings.

Figure 7-1 K_I vs. Crack Tip Temperature for Evaluation Condition 1 - SA-847

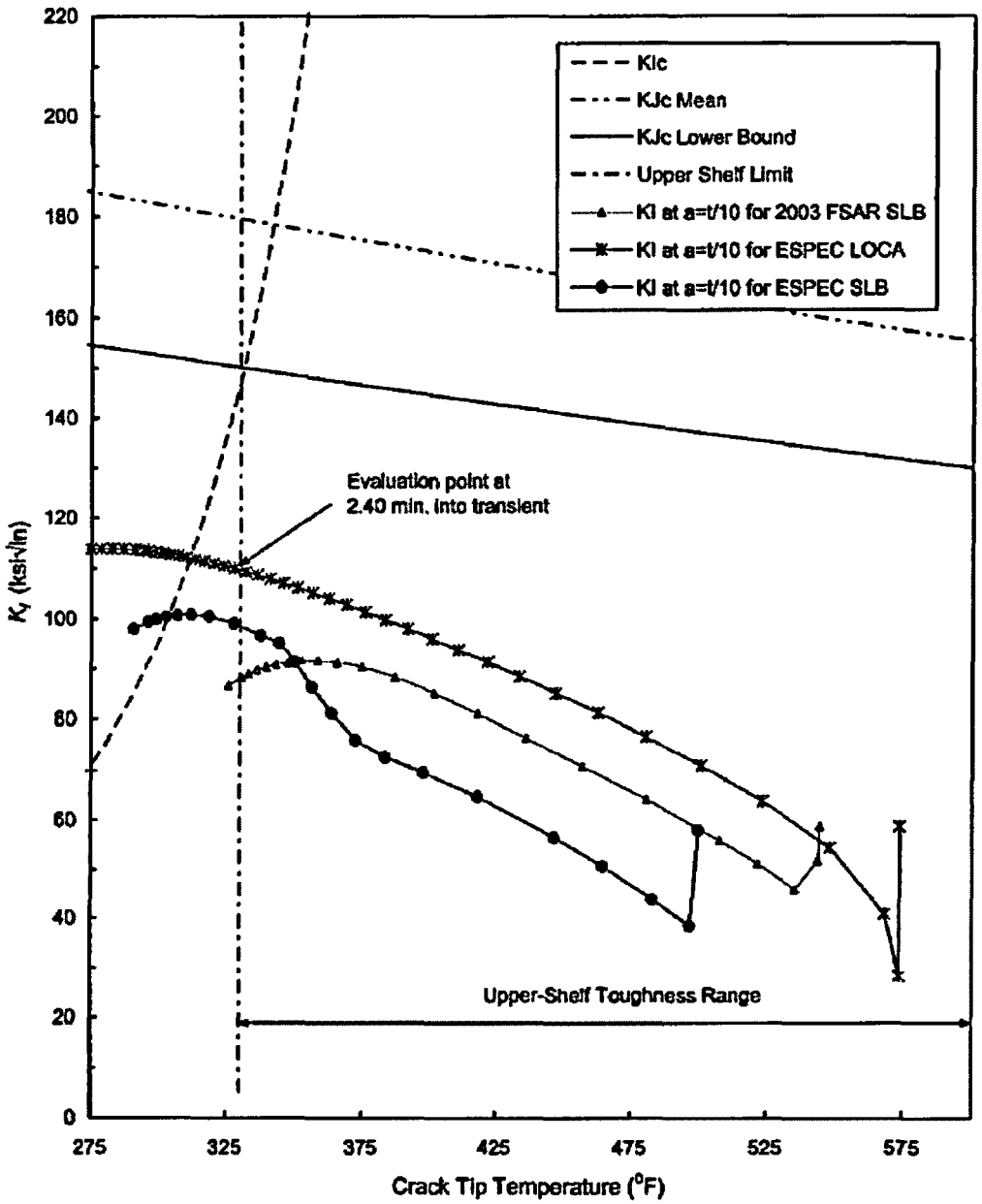

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Figure 7-2 K_I vs. Crack Tip Temperature for Evaluation Condition 1 - SA-1101

