NMC
Committed to Nuclear Excellence

Point Beach Nuclear Plant
Operated by Nuclear Management Company, LLC

June 6, 2006

NRC 2006-0054
10 CFR 50.90
10 CFR 50, App. G

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Point Beach Nuclear Plant Units 1 and 2
Dockets 50-266 and 50-301
License Nos. DPR-24 and DPR 27

Request for Review of Reactor Vessel Toughness Fracture Mechanics Analysis

Reference: 1) Letter from NMC to NRC dated October 25, 2004, NRC 2004-0111, "License Renewal Application Response to Request for Additional Information (TAC Nos. MC2099 and MC2100)"

Pursuant to 10 CFR 50.90, Nuclear Management Company, LLC (NMC), hereby requests a proposed amendment to the licenses for Point Beach Nuclear Plant (PBNP), Units 1 and 2. The proposed amendment would support a change to the PBNP Final Safety Analysis Report (FSAR) by incorporating an updated analysis for satisfying the reactor vessel Charpy upper-shelf energy requirements of 10 CFR 50, Appendix G, Section IV.A.1.

Enclosed for Commission review and approval is Areva Document BAW-2467P, Revision 1, "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", dated October 2004. BAW-2467P incorporates the latest (2004) fluence projections to be consistent with the Pressurized Thermal Shock (PTS) evaluation. This analysis is being submitted in accordance with the requirements of 10 CFR 50, Appendix G, Section IV.A.1.c.

The non-proprietary version of this document (BAW-2467NP, Revision 1), had previously been submitted in Reference 1 as part of the license renewal process in accordance with 10 CFR 54.

Enclosure 1 provides a description, justification, and a significant hazards determination for the reactor vessel toughness fracture mechanics analysis. Enclosure 2 submits BAW-2467NP, Revision 1 (non-proprietary). Enclosure 3 submits BAW-2467P, Revision 1, (proprietary). Enclosure 4 provides a Westinghouse authorization letter, accompanying affidavit, Proprietary Information Notice and Copyright Notice for the proprietary portion of the analysis. Also provided in Enclosure 4 are proprietary and non-proprietary versions of Westinghouse document WEP-06-33, which is the Westinghouse source document for Figure 5-1 of BAW-2467P.

Since the Areva document listed above as Proprietary contains information proprietary to Westinghouse, it is supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity, for each, the considerations listed in paragraph (b)(4) of Section 2.390 of the Commission's regulations.

Accordingly, it is respectfully requested that the information, which is proprietary to Westinghouse, be withheld from public disclosure in accordance with 10 CFR 2.390.

Correspondence with respect to the copyright or proprietary aspects of the above documents, or the supporting Westinghouse affidavit, should reference the appropriate authorization letter (CAW-06-2141) and be addressed to B. F. Maurer, Acting Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

This letter contains no new commitments and no revision to existing commitments.

NMC requests approval of the proposed license amendment by May 2007, with the amendment being implemented within 60 days.

In accordance with 10 CFR 50.91, a copy of this submittal, with enclosures, is being provided to the designated Wisconsin Official.

I declare under penalty of perjury that the foregoing is true and correct. Executed on June 6, 2006.

Dennis L. Koehl
Site Vice-President, Point Beach Nuclear Plant
Nuclear Management Company, LLC

Enclosures (4)

cc: Regional Administrator, Region III, USNRC
Project Manager, Point Beach Nuclear Plant, USNRC
Resident Inspector, Point Beach Nuclear Plant, USNRC
PSCW

ENCLOSURE 1

REQUEST FOR REVIEW OF REACTOR VESSEL TOUGHNESS FRACTURE MECHANICS ANALYSIS

1.0 INTRODUCTION

As required by 10 CFR 50, Appendix G, Section IV.A.1.c, and in accordance with 10 CFR 50.90, Nuclear Management Company, LLC (NMC) requests review and approval of a revised reactor vessel toughness fracture mechanics analysis for the Point Beach Nuclear Plant (PBNP) Units 1 and 2. The proposed license amendment would support a change to the PBNP Final Safety Analysis Report (FSAR) by incorporating the updated analysis for satisfying the reactor vessel Charpy upper-shelf energy requirements of 10 CFR 50, Appendix G, Section IV.A.1.

2.0 DESCRIPTION OF PROPOSED CHANGE

NMC proposes changing the PBNP licensing basis to incorporate a revised equivalent margins assessment for the reactor pressure vessels (RPV), in PBNP Units 1 and 2, for material toughness when the upper-shelf Charpy energy level falls below 50 ft-lb. This assessment applied the 2004 Westinghouse fluence projection using full uprated power (1678 MWt), without crediting the presence of Hafnium power suppression inserts.

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 toughness 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 demonstrates that the limiting reactor vessel beltline welds at Point Beach Units 1 and 2 satisfy 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.

3.0 BACKGROUND

The NRC safety evaluation report associated with license renewal of the Point Beach Nuclear Plant Units 1 and 2 (NUREG-1839) references Areva analysis BAW-2467NP, dated July 2004. The July 2004 analysis was reissued October 2004 as Revision 1 to BAW-2467NP with updates to the fluence values used in the original analysis.

BAW-2467NP, Revision 1, dated October 2004, had previously been submitted in Reference 1 as part of the license renewal process in accordance with 10 CFR 54. In accordance with 10 CFR 50, Appendix G, Section IV.A.1.c, NRC review and approval of this document is required for PBNP to incorporate it into the licensing basis.

4.0 TECHNICAL ANALYSIS

The technical justification for the revised reactor vessel toughness fracture mechanics analysis is contained in the enclosed Areva document. The revised reactor vessel toughness fracture mechanics analysis provides technical justification for satisfying the reactor vessel Charpy upper-shelf energy requirements of 10 CFR 50, Appendix G, Section IV.A.1.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Determination

As required by 10 CFR 50, Appendix G, Section IV.A.1.c, and in accordance with 10 CFR 50.90, Nuclear Management Company, LLC (NMC) requests review and approval of a revised reactor vessel toughness fracture mechanics analysis for the Point Beach Nuclear Plant (PBNP) Units 1 and 2. The proposed license amendment would support a change to the PBNP Final Safety Analysis Report (FSAR) by incorporating the updated analysis for satisfying the reactor vessel Charpy upper-shelf energy requirements of 10 CFR 50, Appendix G, Section IV.A.1.

NMC has evaluated the proposed amendment in accordance with 10 CFR 50.91 against the standards in 10 CFR 50.92 and has determined that the operation of PBNP Units 1 and 2, in accordance with the proposed amendments, presents no significant hazards. The NMC evaluation against each of the criteria in 10 CFR 50.92 follows:

1. **Would the proposed amendment involve a significant increase in the probability or consequences of any accident previously evaluated?**

   The proposed change incorporates the updated analysis for satisfying the reactor vessel Charpy upper-shelf energy requirements of 10 CFR 50, Appendix G, Section IV.A.1 into the FSAR. The proposed change does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or the manner in which the plant is operated and maintained. The proposed change does not alter or prevent the ability of structures, systems, and components from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed change does not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Further, the proposed change does not increase the types or amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposures. The proposed change is consistent with safety analysis assumptions and resultant consequences. Therefore, it is concluded that this change does not significantly increase the probability of occurrence of an accident previously evaluated.

