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Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads

BAW-2192NP
Supplement 1
Revision 0

Topical Report

December 2017

AREVA Inc.

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial Issue

Contents

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Nomenclature

Acronym	**Definition**
B&W	Babcock and Wilcox
B&WOG	Babcock and Wilcox Owners Group
CvUSE	Charpy Upper Shelf Energy
EFPY	Effective Full Power Years
EMA	Equivalent Margins Analysis
INF	Inlet Nozzle Forging
Jd	J deformation
J-R	J-integral Resistance
LAR	License Amendment Request
ONF	Outlet Nozzle Forging
ONS	Oconee Nuclear Station
PTN	Turkey Point Plant
RV	Reactor Vessel
RVWG	Reactor Vessel Working Group
SLR	Subsequent License Renewal
SRP	Standard Review Plan
Sy	Yield Strength
TSs	Technical Specifications
USE	Upper Shelf Energy

ABSTRACT

This Supplement 1 to BAW-2192PA reports an equivalent margins analysis (EMA) considering Levels A and B service loads for high copper Linde 80 weld metals and applicable non-Linde 80 welds using fluence values expected at 80-years (subsequent license renewal--SLR). This supplement to BAW-2192 applies to the following B&W-designed and Westinghouse-designed reactor vessels fabricated by B&W/Rotterdam: Oconee Nuclear Station (ONS) Units 1, 2, and 3 (Oconee Units 1, 2, and 3), Surry Units 1 and 2, and Turkey Point Plant (PTN) Units 3 and 4.

The analytical procedure used in this supplement is in accordance with ASME Section XI, Appendix K, Subarticle K-1200, with selection of design transients based on the guidance in Regulatory Guide 1.161, Section 4.0. EMA results are reported for all reactor vessel weld locations with 80-year fluence projections that exceed 1.0 E+17 n/cm^2 (E> 1.0 MeV). The ASME Section XI, acceptance criteria for Levels A & B Service Loads for all reactor vessel shell welds are satisfied. The acceptance criteria for Levels A & B Service Loads for RV transition welds and RV nozzle welds are also satisfied. Consistent with BAW-2192PA, Revision 00, the B&WOG J-R Model 4B is used for Linde 80 welds and Rotterdam welds. Plants that reference this report must calculate 80-year neutron fluence at reactor vessel weld locations in accordance with the requirements of NUREG-2192 (SLR-SRP), to demonstrate that the fluence estimates provided in Section 3.0 are applicable to their plants.

The EMA utilizes the B&WOG J-integral resistance (J-R) Model 4B reported in BAW-2192PA, Appendix B. Model 4B was developed based on fracture toughness test data obtained through approximately 1990, with specimen fluence that ranges from 0.0 to 8.45E18 n/cm^2. Eighty-year fluence estimates for the participating plants exceeds 8.45E18 n/cm^2 (e.g., maximum 80-year 1/4 T fluence is estimated at 6.5E19 n/cm^2) and use of Model 4B to estimate J-integral resistance values, including the associated model uncertainty, for 80-years is made by extrapolation of the model. To assess the model extrapolation uncertainty, Model 4B is compared to new fracture toughness test data (1990 to 2017) irradiated to fluence ranging from 8.0E18 n/cm^2 to 5.8E19 n/cm^2. The majority of test data fell above the Model 4B mean and all of the test data fell above the Model 4B mean minus 2 standard error band. Therefore, use of Model 4B and associated uncertainty to extrapolate J-integral resistance for 80-year fluence applications was determined to be appropriate. This assessment is reported in Appendix A.

To further substantiate the use of Model 4B, all of the original fracture toughness data used to develop Model 4B was combined with new fracture toughness data, using the same model form, and a new Model 6B was generated. Model 6B was found to be essentially equivalent to Model 4B with respect to model mean and 2 standard errors. The EMA results reported herein using Model 4B were reconciled to Model 6B, with little or no change to the EMA results. Model 6B development and the EMA reconciliation to Model 4B are reported in Appendix A.