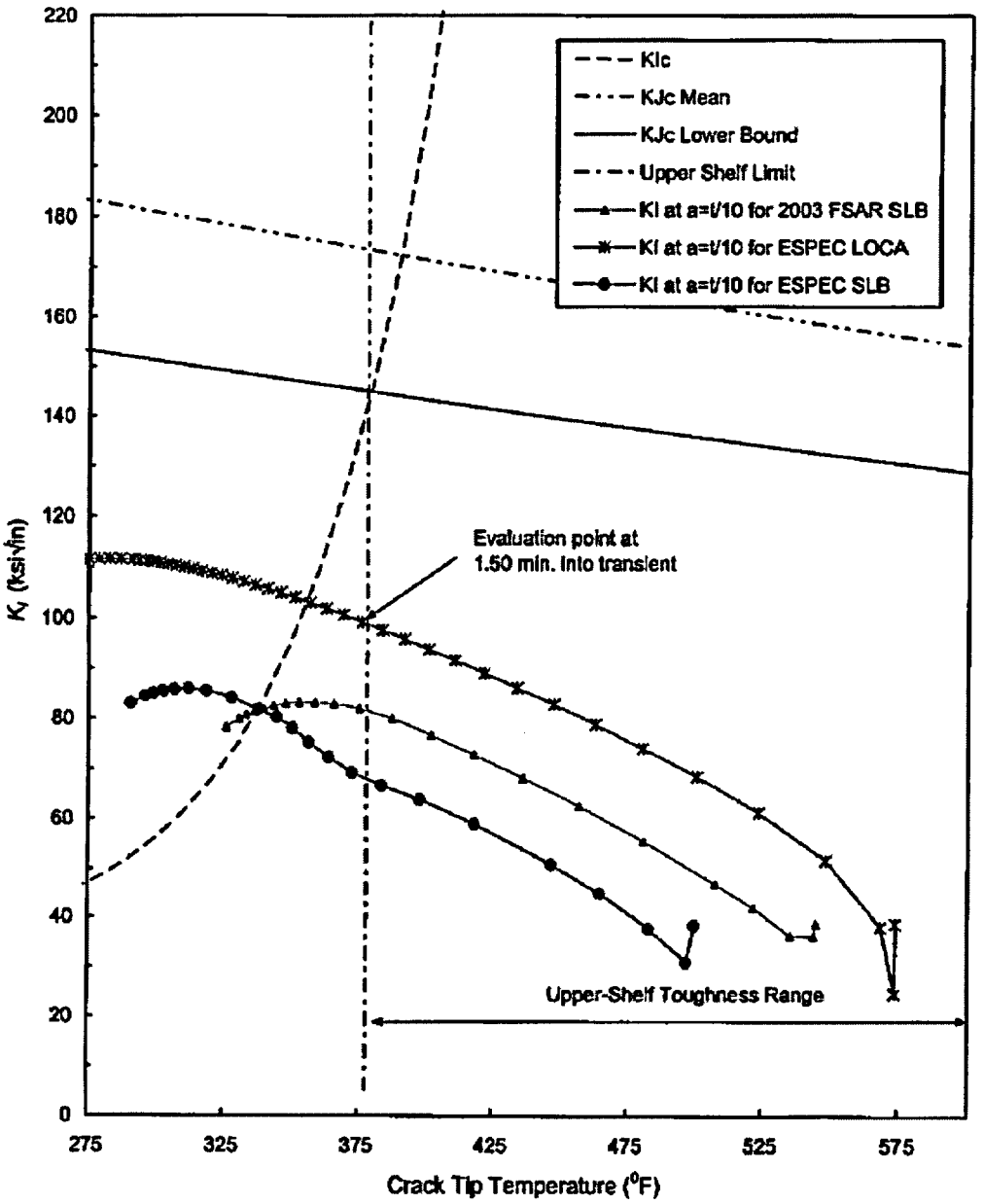

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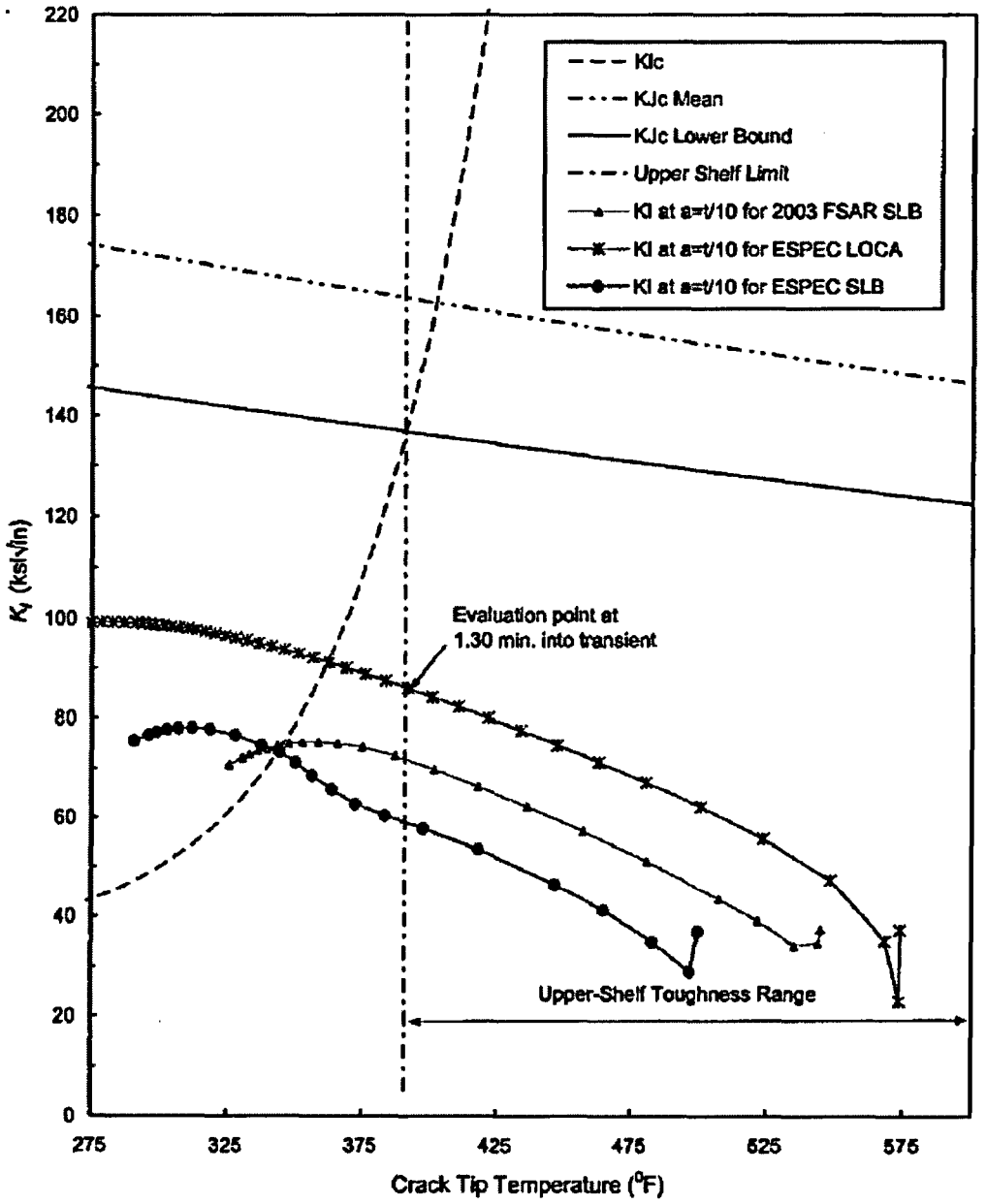
Figure 7-3 K_I vs. Crack Tip Temperature for Evaluation Condition 1 - SA-1484