2. **Would the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change incorporates the updated analysis for satisfying the reactor vessel Charpy upper-shelf energy requirements of 10 CFR 50, Appendix G, Section IV.A.1 into the FSAR. The change does not impose any new or different requirements or eliminate any existing requirements. The change does not alter assumptions made in the safety analysis. The proposed change is consistent with the safety analysis assumptions and current plant operating practice. Therefore, the proposed change would not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Would the proposed amendment result in a significant reduction in a margin of safety?**

   The proposed change incorporates the updated analysis for satisfying the reactor vessel Charpy upper-shelf energy requirements of 10 CFR 50, Appendix G, Section IV.A.1 into the FSAR. The proposed change does not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The setpoints at which protective actions are initiated are not altered by the proposed change. Therefore, the proposed amendment does not result in a significant reduction in a margin of safety.

Conclusion

Operation of PBNP in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of any accident previously analyzed; would not create the possibility of a new or different kind of accident from any accident previously analyzed; and, would not result in a significant reduction in any margin of safety. Therefore, operation of PBNP in accordance with the proposed amendment presents no significant hazards.

5.2 Applicable Regulatory Requirements

10 CFR 50, Appendix G, Section IV.A.1 promulgates reactor vessel Charpy upper-shelf energy requirements. 10 CFR 50, Appendix G, Section IV.A.1.c

requires that the analysis for satisfying the reactor vessel Charpy upper-shelf energy requirements of 10 CFR 50, Appendix G, Section IV.A.1 must be submitted for review and approval on an individual case basis at least three years prior to the date when the predicted Charpy upper-shelf energy will no longer satisfy the requirements of Section IV.A.1.

10 CFR 50.71(e) requires that licensees shall periodically update their final safety analysis report (FSAR), to assure that the information included in the report contains the latest information developed. This update shall contain all the changes necessary to reflect information and analyses submitted to the Commission by the licensee or prepared by the licensee pursuant to Commission requirement. The update shall also include the effects of all analyses of new safety issues performed by or on behalf of the licensee at Commission request.

Based upon the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in accordance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.3 Commitments

There are no actions committed to by NMC in this document. Statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

6.0 ENVIRONMENTAL EVALUATION

NMC has determined that the information for the proposed amendment does not involve a significant hazards consideration, authorize a significant change in the types or total amounts of effluent release, or result in any significant increase in individual or cumulative occupational radiation exposure.

Accordingly, this proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with this proposed amendment.

ENCLOSURE 2 TO
REQUEST FOR REVIEW OF REACTOR VESSEL TOUGHNESS
FRACTURE MECHANICS ANALYSIS

AREVA DOCUMENT,
BAW-2467NP, REVISION 1, "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", DATED OCTOBER 2004
(NON-PROPRIETARY)

(46 pages follow)

BAW-2467NP, Rev. 1
October 2004

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

AREVA Document No. 77-2467NP-01
(See Section 11 for document signatures.)

Prepared for

Nuclear Management Company

Prepared by

Framatome ANP, Inc.
An AREVA and Siemens company
3315 Old Forest Road
P. O. Box 10935
Lynchburg, Virginia 24506-0935

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 toughness 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.

RECORD OF REVISIONS

Revision	Affected Pages	Description	Date
0	All	Original release	July 2004
1	All	Updated fluence values used for Evaluation Condition 1	October 2004

AREVA

TABLE OF CONTENTS

1.0 Introduction 1-1

2.0 Changes in Operating Condition Parameters 2-1

3.0 Material Properties and Reactor Vessel Design Data 3-1
3.1 J-Integral Resistance Model for Mn-Mo-Ni/Linde 80 Welds 3-1
3.2 Reactor Vessel Design Data 3-1
3.3 Mechanical Properties for Weld Material 3-1
3.3.1 Axial Weld SA-847 3-2
3.3.2 Circumferential Weld SA-1011 3-3
3.3.3 Circumferential Weld SA-1484 3-4

4.0 Analytical Methodology 4-1
4.1 Procedure for Evaluating Levels A and B Service Loadings 4-1
4.2 Procedure for Evaluating Levels C and D Service Loadings 4-1
4.3 Temperature Range for Upper-Shelf Fracture Toughness Evaluations 4-3
4.4 Effect of Cladding Material 4-3

5.0 Applied Loads 5-1
5.1 Levels A and B Service Loadings 5-1
5.2 Levels C and D Service Loadings 5-1

6.0 Evaluation for Levels A and B Service Loadings 6-1

7.0 Evaluation for Levels C and D Service Loadings 7-1

8.0 Summary of Results 8-1

9.0 Conclusion 9-1

10.0 References 10-1

11.0 Certification 11-1

12.0 Appendix A 12-1

13.0 Appendix B 13-1

LIST OF TABLES

Table 2-1 Evaluation Conditions 2-2

Table 3-1 Mechanical Properties for SA-847 Weld of Point Beach Unit 1 3-2

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

Table 3-3 Mechanical Properties for SA-1484 Weld of Point Beach Unit 2 3-4

Table 6-1 Material *J*-Integral Resistance for Levels A and B Service Loadings – Evaluation Condition 1 – Uprated Power Conditions Without Hafnium Assemblies 6-2

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

Table 6-3 Material *J*-Integral Resistance for Levels A and B Service Loadings – Evaluation Condition 3 – Current Power Conditions With Hafnium Assemblies 6-2

Table 6-4 Flaw Evaluation for Levels A and B Service Loadings – Evaluation Condition 1 – Uprated Power Conditions Without Hafnium Assemblies 6-3

Table 6-5 Flaw Evaluation for Levels A and B Service Loadings – Evaluation Condition 2 – Current Power Conditions Without Hafnium Assemblies 6-3

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

Table 7-1 *J*-Integral vs. Flaw Extension for Evaluation Condition 1 - SA-847 7-6

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

Table 7-3 *J*-Integral vs. Flaw Extension for Evaluation Condition 1 - SA-1484 7-8

Table 7-4 Level D Service Loadings - Internal Pressure at Tensile Instability - SA-847 7-9

LIST OF FIGURES

2-1 Reactor Vessel of Point Beach Unit 1 2-3

2-2 Reactor Vessel of Point Beach Unit 2 2-4

5-1 Level D transients – Reactor Coolant Temperature and Pressure vs. Time 5-2

6-1 *J*-Integral vs. Flaw Extension for Levels A and B Service Loading - Evaluation Condition 2 - Current Power Conditions Without Hafnium Assemblies – Weld SA-847 6-4

7-1 K_I vs. Crack Tip Temperature for Evaluation Condition 1 – SA-847 7-2

7-2 K_I vs. Crack Tip Temperature for Evaluation Condition 1 – SA-1101 7-3

7-3 K_I vs. Crack Tip Temperature for Evaluation Condition 1 – SA-1484 7-4

7-4 *J*-Integral vs. Flaw Extension – All Welds 7-10

7-5 *J*-Integral vs. Flaw Extension – SA-847 7-11

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.