1.0 INTRODUCTION

The purpose of Supplement 1 to BAW-2192PA is to report an equivalent margins analysis (EMA) considering Levels A and B service loads for high copper Linde 80 weld metals and applicable non-Linde 80 welds using fluence values expected at 80-years (subsequent license renewal--SLR). This supplement to BAW-2192 applies to the following B&W-designed and Westinghouse-designed reactor vessels fabricated by B&W/Rotterdam: Oconee Nuclear Station (ONS) Units 1, 2, and 3 (Oconee Units 1, 2, and 3), Surry Units 1 and 2, and Turkey Point Plant (PTN) Units 3 and 4, also referred to as Turkey Point Units 3 and 4.

Equivalent margins analyses for the plants within the scope of this report are reported for all reactor vessel weld locations with 80-year fluence projections that exceed 1.0 E+17 n/cm^2 (E> 1.0 MeV) [2]. Plants that reference this report must calculate 80-year neutron fluence at reactor vessel weld locations in accordance with the requirements of NUREG-2192 [3], Standard Review Plan for Review of Subsequent License Renewal Applications, to demonstrate that the fluence estimates provided in Section 3.0 are applicable to their plants. Upper shelf energy evaluations at reactor vessel base metal locations with 80-year fluence projections greater than 1.0 E+17 n/cm^2, if needed, will be addressed separately by each license renewal applicant and are not within the scope of this report.

The EMA utilizes the B&WOG J-integral resistance (J-R) model reported in BAW-2192PA, Appendix B, [1]. The following groups are used for the welds within the scope of this report:

- Reactor Vessel Shell Welds—circumferential and longitudinal welds (if applicable) within the upper and lower shell assemblies for Oconee Units 1, 2, and 3 (also referred to as Oconee reactor vessels), and within the intermediate and lower shell assemblies for Surry Units 1 and 2 (also referred to as Surry reactor vessels) and Turkey Point Units 3 and 4 (also referred to as Turkey Point reactor vessels). There are no geometric discontinuities at these weld locations and all reactor vessel shell welds surround the effective height of the active core. These locations have historically been considered "beltline" or "beltline region" as defined by 10 CFR 50, Appendix G. All reactor vessel shell welds are Linde 80 welds with the exception of Surry Unit 2 weld R-3008 (Figure 3—6), which is a Rotterdam weld.
- Transition Welds and RV Nozzle Welds—welds that are located above and below the reactor vessel shell welds that may experience 80-year fluence greater than 1.0 E+17 n/cm^2 [2] and, must consider the effects of neutron irradiation embrittlement. In addition, the transition welds are located at geometric discontinuities (e.g., lower shell to lower head and upper shell to nozzle belt forging). These locations may or may not have been included as part of the 10 CFR 50 Appendix G [4] "beltline" definition for 60-years for the participating plants. All transition welds and RV nozzle welds (also referred to as RV nozzle-to-shell welds) are Linde 80 welds with the exception of the following: Surry Unit 1 transition weld J726 (Figure 3—5), Surry Unit 2 transition weld L737 (Figure 3—6), and Surry Unit 2 RV outlet nozzle-to-nozzle belt forging welds, which are Rotterdam welds.

The EMA evaluations in this report are for all weld locations expected to receive fluence > 1.0E17 n/cm^2 [2] at 80 years. The use of the terms "beltline" and/or "extended beltline" are not used in this report. It is the responsibility of each license renewal applicant to address the 80-year 10 CFR 50 Appendix G beltline definition for their plant(s) in their license renewal application submittal.

The 60-year EMA summary reports for Oconee Units 1, 2, and 3, Surry Units 1 and 2, and Turkey Point Units 3 and 4 are reported in Section 1.1. Section 2.0 provides the current NRC regulatory requirements for the EMA. Section 3.0 provides a description of all reactor vessels within the scope of this report, with illustrations of reactor vessel welds that are evaluated for equivalent margins in Figures 3-1 through 3-8. Section 4.0 provides the material properties that are required for the EMA, and Section 5.0 presents the results of the EMA. Section 6.0 provides the summary and conclusions, Section 7.0 lists all references and Appendix A provides the technical basis for the use of B&WOG J-R Model 4B for the EMA reported herein.