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Applied *J*-integrals for the LOCA transient are calculated for each weld at the critical time points identified above for various flaw depths in Table 7-1, 7-2, and 7-3 using stress intensity factors from PCRIT and adding 9.0 ksi√in to account for cladding effects. Stress intensity factors are converted to *J*-integrals by the plain strain relationship,

$$J_{applied}(a) = 1000\frac{K^2_{Itotal}(a)}{E}(1-\nu^2) \qquad \text{[eqn. 9]}$$

Tables 7-1, 7-2, and 7-3 lists flaw extensions vs. applied *J*-integrals. As the Point Beach vessels are 6.5 in. thick, the initial flaw depth of $^1/_{10}$ of the wall thickness is 0.65 in. Flaw extension from this flaw depth is calculated by subtracting 0.65 in. from the built-in PCRIT flaw depths in the base metal. The results, along with mean *J-R* curve, are plotted in Figure 7-4. This figure indicates that Weld SA-847 is limiting as the ratio of the applied *J*-integral to the material *J-R* curve is less than the other two welds. Figure 7-5 is a plot of the applied *J*-integrals and the mean *J-R* curves for the three Evaluation Conditions from Table 2-1 for Weld SA-847. Evaluation Condition 1, uprated power conditions without hafnium power suppression assemblies, is the limiting case as the ratio of the mean *J-R* curves to applied *J*-integrals is the minimum of the three Evaluation Conditions. The requirements for ductile and stable crack growth are demonstrated by Figure 7-5 since the slopes of the applied *J*-integral curves are considerably less than the slopes mean *J-R* curves at the points of intersection. The Level D Service Loading requirement that the extent of stable flaw extension be no greater than 75% of the vessel wall thickness is easily satisfied since the applied *J*-integral curves intersects the mean *J-R* curves at flaw extensions that are only a small fraction of the wall thickness (less than 1%).

The last requirement is that the internal pressure p shall be less than P_t, the internal pressure at tensile instability of the remaining ligament. Table 7-4 gives the results of the calculations for P_t for flaw depths up to 1.365 inches for Evaluation Condition 1. As the internal pressure p is less than P_t, the remaining ligament is not subject to tensile instability.