Revision 1 of this document utilizes the updated fluence values calculated in 2004 for the uprated power condition of 1678 MWt without the hafnium power suppression assemblies installed. This input was provided by the Nuclear Management Company (NMC) and is included as Appendices A and B.

2.0 Changes in Operating Condition Parameters

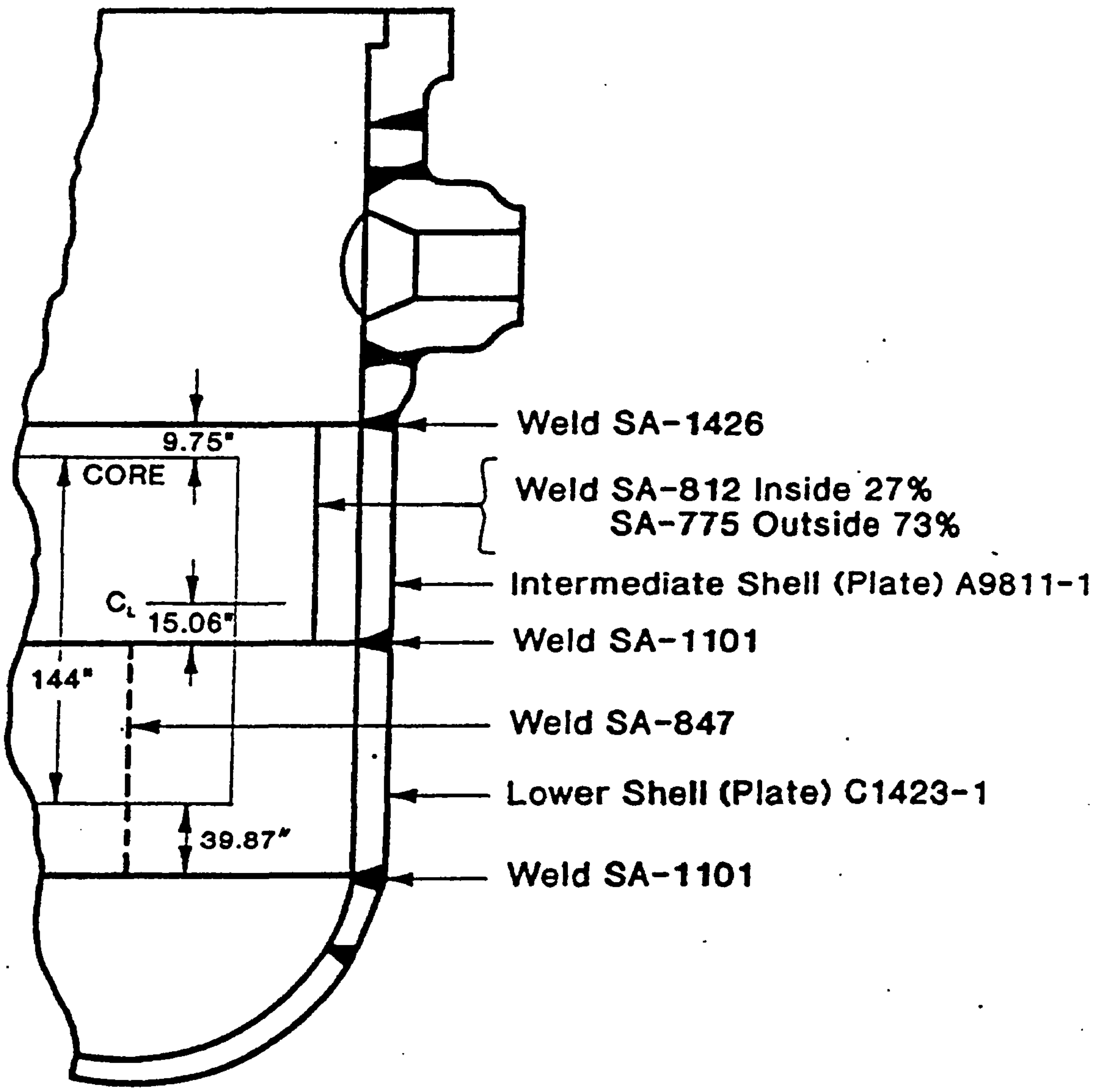
As a result of the planned updates to the Point Beach Units 1 and 2, there are increases in the projected end of life fluences for both the units. There are also changes in the plants' operating temperatures. These inputs were provided by the Nuclear Management Company and included as Appendices A and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