1.1 *Equivalent Margins Analysis—Analysis of Record*

BAW-2192PA, Revision 00 [1] provided the EMA analysis of record for Levels A and B service loads for Oconee Units 1, 2, and 3, Surry Units 1 and 2, and Turkey Point Units 3 and 4 for 40 years. For 60 years, Oconee Units 1, 2, and 3 relied on the EMA reported in BAW-2251A [5], while Surry Units 1 and 2 and Turkey Point Units 3 and 4 reported plant-specific evaluations. The summary reports for EMA analyses of record are as follows.

<u>Oconee Units 1, 2, and 3</u>

In February 2014, Duke Energy submitted a license amendment request to the NRC to revise the Oconee Units 1, 2, and 3 Technical Specifications (TSs) by replacing the reactor pressure vessel pressure-temperature (P-T) limits in TS 3.4.3 with new P-T limits applicable to 54 effective full power years [6]. The Duke P-T limit submittal referenced AREVA topical report ANP-3127, Oconee Nuclear Station Units 1, 2 & 3 Pressure-Temperature Limits at 54 EFPY [7]. This report provided updated 54 EFPY fluence projections for traditional beltline locations.

The NRC staff noted that the 54 EFPY fluence projections reported in ANP-3127 exceeded the 48 EFPY fluence reported in the Oconee Nuclear Station (ONS) License Renewal Application [8], which referenced AREVA report BAW-2251-A [5], Appendix B, for the ONS 48 EFPY equivalent margins analysis. The NRC performed equivalent margins analysis reconciliation for the Oconee reactor vessel Linde 80 welds, including the lower shell to dutchman weld, which was not addressed in BAW-2251-A, Appendix B. The NRC performed CvUSE drop calculations for the plates (Unit 1) and forgings (Units 2 and 3) to show that the upper shelf energy for these items remained above 50 ft-lbs at 54 EFPY [6]. The NRC did not revise the EMA for the RV nozzles reported in Section 3.1, Page 3-6, of BAW-2251A.

Surry Units 1 and 2

The Surry Units 1 and 2 current licensing basis equivalent margins analysis at 48 EFPY is summarized in Section 3.2.3 of NRC document "SURRY POWER STATION, UNIT NOS. 1 AND 2 -ISSUANCE OF AMENDMENTS REGARDING REACTOR VESSEL HEATUP AND COOLDOWN CURVES FOR 48 EFFECTIVE FULL-POWER YEARS," Adams Accession number ML11110A111 [9]. The NRC SER of the 48 EFPY P-T limits references the Dominion submittal entitled, VIRGINIA ELECTRIC AND POWER COMPANY SURRY POWER STATION UNITS 1 AND 2 UPDATE TO NRC REACTOR VESSEL INTEGRITY DATABASE AND EXEMPTION REQUEST FOR ALTERNATE MATERIAL PROPERTIES BASIS PER 10 CFR 50.60(b) [10]. Specifically, Attachment 4 to Reference [10] includes AREVA document BAW-2494, Revision 1, Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessel of Surry Units 1 and 2 for Extended Life through 48 Effective Full Power Years.

Turkey Point Units 3 and 4

The Turkey Point Units 3 and 4 current licensing basis equivalent margins analysis at 48 EFPY is summarized in Section 2.1.2 of NRC document "TURKEY POINT UNITS 3 AND 4 - ISSUANCE OF AMENDMENTS REGARDING EXTENDED POWER UPRATE (TAC NOS. ME4907 AND ME4908)," Adams Accession number ML11293A365 [11]. NRC acceptance of the Turkey Point EMA at 48 EFPY for EPU is based on the following documentation:

- LICENSE AMENDMENT REQUEST FOR EXTENDED POWER UPRATE, ATTACHMENT 4, L-2010-113, Attachment 4 ADAMS --ML103560177 [12]
- Supplemental Response to NRC Request for Additional Information (RAI) Regarding Extended Power Uprate (EPU) License Amendment Request (LAR) No. 205 and Equivalent Margin Analysis (EMA), L-2010-303, ADAMS-ML103610321 [13]. Reference [13] contains AREVA document 77-2312-03 (P), 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
- Response to NRC Request for Additional Information Regarding Extended Power Uprate License Amendment Request No. 205 and Reactor Materials Issues – Round 1, L-2011-029, ADAMS--ML110700068 [14]. Reference [14] provides a response to RAI CVIB-1.2 regarding the code year used to perform the equivalent margins analysis (i.e., 1998 Edition vs 2004 Edition). The NRC SER for the Turkey Point Extended Power Uprate, Section 2.1.2, contains the evaluation of upper shelf energy--ADAMS-ML 11293A365

2.0 REGULATORY REQUIREMENTS

2.1 *Regulatory Requirements*

In accordance with 10 CFR 50 Appendix G [4], IV, A, 1., Reactor Vessel Upper Shelf Energy Requirements are as follows:

a. Reactor vessel beltline materials must have Charpy upper-shelf energy in the transverse direction for base material and along the weld for weld material according to the ASME Code, of no less than 75 ft-lb (102 J) initially and must maintain Charpy upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb (68 J), unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation or Director, Office of New Reactors, as appropriate, that lower values of Charpy upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code. This analysis must use the latest edition and addenda of the ASME Code incorporated by reference into 10 CFR 50.55a (b)(2) at the time the analysis is submitted.

b. Additional evidence of the fracture toughness of the beltline materials after exposure to neutron irradiation may be obtained from results of supplemental fracture toughness tests for use in the analysis specified in section IV.A.1.a.

c. The analysis for satisfying the requirements of section IV.A.1 of this appendix must be submitted, as specified in § 50.4, 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 of this appendix, or on a schedule approved by the Director, Office of Nuclear Reactor Regulation or Director, Office of New Reactors, as appropriate.

When the reactor vessels within the scope of this report were fabricated, charpy V-notch testing of the reactor vessel welds were in accordance with the original construction code, which did not specifically require charpy V-notch tests on the upper shelf.

Applicable construction codes are as follows:

- Oconee--ASME Section III, 1965 Edition with Addenda through Summer 1967 (BAW-2251-A)
- Surry--ASME B&PV Code, Section III, 1968 Edition through Winter 1968 Addenda (UFSAR Table 4.1-9)
- Turkey Point--ASME B&PV Code, Section III, 1965 Edition through Summer 1966 Addenda

In accordance with NRC Regulatory Guide 1.161 [15], the NRC has determined that the analytical methods described in ASME Section XI, Appendix K, provide acceptable guidance for evaluating reactor pressure vessels when the Charpy upper-shelf energy falls below the 50 ft-lb limit of Appendix G to 10 CFR Part 50. However, the staff noted that Appendix K does not provide information on the selection of transients and gives very little detail on the selection of material properties. Selection of the limiting design transient (i.e., cooldown at 100 F/h) is consistent with BAW-2192PA [1], Section 5.3. Section 4.1 of this report includes a summary of the B&WOG J-integral resistance model, and Section 4.2 provides mechanical properties of weld metals.

The Linde 80 and Rotterdam weld locations that are included within the scope of this report (i.e., weld locations with 80-year projected fluence > 1.0E+17 n/cm^2) are all assumed to have upper shelf energy values below 50 ft-bs and thus require an equivalent margins analysis.

2.2 *Compliance with 10 CFR 50 Appendix G and Acceptance Criteria*

The analyses reported herein are performed in accordance with the 2007 Edition with 2008 Addenda [16] of Section XI of the ASME Code, Appendix K. The current edition of ASME Section XI listed in 10 CFR 50.55a is the 2013 Edition [17]. With regard to Appendix K, there are no differences between the 2007 Edition with 2008 Addenda and the 2013 Edition of ASME Section XI, and hence these ASME Section XI, Appendix K analyses are equally applicable to the 2013 Edition of the ASME Code.