Table 7-1 J-Integral vs. Flaw Extension for Evaluation Condition 1 - SA-847

Time =	2.40 min				E =	26751.6	ksi
Crack tip at t/10		t =	6.5	in.	v =	0.3	
$(a^{++}/t)*40$	a^{++} (in.)	Δa (in.)	Temp. (F)	K_{Isum}	K_{Iclad}	K_{Itotal}	J_{app} (lb/in)
1	0.1625		246.40	62.06	9.0	71.1	172
2	0.3250		274.80	83.65	9.0	92.7	292
3	0.4875		302.10	94.64	9.0	103.6	365
4	0.6500	0.0000	328.00	100.97	9.0	110.0	411
5	0.8125	0.1625	352.70	104.24	9.0	113.2	436
6	0.9750	0.3250	375.90	105.82	9.0	114.8	448
7	1.1375	0.4875	397.70	106.12	9.0	115.1	451
8	1.3000	0.6500	417.90	105.76	9.0	114.8	448
9	1.4625	0.8125	436.50	104.86	9.0	113.9	441
10	1.6250	0.9750	453.60	103.22	9.0	112.2	428
12	1.9500	1.3000	483.10	98.74	9.0	107.7	395
14	2.2750	1.6250	507.00	93.05	9.0	102.1	354
16	2.6000	1.9500	525.80	88.28	9.0	97.3	322
18	2.9250	2.2750	540.10	82.87	9.0	91.9	287
20	3.2500	2.6000	550.70	77.27	9.0	86.3	253
22	3.5750	2.9250	558.40	71.71	9.0	80.7	222
24	3.9000	3.2500	563.90	66.53	9.0	75.5	194
26	4.2250	3.5750	567.60	61.81	9.0	70.8	171
28	4.5500	3.9000	570.00	57.20	9.0	66.2	149
30	4.8750	4.2250	571.60	52.58	9.0	61.6	129
32	5.2000	4.5500	572.60	48.13	9.0	57.1	111

Note: a^{++} is the flaw depth in the base metal

Table 7-2 J-Integral vs. Flaw Extension for Evaluation Condition 1 - SA-1101

Time =	1.50 min				E =	26751.6	ksi
Crack tip at t/10		t =	6.5	in.	v =	0.3	
$(a^{++}/t)*40$	a^{++} (in.)	Δa (in.)	Temp. (F)	K_{Isum}	K_{Iclad}	K_{Itotal}	J_{app} (lb/in)
1	0.1625		280.80	59.65	9.0	68.7	160
2	0.3250		314.80	78.57	9.0	87.6	261
3	0.4875		346.70	86.65	9.0	95.7	311
4	0.6500	0.0000	376.30	90.22	9.0	99.2	335
5	0.8125	0.1625	403.60	91.26	9.0	100.3	342
6	0.9750	0.3250	428.40	90.74	9.0	99.7	338
7	1.1375	0.4875	450.60	89.06	9.0	98.1	327
8	1.3000	0.6500	470.50	86.71	9.0	95.7	312
9	1.4625	0.8125	488.00	83.66	9.0	92.7	292
10	1.6250	0.9750	503.10	80.42	9.0	89.4	272
12	1.9500	1.3000	527.20	72.98	9.0	82.0	229
14	2.2750	1.6250	544.30	65.06	9.0	74.1	187
16	2.6000	1.9500	555.90	57.27	9.0	66.3	149
18	2.9250	2.2750	563.40	49.24	9.0	58.2	115
20	3.2500	2.6000	568.10	41.31	9.0	50.3	86
22	3.5750	2.9250	570.90	34.09	9.0	43.1	63
24	3.9000	3.2500	572.40	27.47	9.0	36.5	45
26	4.2250	3.5750	573.30	21.94	9.0	30.9	33
28	4.5500	3.9000	573.70	17.63	9.0	26.6	24
30	4.8750	4.2250	573.90	14.36	9.0	23.4	19
32	5.2000	4.5500	574.00	11.59	9.0	20.6	14

Note: a^{++} is the flaw depth in the base metal

Table 7-3 J-Integral vs. Flaw Extension for Evaluation Condition 1 - SA-1484

Time =	1.30 min				E =	25459.9	ksi
Crack tip at t/10		t =	6.5	in.	ν =	0.3	
$(a^{++}/t)*40$	a^{++} (in.)	Δa (in.)	Temp. (F)	K_{Isum}	K_{Iclad}	K_{Itotal}	J_{app} (lb/in)
1	0.1625		292.60	51.19	9.0	60.2	129
2	0.3250		328.30	67.16	9.0	76.2	207
3	0.4875		361.60	73.97	9.0	83.0	246
4	0.6500	0.0000	392.10	76.91	9.0	85.9	264
5	0.8125	0.1625	419.80	77.72	9.0	86.7	269
6	0.9750	0.3250	444.70	77.16	9.0	86.2	265
7	1.1375	0.4875	466.60	75.59	9.0	84.6	256
8	1.3000	0.6500	485.80	73.43	9.0	82.4	243
9	1.4625	0.8125	502.50	70.67	9.0	79.7	227
10	1.6250	0.9750	516.40	67.71	9.0	76.7	210
12	1.9500	1.3000	538.10	61.07	9.0	70.1	175
14	2.2750	1.6250	552.60	54.04	9.0	63.0	142
16	2.6000	1.9500	561.80	47.18	9.0	56.2	113
18	2.9250	2.2750	567.40	40.21	9.0	49.2	87
20	3.2500	2.6000	570.60	33.42	9.0	42.4	64
22	3.5750	2.9250	572.40	27.38	9.0	36.4	47
24	3.9000	3.2500	573.30	21.99	9.0	31.0	34
26	4.2250	3.5750	573.80	17.69	9.0	26.7	25
28	4.5500	3.9000	574.00	14.53	9.0	23.5	20
30	4.8750	4.2250	574.00	12.34	9.0	21.3	16
32	5.2000	4.5500	574.10	10.58	9.0	19.6	14