Table 2-1 Evaluation Conditions

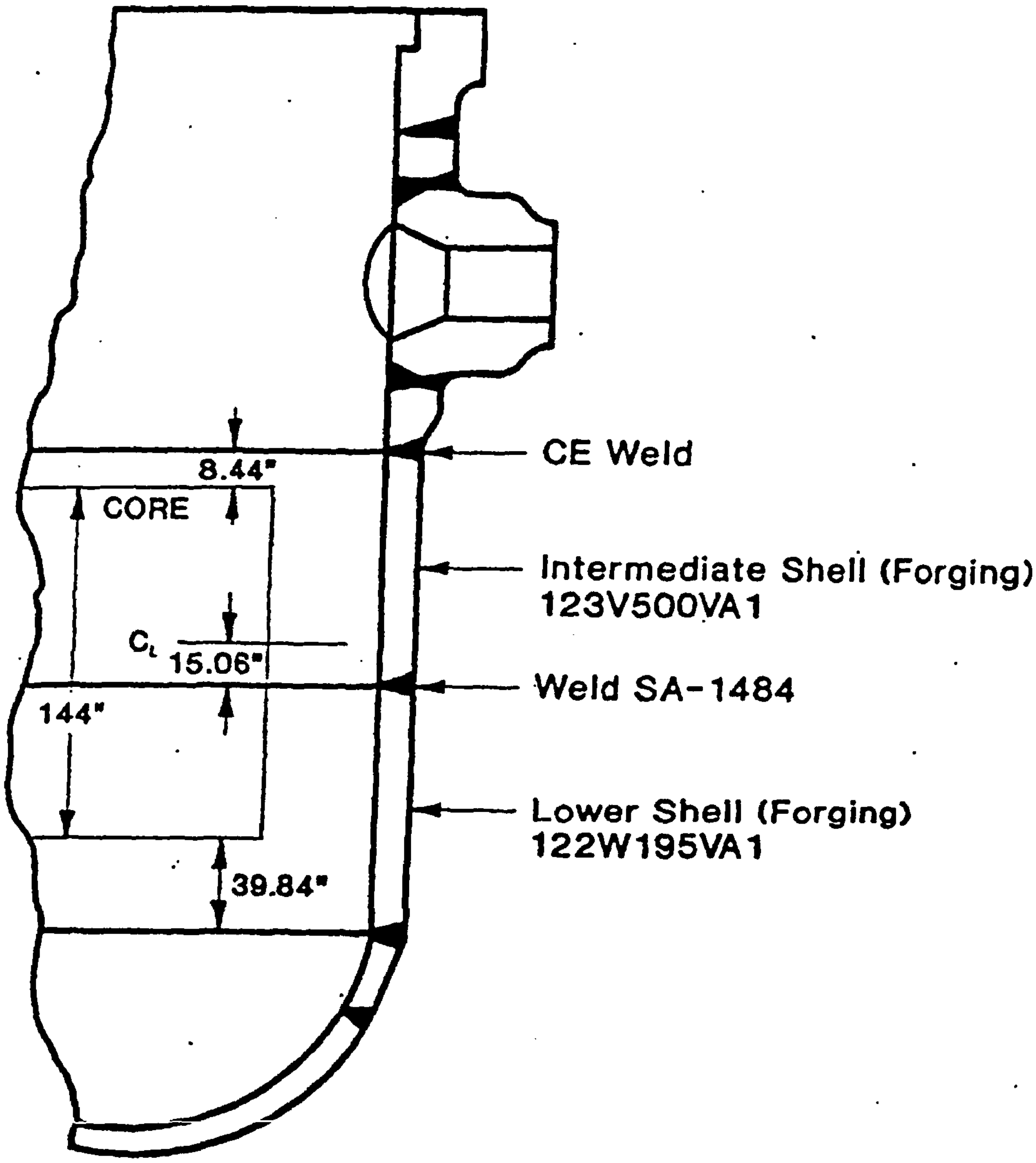
Plant	Weld Location [1]	Weld Number [1]	Cu (wt%) [5]	Ni (wt%) [5]	Fluence (n/cm^2) at 53 EFPY		
					EVALUATION CONDITION 1 Uprated Power Conditions Without Hafnium Assemblies Cold Leg Temp: 541.4°F	EVALUATION CONDITION 2 Current Power Conditions Without Hafnium Assemblies Cold Leg Temp: 544.5°F	EVALUATION 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	3.25E+19	3.12E+19	2.67E+19
	Inter. Shell/Lower Shell Circ.	SA-1101	0.23	0.59	4.71E+19	4.52E+19	3.82E+19
PB-2	Inter. Shell/Lower Shell Circ.	SA-1484	0.26	0.60	4.85E+19	4.65E+19	3.79E+19

Figure 2-1 Reactor Vessel of Point Beach Unit 1 [1]



AREVA

Figure 2-2 Reactor Vessel of Point Beach Unit 2 [1]



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_{Iclad}, 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_{Iclad}, 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_{Iclad} 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 2, current 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 (2) 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	32.45	21.45	886	619
PB-1	541.4	SA-1101	C	0.23	47.10	31.13	871	609
PB-2	541.4	SA-1484	C	0.26	48.45	32.03	828	579

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	619	331	1.87	388	1.60
PB-1	SA-1101	C	609	98	6.21	113	5.39
PB-2	SA-1484	C	579	104	5.57	119	4.87

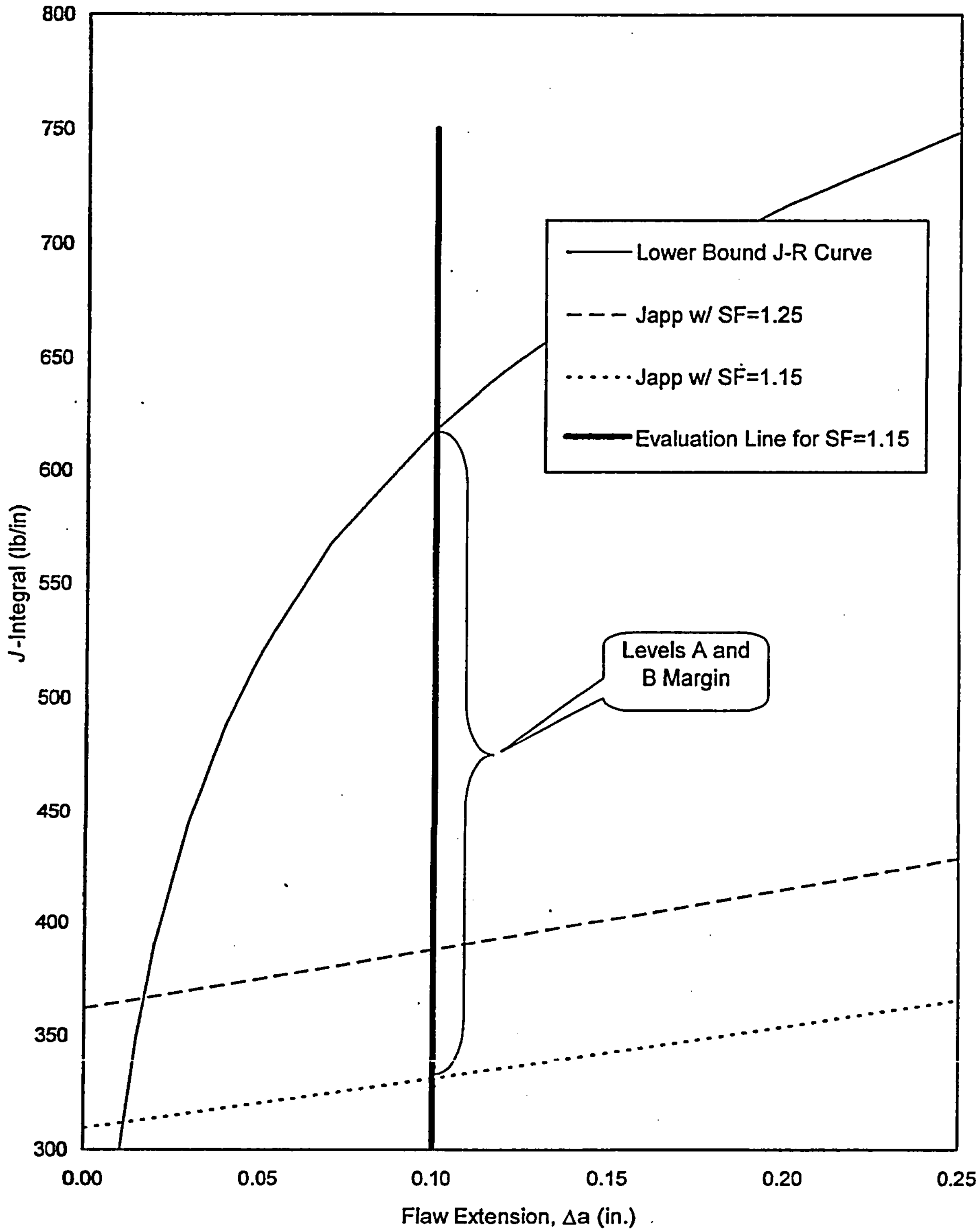
Table 6-5 Flaw Evaluation for Levels A and B Service Loadings – Evaluation Condition 2 – Current Power Conditions Without Hafnium Assemblies

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 2 - Current Power Conditions Without Hafnium Assemblies - Weld SA-847



AREVA

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].