The material properties used in this analysis are based on ASME Section II, Part D, 2007 Edition with 2008 Addenda. The only change in the material properties listed in the 2013 Edition of ASME Section II for the applicable properties is the coefficient of thermal expansion for stainless steel at 600°F; this value was changed from 9.8E-6 in/in/°F to 9.9E-6 in/in/°F. This does not impact the Levels A and B evaluation reported herein.

2.2.1 Acceptance Criteria Levels A and B

ASME Section XI [17], Subarticle K-2200, provides the following acceptance criteria for Levels A and B Service Conditions:

a) When evaluating adequacy of the upper shelf toughness for the weld material for Levels A and B Service Loadings, an interior semi-elliptical surface flaw with a depth one-quarter of the wall thickness and a length six times the depth shall be postulated, with the flaw's major axis oriented along the weld of concern, and the flaw plane oriented in the radial direction. When evaluating adequacy of the upper shelf toughness for the base material, both interior axial and circumferential flaws with depths one quarter of the wall thickness and lengths six times the depth shall be postulated, and toughness properties for the corresponding orientation shall be used. Smaller flaw sizes may be used when justified. Two criteria shall be satisfied:

1. The applied J-integral evaluated at a pressure 1.15 times the accumulation pressure as defined in the plant specific Overpressure Protection Report, with a structural factor of 1 on thermal loading for the plant specific heatup and cooldown conditions, shall be less than the J-integral of the material at a ductile flaw extension of 0.1 in. (2.5 mm).

2. Flaw extensions at pressures up to 1.25 times the accumulation pressure of K-2200(a)(1) shall be ductile and stable, using a structural factor of 1 on thermal loading for the plant specific heatup and cooldown conditions.

b) The J-integral resistance versus flaw extension curve shall be a conservative representation for the vessel material under evaluation.

3.0 DESCRIPTION OF OCONEE, SURRY AND TURKEY POINT REACTOR VESSELS

The Oconee, Surry and Turkey Point reactor vessels and applicable weld locations are shown in Figures 3-1 through 3-8. All weld locations evaluated for equivalent margins in this report are identified by an asterisk (*) in each Figure. Plant-specific weld copper and nickel content and 80-year fluence projections data needed for the equivalent margins analysis are provided in Table 3—1. The fluence projections are reported for all reactor vessel weld locations that are expected to exceed 1.0E+17 n/cm^2 at 80 years for the participating plants. All plants that reference this report must generate 80-year neutron fluence at reactor vessel locations in accordance with NUREG-2192, SLR-SRP, to demonstrate that the fluence estimates provided in Table 3-1 are applicable to their plants. Note that all fluence values are inside wetted surface with the exception of selected locations for Surry Units 1 and 2 and Turkey Point 3 and 4 marked by an * (clad/base metal interface). Copper and nickel content of the reactor vessel shell welds is consistent with EMA analyses of record reported in Section 1.1; the copper and nickel content for transition welds and RV nozzle-to-nozzle belt forging welds reported in Table 3-1 were obtained from either the EMA analysis of record or a search of Oconee, Surry, and Turkey Point reactor vessel fabrication reports.

Table 3—1
Reactor Vessel Weld Locations--Copper Content and 80-Year Fluence Projections