Note: a^{++} is the flaw depth in the base metal

Table 7-4 Level D Service Loadings - Internal Pressure at Tensile Instability - SA-847

flaw depth *a* (in.)	P_i (ksi)
0.0650	9.18
0.1300	9.16
0.1950	9.14
0.2600	9.12
0.3250	9.09
0.3900	9.06
0.4550	9.02
0.5200	8.98
0.5850	8.93
0.6500	8.88
0.7150	8.84
0.7800	8.78
0.8450	8.73
0.9100	8.68
0.9750	8.62
1.0400	8.56
1.1050	8.51
1.1700	8.45
1.2350	8.39
1.3000	8.32
1.3650	8.26

Figure 7-4. J-Integral vs. Flaw Extension – All Welds

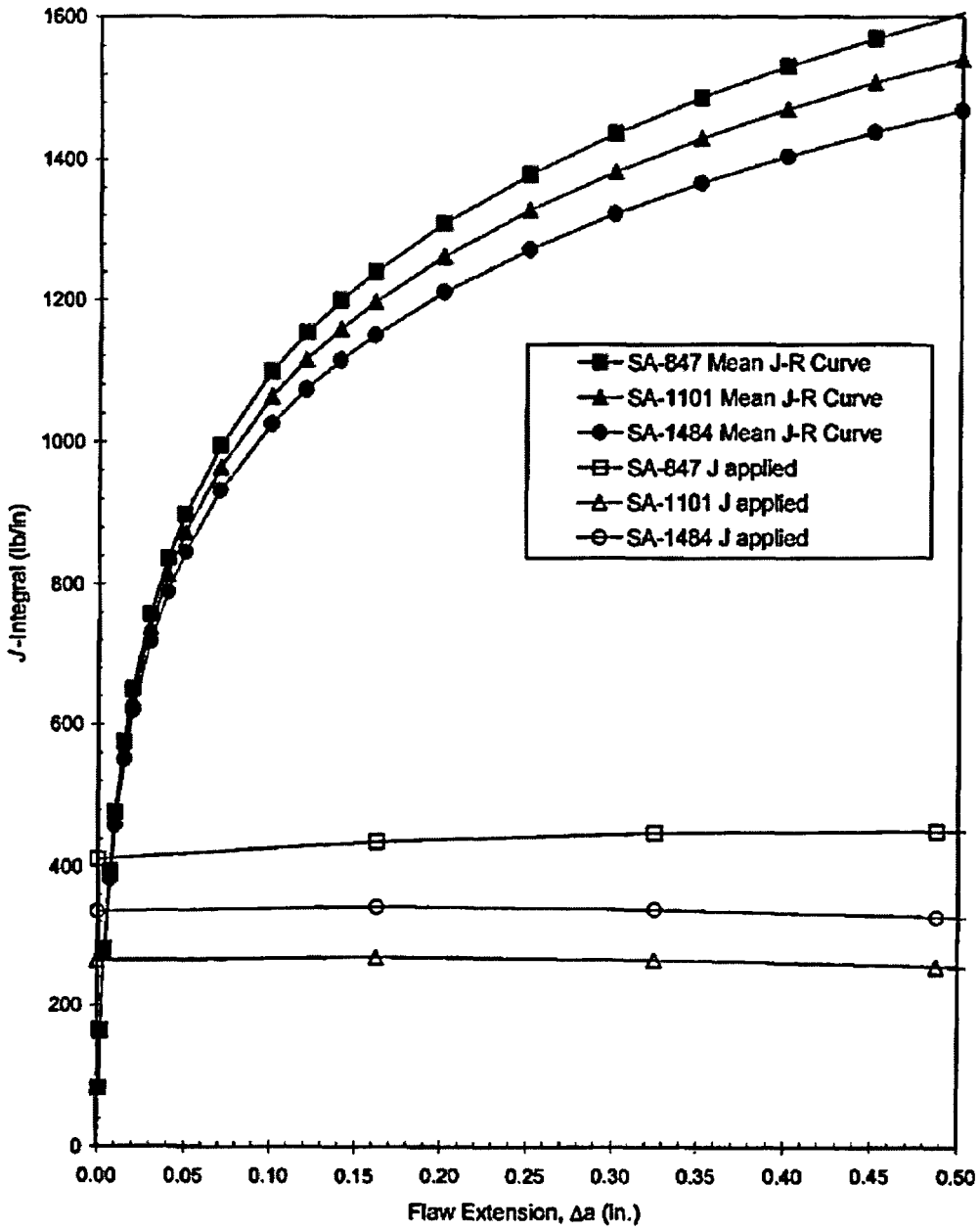

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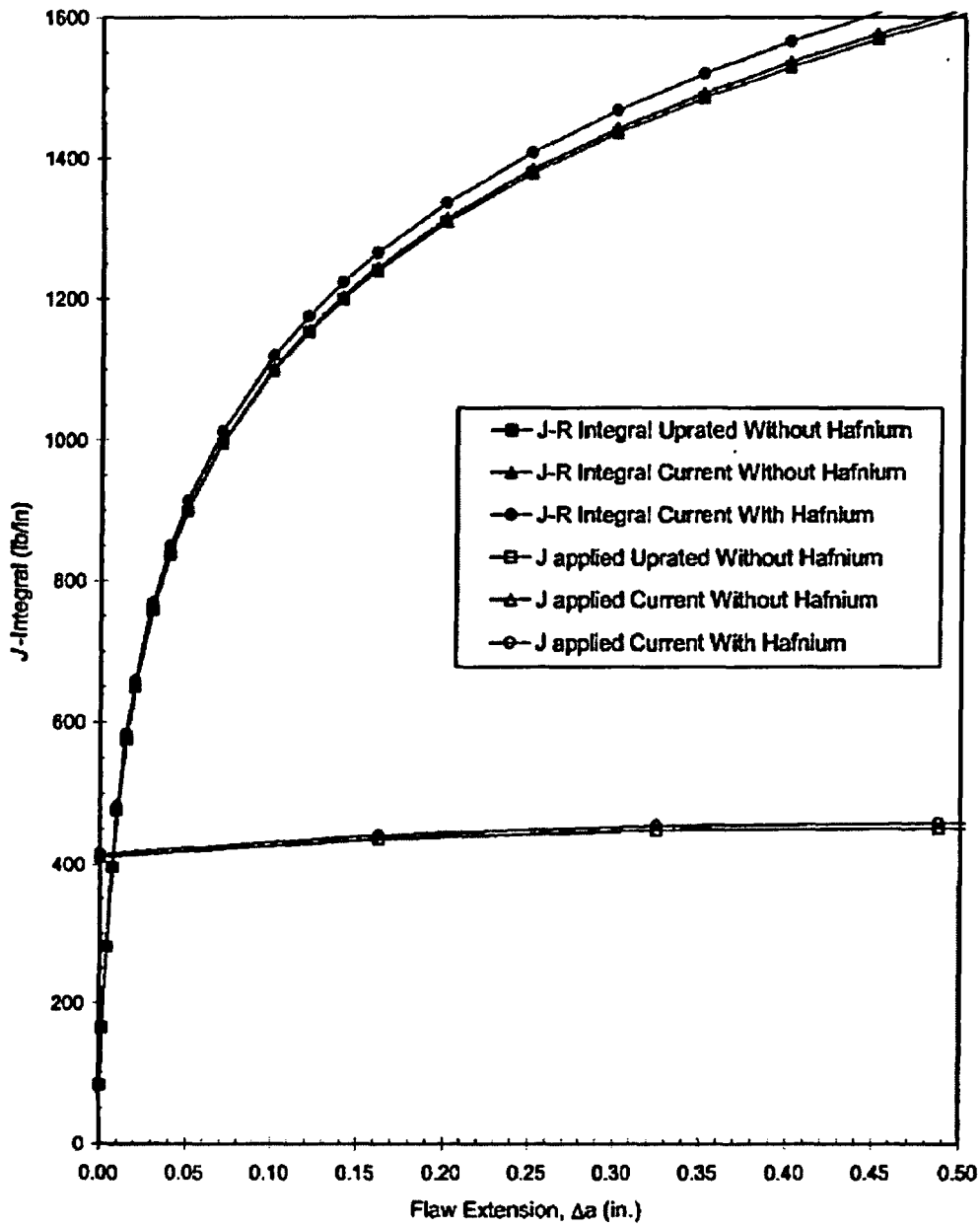
Figure 7-5. J-Integral vs. Flaw Extension – Weld SA-847



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8.0 Summary of Results

A low upper-shelf toughness fracture mechanics analysis has been performed to evaluate the reactor vessel welds at Point Beach Units 1 and 2 for projected low upper-shelf energy levels at 53 EFPY, considering Levels A, B, C, and D Service Loadings of the ASME Code.

Evidence that the ASME Code, Section XI, Appendix K [4] acceptance criteria have been satisfied for Levels A and B Service Loadings is provided by the following:

(1) The limiting weld is the SA-847 axial weld of Point Beach Unit 1 in the uprated power condition without hafnium power suppression assemblies. Figure 6-1 shows that with factors of safety of 1.15 on pressure and 1.0 on thermal loading, the applied *J*-integral (J_1) is less than the *J*-integral of the material at a ductile flaw extension of 0.10 in. ($J_{0.1}$). The ratio $J_{0.1}/J_1$ = 1.87 which is significantly greater than the required value of 1.0.

(2) Figure 6-1 shows that with a factor of safety of 1.25 on pressure and 1.0 on thermal loading, flaw extensions are ductile and stable since the slope of the applied *J*-integral curve is less than the slope of the lower bound *J-R* curve at the point where the two curves intersect.

Evidence that the ASME Code, Section XI, Appendix K [4] acceptance criteria have been satisfied for Level D Service Loadings is provided by the following:

(1) Figure 7-5 shows that with a factor of safety of 1.0 on loading, flaw extensions are ductile and stable since the slope of the applied *J*-integral curve is less than the slopes of both the lower bound and mean *J-R* curves at the points of intersection.