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

where:

K_{Ic} = transition region toughness, ksi$\sqrt{\text{in}}$
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$\sqrt{\text{in}}$
$J_{0.1}$ = 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

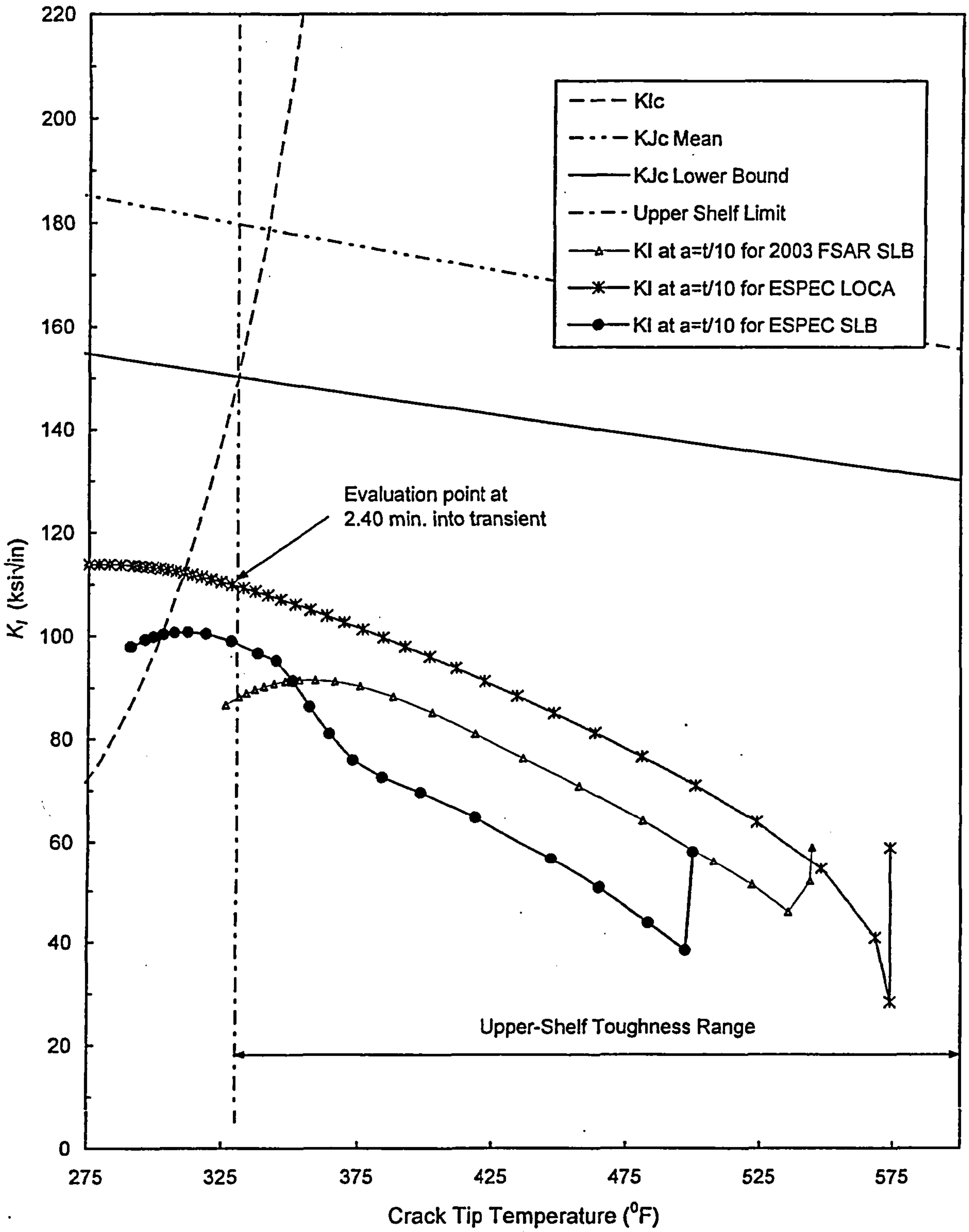


Figure 7-2 K_I vs. Crack Tip Temperature for Evaluation Condition 1 - SA-1101

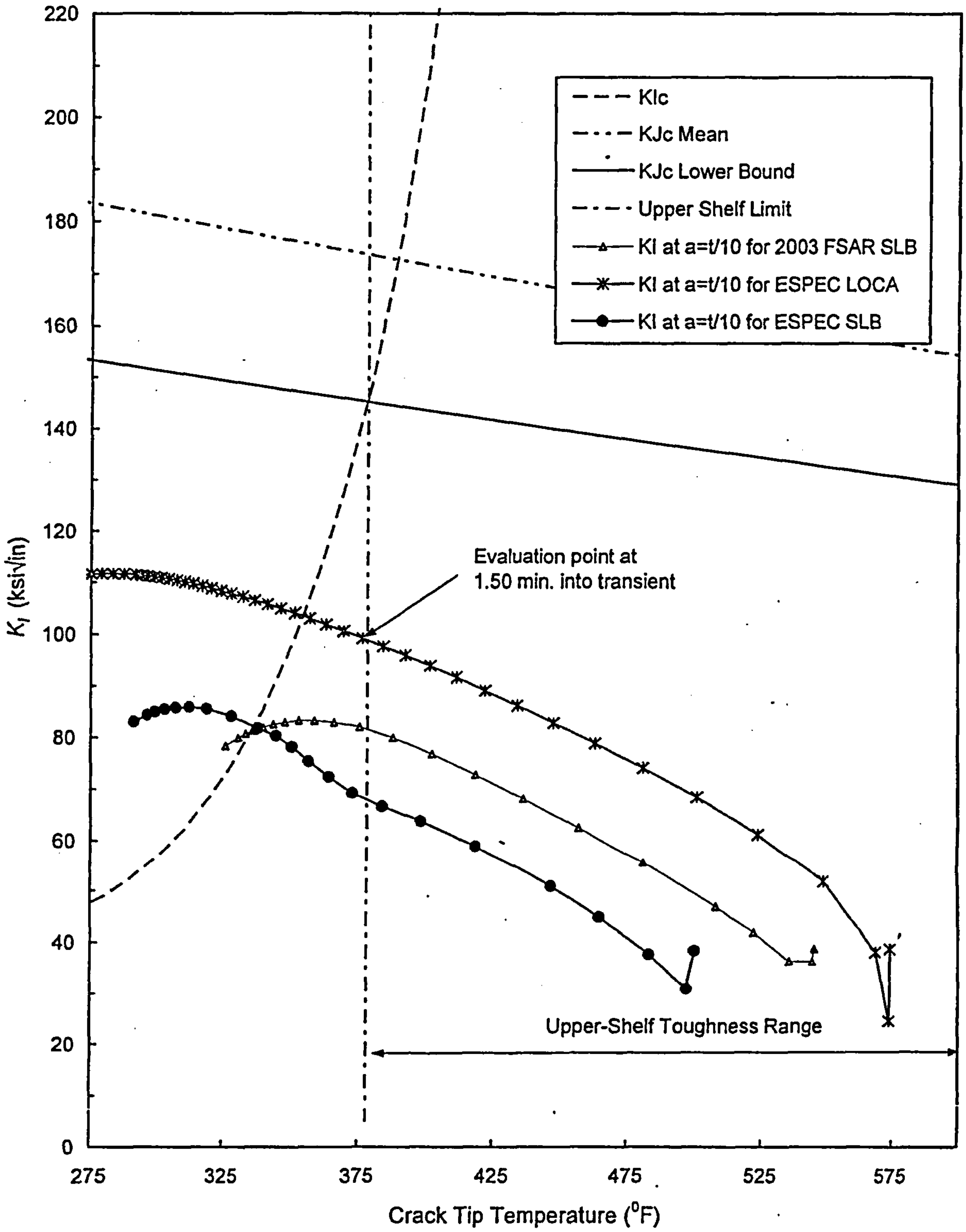
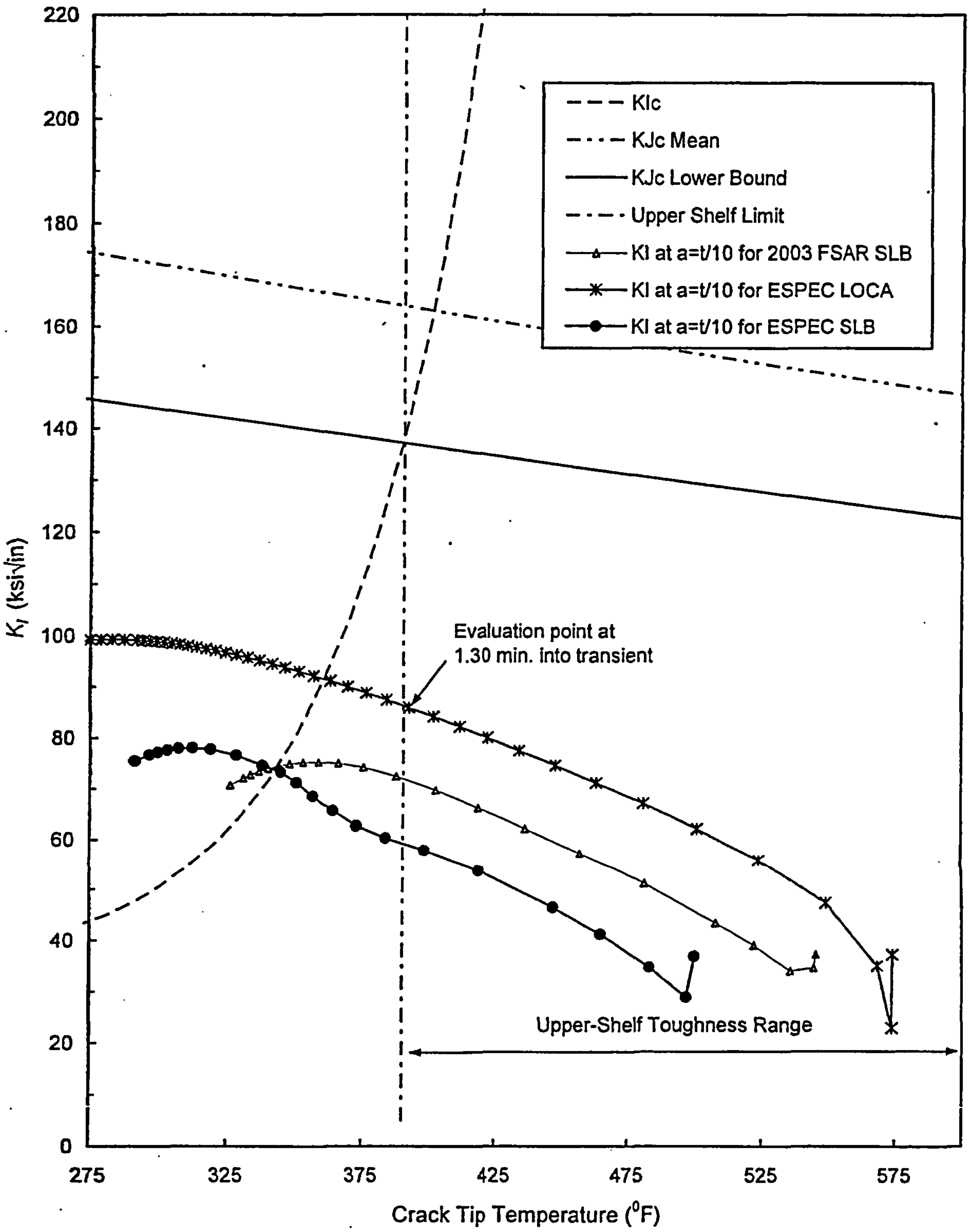


Figure 7-3 K_I vs. Crack Tip Temperature for Evaluation Condition 1 - SA-1484



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$\sqrt{\text{in}}$ to account for cladding effects. Stress intensity factors are converted to J-integrals by the plain strain relationship,

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

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_l, the internal pressure at tensile instability of the remaining ligament. Table 7-4 gives the results of the calculations for P_l for flaw depths up to 1.365 inches for Evaluation Condition 1. As the internal pressure p is less than P_l, the remaining ligament is not subject to tensile instability.