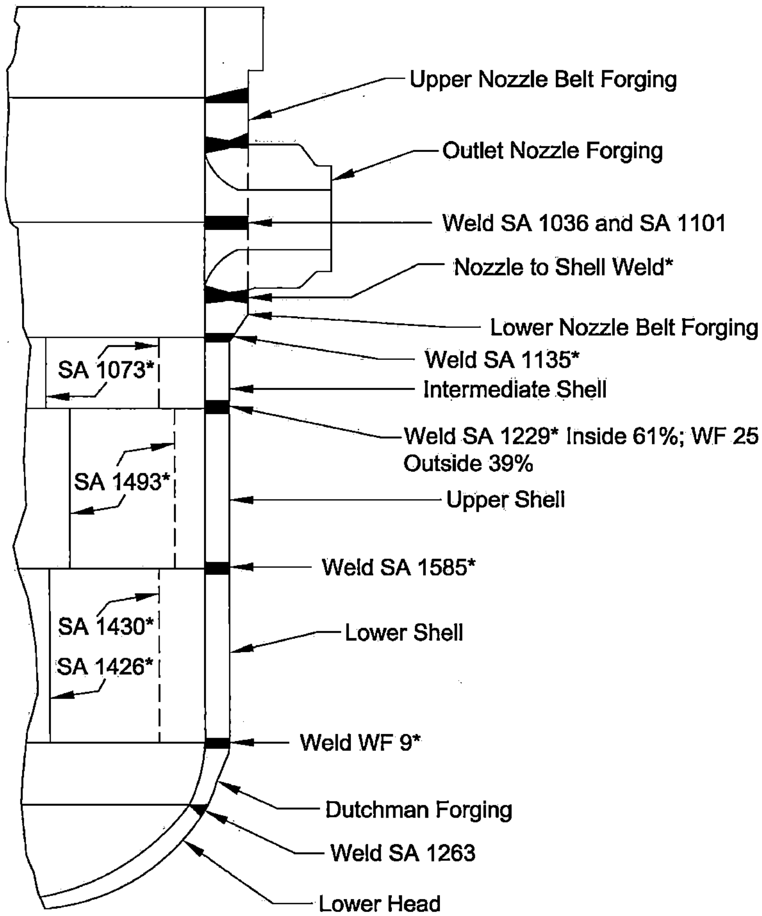
Reactor Vessel Material	Material ID and/or Heat Number	Cu, wt%	Ni, wt%	(IS) Inside Wetted Surface Fluence or (*) clad/base metal n/cm^2 E> 1.0 MeV)
Oconee Unit 1, 80-YearFluence (E > 1.0 MeV)				
Lower Nozzle Belt (LNB) to Outlet Nozzle Forging (ONF) Welds	Wire Ht. 8T1762	0.19	0.57	(IS) 1.50E+18
	Wire Ht. 299L44	0.34	0.68	(IS) 1.50E+18
	Wire Ht. 8T1554B	0.16	0.57	(IS) 1.50E+18
LNB to Inlet Nozzle Forging (INF) Welds	Wire Ht. 8T1762	0.19	0.57	(IS) 1.50E+18
	Wire Ht. 299L44	0.34	0.68	(IS) 1.50E+18
	Wire Ht. 8T1554B	0.16	0.57	(IS) 1.50E+18
LNB to Intermediate Shell (IS) Circ. Weld	SA-1135 (Wire Ht. 61782)	0.23	0.52	(IS) 1.90E+18
IS Long. Welds (Both)	SA-1073 (Wire Ht. 1P0962)	0.21	0.64	(IS) 1.58E+19
IS to Upper Shell (US) Circ. Weld (ID 61%)	SA-1229 (Wire Ht. 71249)	0.23	0.59	(IS) 2.02E+19
US Long. Welds (Both)	SA-1493 (Wire Ht. 8T1762)	0.19	0.57	(IS) 2.05E+19
US to Lower Shell (LS) Circ. Weld	SA-1585 (Wire Ht. 72445)	0.22	0.54	(IS) 2.14E+19
LS Long. Weld (1)	SA-1426 (Wire Ht. 8T1762)	0.19	0.57	(IS) 1.82E+19
LS Long. Weld (2)	SA-1430 (Wire Ht. 8T1762)	0.19	0.57	(IS) 1.82E+19
LS to Dutchman Circ. Weld	WF-9 (Wire Ht. 72445)	0.22	0.54	(IS) 4.88E+17
Oconee Unit 2, 80-Year Fluence (E > 1.0 MeV)				