(3) Figure 7-5 shows that the flaw remains stable at much less than 75% of the vessel wall thickness. It has also been shown that the remaining ligament is sufficient to preclude tensile instability by a large margin.

9.0 Conclusion

The limiting Point Beach Units 1 and 2 reactor vessel beltline weld (axial weld SA-847 of Unit 1) satisfies the acceptance criteria of Appendix K to Section XI of the ASME Code [4] for projected low upper-shelf Charpy impact energy levels at 53 effective full power years of plant operation for the three conditions evaluated: uprated power conditions (1678 MWt) without hafnium power suppression assemblies, current power conditions (1540 MWt) without hafnium power suppression assemblies, and current power conditions (1540 MWt) with hafnium power suppression assemblies.

10.0 References

1. BAW-2192PA, Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group For Level A & B Service Loads, April 1994.

2. BAW-2178PA, Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group For Level C & D Service Loads, April 1994.

3. BAW-2255, Effect of Power Upgrade on Low Upper-Shelf Toughness Issue, May 1995.

4. ASME Boiler and Pressure Vessel Code, Section XI, 1998 Edition with Addenda through 2000.

5. USNRC Reactor Vessel Integrity Database Version 2.0.1 (RVID).

6. BAW-2275, Low Upper-Shelf Toughness Fracture Mechanics Analysis of B&W Designed Reactor Vessels for 48 EFPY, August 1996.

7. BAW-2312, Revision 1, Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of Turkey Point Units 3 and 4 for Extended Life through 48 Effective Full Power Years, December 2000.

8. BAW-2150, Materials Information for Westinghouse-Designed Reactor Vessels Fabricated by B&W, December 1990.

9. ASME Boiler and Pressure Vessel Code, Section III, Appendices, 1989 Edition with no Addenda.

10. WCAP-13902, Analysis of Capsule S from the Rochester Gas and Electric Corporation R. E. Ginna Reactor Vessel Radiation Surveillance Program, December 1993.

11. WCAP-15916, Analysis of Capsule X from the Florida Power and Light Turkey Point 3 Reactor Vessel Radiation Surveillance Program, September 2002.

12. BAW-2254, Test Results of Capsule CR3-LG2: B&W Owners Group – Master Integrated Reactor Vessel Surveillance Program, October 1995.

13. Point Beach Nuclear Plant Units 1 and 2 Final Safety Analysis Report, June 2003.

14. ASME Boiler and Pressure Vessel Code, Appendix K, Section XI, 2001 Edition.

15. EPRI NP-719-SR, T.U. Marston, Flaw Evaluation Procedures: ASME Section XI, Electric Power Research Institute, Palo Alto, California, August 1978.

11.0 *Certification*

This report is an accurate description of the low upper-shelf toughness fracture mechanics analysis performed for the reactor vessels at Point Beach.

7/21/04

H. P. Gunawardane, Engineer III — Date
Materials and Structural Analysis Unit

This report has been reviewed and found to be an accurate description of the low upper-shelf toughness fracture mechanics analysis performed for the reactor vessels at Point Beach.

7/21/04

A. D. Nana, Principal Engineer — Date
Materials and Structural Analysis Unit

Verification of independent review.

7/22/2004

A. D. McKim, Manager — Date
Materials and Structural Analysis Unit

This report is approved for release.

for R.E. Austin 7/22/04

R. E. Austin, Project Development Manager — Date

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12.0 Appendix A

The following pages contain input information from Nuclear Management Company.

NMC
Committed to Nuclear Excellence

Point Beach Nuclear Plant
Operated by Nuclear Management Company, LLC

NPL 2004-0139

June 29, 2004

Heshan Gunawardane
AREVA / Framatome ANP, Inc.
MS OF50
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Lynchburg, VA 24501

Heshan:

This correspondence will serve to formally document the requested inputs for the PBNP Units 1 and 2 RPV Equivalent Margins Assessment that is being performed in accordance with AREVA Proposal FANP-04-1067, April 2, 2004.

Applicable ASME Section XI Code

The PBNP ISI Program is in the fourth ten-year interval, which began on July 1, 2002 for both PBNP-1 and PBNP-2. The program is in accordance with the 1998 edition through 2000 addenda (98A00) of ASME Section XI Code as modified by 10 CFR 50.55a and approved relief requests and code cases. (Reference 1)

Fluence Projections

For the case of full uprated power condition (1678 MWt), without hafnium absorber assemblies, for EOLE (53 EFPY) use the older calculated fluence projections contained in Section 2 of Reference 2. This is requested for input consistency with the remaining RV embrittlement analyses.

For the cases of mini uprated power condition (1540 MWt), with and without hafnium absorber assemblies, for EOLE (53 EFPY) use the revised calculated fluence projections contained in Section 2 of Reference 3.