AREVA

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.		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		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

AREVA

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.	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		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

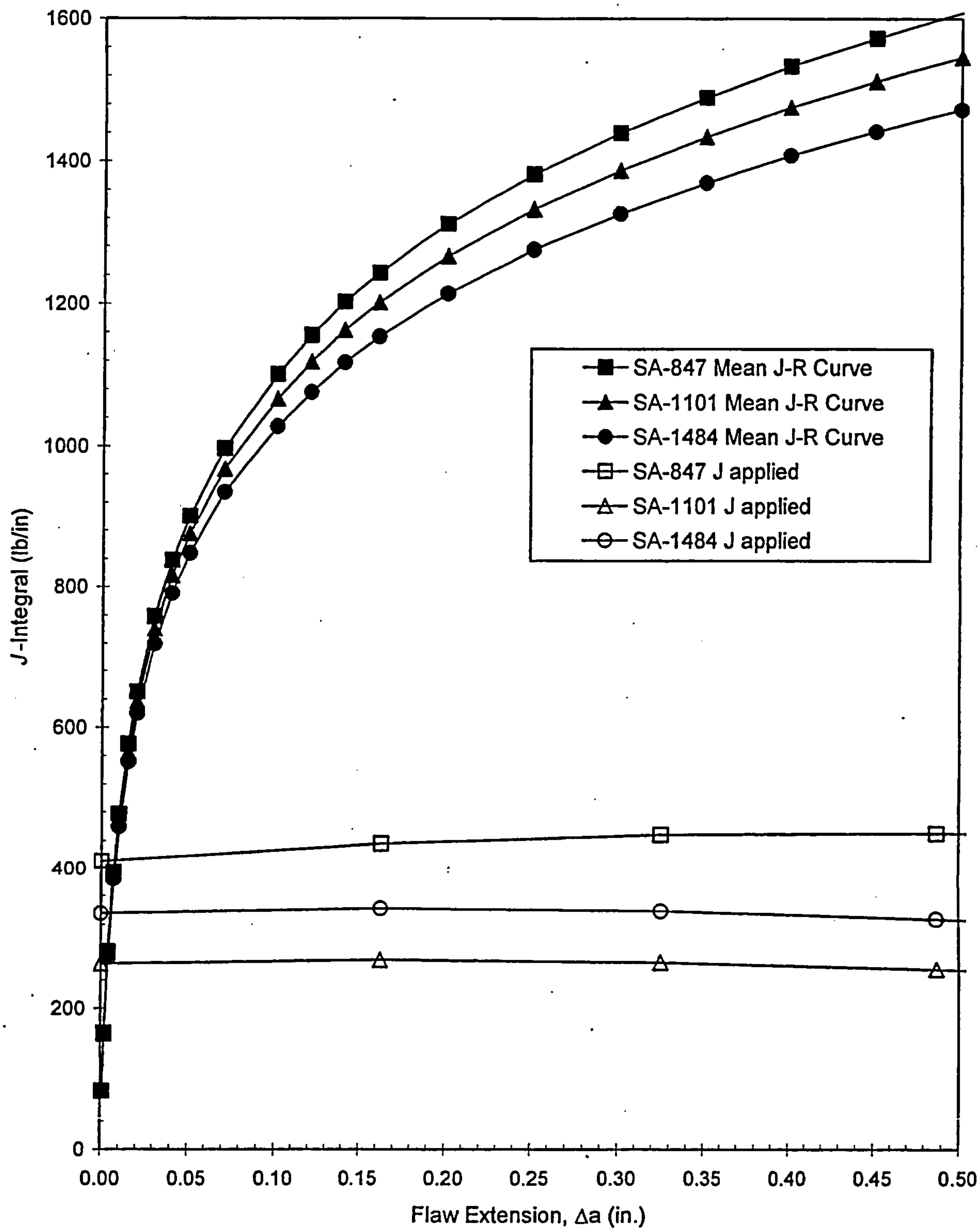
Time =	1.30 min				E =	25459.9	ksi
Crack tip at t/10		t =	6.5	in.	ν =	0.3	
$(a^{++}/t)*40$	a^{++} (in.)	Δa (in.)	Temp. (F)	K_{Isum}	K_{Iclad}	K_{Itotal}	J_{app} (lb/in)
1	0.1625		292.60	51.19	9.0	60.2	129
2	0.3250		328.30	67.16	9.0	76.2	207
3	0.4875		361.60	73.97	9.0	83.0	246
4	0.6500	0.0000	392.10	76.91	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

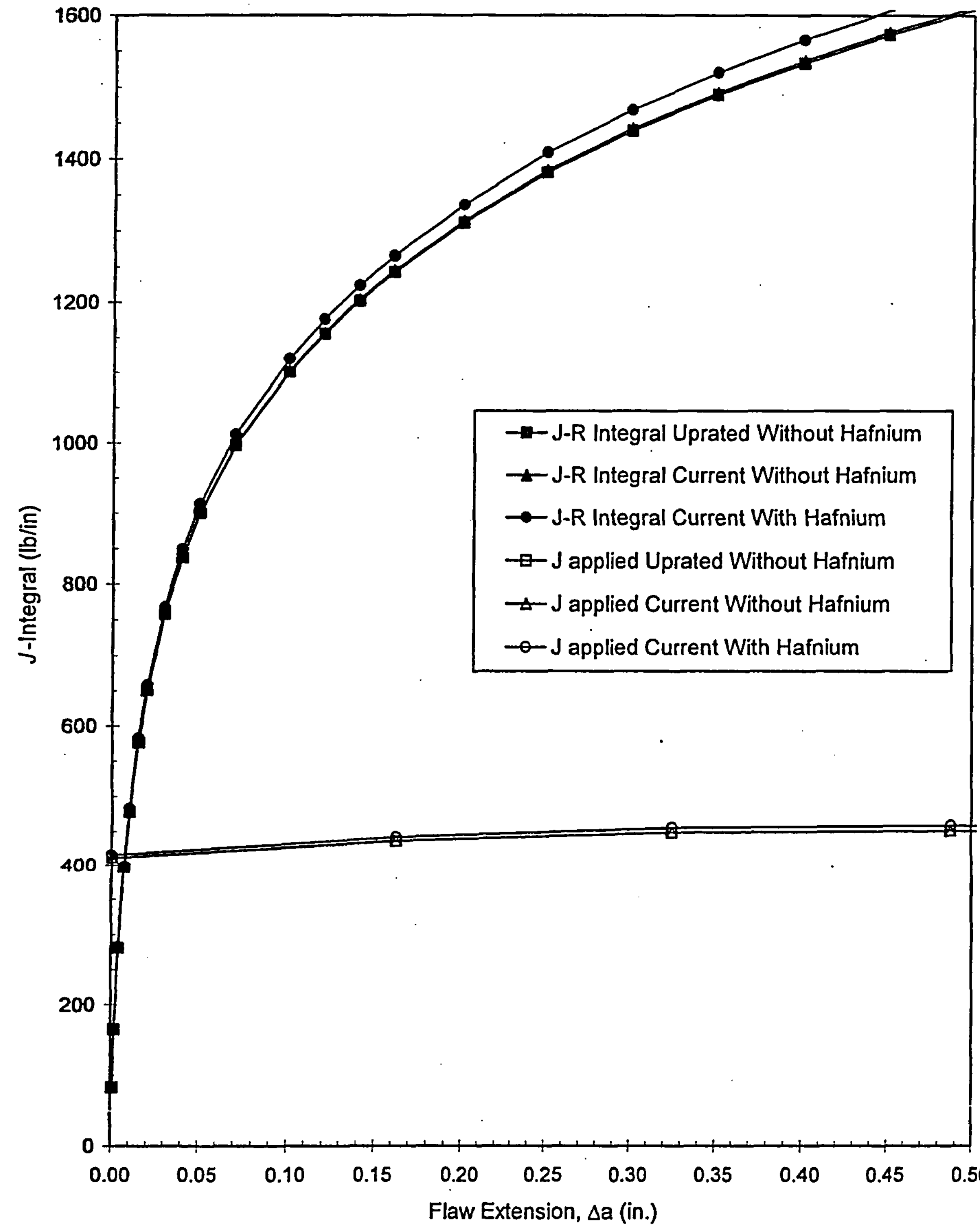
flaw depth *a* (in.)	P_l (ksi)
0.0650	9.18
0.1300	9.16
0.1950	9.14
0.2600	9.12
0.3250	9.09
0.3900	9.06
0.4550	9.02
0.5200	8.98
0.5850	8.93
0.6500	8.88
0.7150	8.84
0.7800	8.78
0.8450	8.73
0.9100	8.68
0.9750	8.62
1.0400	8.56
1.1050	8.51
1.1700	8.45
1.2350	8.39
1.3000	8.32
1.3650	8.26

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



AREVA

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



AREVA

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 axial weld SA-847 of Point Beach Unit 1 in the current 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.

10/15/04

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.

10/15/04

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

Verification of independent review.

10/15/04

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

This report is approved for release.

for R.E. Austin 10/15/04

R. E. Austin, Project Development Manager — Date

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
3315 Old Forest Road
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.

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,

WB Fromm

Brad Fromm
PBNP License Renewal
Nuclear Management Company

James E. Knorr

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.

13.0 Appendix B

The following page contains input information from Nuclear Management Company.

NMC
Committed to Nuclear Excellence

Point Beach Nuclear Plant
Operated by Nuclear Management Company, LLC

NPL 2004-0236

October 14, 2004

Heshan Gunawardane
AREVA / Framatome ANP, Inc.
MS OF50
3315 Old Forest Road
Lynchburg, VA 24501

Heshan:

Subject: PBNP Units 1 and 2 Equivalent Margins Assessment Revision, Framatome ANP, Inc. Proposal Number 416 0645, Addendum No. 1

This correspondence will serve to formally document NMC's request to revise the PBNP Units 1 and 2 RPV Equivalent Margins Assessment, Frametome ANP, Inc. Calculation Numbers 77-2647-00 and 77—2647NP-00, to use the 2004 Westinghouse fluence projection as the input to Evaluation Condition 1. Evaluation Condition 1 is full uprated power (1678 MWt), without the presence of Hafnium power suppression inserts.

Sincerely,

WB Fromm

Brad Fromm
PBNP License Renewal
Nuclear Management Company

John G. Thorgersen for James E. Knorr
Manager of License Renewal PBNP
Nuclear Management Company

bms

6590 Nuclear Road • Two Rivers, Wisconsin 54241
Telephone: 920.755.2321

ENCLOSURE 4 TO
REQUEST FOR REVIEW OF REACTOR VESSEL TOUGHNESS
FRACTURE MECHANICS ANALYSIS

WESTINGHOUSE DOCUMENT,
WEP-06-33 P ATTACHMENT, "TRANSMITTAL OF FIGURE 5-1 REACTOR COOLANT TEMPERATURE AND PRESSURE VS. TIME LEVEL D TRANSIENTS",
DATED MAY 10, 2006
(PROPRIETARY)

WESTINGHOUSE DOCUMENT,
WEP-06-33 NP ATTACHMENT, "TRANSMITTAL OF FIGURE 5-1 REACTOR COOLANT TEMPERATURE AND PRESSURE VS. TIME LEVEL D TRANSIENTS",
DATED MAY 8, 2006
(NON-PROPRIETARY)

WESTINGHOUSE AUTHORIZATION LETTER
AFFIDAVIT
PROPRIETARY INFORMATION NOTICE
COPYRIGHT NOTICE

(11 pages follow)

Westinghouse

Westinghouse Electric Company
Nuclear Services
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355
USA

May 8, 2006
WEP-06-33

Mr. Jack Gadzala
Nuclear Management Company
Point Beach Nuclear Plant
6610 Nuclear Road
Two Rivers, WI 54241

Nuclear Management Company
Point Beach Units 1 and 2
Transmittal of Figure 5-1 Reactor Coolant Temperature and Pressure vs. Time Level D. Transients

Dear Mr. Gadzala:

In response to your request attached please find proprietary and non-proprietary versions of the subject Figure 5-1 along with the appropriate documents to submit this information to the NRC.

If you have any questions regarding the attached, please call Mike Miller at 412-374-3353..

Very truly yours,

WESTINGHOUSE ELECTRIC COMPANY

Kerry B. Hanahan

Kerry B. Hanahan
Customer Project Manager

WEP-06-33 NP Attachment

Transmittal of Figure 5-1 Reactor Coolant Temperature and Pressure vs. Time Level D Transients

May 8, 2006

Westinghouse Electric Company LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

Figure 5-1 Level D transients – Reactor Coolant Temperature and Pressure vs. Time [b,c]

bcc: M. Miller
R. Fagan

Westinghouse

Westinghouse Electric Company
Nuclear Services
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

Direct tel: (412) 374-4419
Direct fax: (412) 374-4011
e-mail: maurerbf@westinghouse.com

Our ref: CAW-06-2141

May 12, 2006

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: WEP-06-33 P-Attachment, "Transmittal of Figure 5-1 Reactor Coolant Temperature and Pressure vs. Time Level D Transients" (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-06-2141 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by Nuclear Management Company.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-06-2141 and should be addressed to B. F. Maurer, Acting Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

BF Maurer

B. F. Maurer, Acting Manager
Regulatory Compliance and Plant Licensing

Enclosures

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared B. F. Maurer, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

B F Maurer

B. F. Maurer, Acting Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed
before me this 12th day
of May, 2006

Sharon L. Fiori

Notary Public

Notarial Seal
Sharon L. Fiori, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires January 29, 2007
Member, Pennsylvania Association Of Notaries

(1) I am Acting Manager, Regulatory Compliance and Plant Licensing, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse "Application for Withholding" accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in WEP-06-33 P-Attachment, "Transmittal of Figure 5-1 Reactor Coolant Temperature and Pressure vs. Time Level D Transients" (Proprietary) for submittal to the Commission, being transmitted by Nuclear Management Company letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse is that associated with Nuclear Management Company's request for NRC approval of BAW-2467P, Revision 1 October 2004.

This information is part of that which will enable Westinghouse to:

(a) Facilitate NMC in obtaining NRC approval of WEP-06-33 P-Attachment, "Transmittal of Figure 5-1 Reactor Coolant Temperature and Pressure vs. Time Level D Transients" (Proprietary).

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of this information to its customers for purposes of meeting NRC requirements for licensing documentation.

(b) Westinghouse can sell support and defense of the use of this information to its customers in the licensing process.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar calculations and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

Proprietary Information Notice

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

Copyright Notice