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Normal Heatup and Cooldown Rates

The PBNP RCS heatup and cooldown rates for normal operation are 100 degrees Fahrenheit per hour for both heatups and cooldowns. (Reference 4)

Predicted Operating Temperatures

The analyses for current licensed rated power conditions (1540 MWt) include a range of full load T(avg)'s from 558.1 to 574 degrees Fahrenheit. The resulting T(hot) and T(cold) ranges are 588.1 to 603.5, and 528 to 544.5 degrees Fahrenheit, respectively (Reference 5). PBNP currently uses a T(avg) program of 547 to 570 degrees Fahrenheit (no load to full load) (Reference 6), resulting in a T(hot) and T(cold) of approximately 597 and 542 degrees Fahrenheit, respectively (Reference 7).

The analyses for the 10.5 percent uprated power condition (1678 MWt) include a range of T(avg) from 558.6 to 573.4 degrees Fahrenheit. The resulting T(hot) and T(cold) ranges are 591.2 to 605.5, and 526 to 541.4 degrees Fahrenheit, respectively (Reference 8).

Transient Information

The original component transients are defined in each RPV design specification (References 9 and 10 for Units 1 and 2, respectively). A revised set of component design transients was generated to support steam generator replacement, a partial power uprate (8.7 percent), and license renewal (Reference 11). The RPV transients were evaluated and characterized for the partial power uprated condition in Reference 12. The RPV transients were further evaluated and characterized for full uprated conditions in Reference 13.

In addition, Chapter 14 of the PBNP FSAR (Reference 14) has been provided via previous correspondence. Chapter 14 contains the PBNP safety analysis summaries. These transients should be reviewed for bounding conditions with respect to the component design transients.

Applicable ASME Section II and III Code

ASME Boiler and Pressure Vessel Code, Section II, 1989, no Addenda.

ASME Boiler and Pressure Vessel Code, Section III, 1989, no Addenda.

Sincerely,

Brad Fromm
PBNP License Renewal
Nuclear Management Company

James E. Knorr
Manager of License Renewal PBNP
Nuclear Management Company

bms

References:

1. SER 2001-0010, "Point Beach Nuclear Plant, Units 1 and 2 – Relief Requests RR 1-24 (Unit 1) And RR-2-30 (Unit 2) Re: Use Of ASME Code Section XI, 1998 Edition With Addenda Through 2000 (TAC Nos. MB2230 And MB2231)", dated November 6, 2001.

2. Westinghouse Letter Report, LTR-REA-02-23, "Pressure Vessel Neutron Exposure Evaluations, Point Beach Units 1 and 2, S. L. Anderson, Radiation Engineering and Analysis, February 2002.

3. Westinghouse Letter Report, LTR-REA-04-64, "Pressure Vessel Neutron Exposure Evaluations, Point Beach Units 1 and 2, S. L. Anderson, Radiation Engineering and Analysis, June 2004.

4. Point Beach Nuclear Plant Technical Requirements Manual Pressure Temperature Limits Report, Section 2.1, "RCS Pressure and Temperature Limits (LCO 3.4.3)", page 2.2-2, Revision 1, dated December 20, 2002.

5. NMC Letter, NRC 2002-0075, "Responses to Requests for Additional Information, License Amendment Request 226, Measurement Uncertainty Recapture Power Uprate", August 29, 2002.

6. Setpoint Document, STPT 5.1, "Primary Control Systems Rod Speed Control", Revision 7.

7. Internal PBNP email, Steve Barkhahn to Brad Fromm, dated 4/17/04.

8. Westinghouse, Power Uprate Project, PBNP Units 1 and 2, Volume 1 NSSS Engineering Report, and Volume 2 BOP Engineering Report, April 2002.

9. Section 4 of Westinghouse Equipment Specification G - 676243, "Reactor Coolant System – Reactor Vessel", Revision 0, 05/05/1966.

10. Section 4, and Figures 1 through 15 of Westinghouse Equipment Specification E-spec 677456, "Addendum to Equipment Specification 676413 Rev. 1, Reactor Coolant System – Reactor Vessel", Revision 2, 07/06/1971.

11. Appendix A of Westinghouse Design Specification, 414A83, "Point Beach Nuclear Plants Units 1 and 2, replacement Reactor Vessel Closure Head (RRVCH)", Revision 0.

12. Appendix B of WCAP-14448, "Addendum to the Stress Reports for the Point Beach Unit Nos. 1 and 2 Reactor Vessels (RSG/Uprating Evaluation), August 1995.

13. Section 5.1.4 of Westinghouse Report, "Power Uprate Project, Point Beach Nuclear Plant, Units 1 and 2, NSSS Engineering Report", April 2002.

14. Chapter 14 of the PBNP Units 1 and 2 Final Safety Analysis Report, June, 2003.

Notes:

References 1, 4, 5, 6, 7, 8, and 14 document the sources of the information.

References 2, 3, 9, 10, 11, 12, and 13 are enclosed.

References 9, 10, 11, 12, and 13 are Westinghouse Proprietary and shall be treated in accordance with the associated Westinghouse Proprietary Agreement established between AREVA/Framatome-ANP, NMC, and Westinghouse in June 2004.