Reactor Vessel Material	Material ID and/or Heat Number	Cu, wt%	Ni, wt%	(IS) Inside Wetted Surface Fluence or (*) clad/base metal n/cm^2 E> 1.0 MeV)
Lower Nozzle Belt (LNB) to Outlet Nozzle Forging (ONF) Welds	Wire Ht. 8T1762	0.19	0.57	(IS) 1.50E+18
	Wire Ht. 72445	0.22	0.54	(IS) 1.50E+18
LNB to Inlet Nozzle Forging (INF) Welds	Wire Ht. 8T1762	0.19	0.57	(IS) 1.50E+18
	Wire Ht. 72445	0.22	0.54	(IS) 1.50E+18
LNB to Upper Shell (US) Circ. Weld	WF-154 (Wire Ht. 406L44)	0.27	0.59	(IS) 1.99E+19
US to Lower Shell (LS) Circ. Weld	WF-25 (Wire Ht. 299L44)	0.34	0.68	(IS) 2.18E+19
LS to Dutchman Circ. Weld	WF-112 (Wire Ht. 406L44)	0.27	0.59	(IS) 5.20E+17
Oconee Unit 3, 80-Year Fluence (E > 1.0 MeV)				
Lower Nozzle Belt (LNB) to Outlet Nozzle Forging (ONF) Welds	Wire Ht. 72105	0.32	0.58	(IS) 1.50E+18
	Wire Ht. 406L44	0.27	0.59	(IS) 1.50E+18
LNB to Inlet Nozzle Forging (INF) Welds	Wire Ht. 72105	0.32	0.58	(IS) 1.50E+18
	Wire Ht. 72102	0.21	0.58	(IS) 1.50E+18
	Wire Ht. 82102	0.35	1.00	(IS) 1.50E+18
LNB to Upper Shell (US) Circ. Weld	WF-200 (Wire Ht. 821T44)	0.24	0.63	(IS) 1.92E+19
US to Lower Shell (LS) Circ. Weld (ID 75%)	WF-67 (Wire Ht. 72442)	0.26	0.60	(IS) 2.04E+19
LS to Dutchman Circ. Weld	WF-169-1 (Wire Ht. 8T1554)	0.16	0.57	(IS) 4.78E+17
Surry Unit 1, 80-Year Fluence (E > 1.0 MeV)				

Reactor Vessel Material	Material ID and/or Heat Number	Cu, wt%	Ni, wt%	(IS) Inside Wetted Surface Fluence or (*) clad/base metal n/cm^2 E> 1.0 MeV)
Nozzle Shell (NS) to Outlet Nozzle Forging Welds	SA-1493(Wire Ht. 8T1762)	0.19	0.57	(IS) 1.50E+18
	SA-1494(Wire Ht. 8T1554B)	0.16	0.57	(IS) 1.50E+18
NS to Inlet Nozzle Forging Welds	SA-1526 (Wire Ht. 299L44)	0.34	0.68	(IS) 1.50E+18
	SA-1580 (Wire Ht. 8T1762)	0.19	0.57	(IS) 1.50E+18
NS to Intermediate Shell (IS) Circ. Weld	J726 (Wire Ht. 25017)	0.33	0.10	(*) 7.98E+18
IS Long. Welds (Both)	SA-1494(Wire Ht. 8T1554)	0.16	0.57	(*)1.33E+19
IS to Lower Shell (LS) Circ. Weld (ID 40% 0	SA-1585 (Wire Ht. 72445)	0.22	0.54	(*)6.67E+19
IS to LS Circ. Weld (OD 60%)	SA-1650 (Wire Ht. 72445)	0.22	0.54	NA
LS Long. Weld (1)	SA-1494 (Wire Ht. 8T1554)	0.16	0.57	(*)1.34E+19
LS Long. Weld (2)	SA-1526 (Wire Ht. 299L44)	0.34	0.68	(*)1.34E+19
Surry Unit 2, 80-Year Fluence (E > 1.0 MeV)				
Nozzle Shell (NS) to Outlet Nozzle Forging Welds	Rotterdam	0.35[3]	1.0[3]	(IS) 1.50E+18
NS to Inlet Nozzle Forging Welds	WF-4 (Wire Ht. 8T1762)	0.19	0.57	(IS) 1.50E+18
	WF-8 (Wire Ht. 8T1762)	0.19	0.57	(IS) 1.50E+18
NS to Intermediate Shell (IS) Circ. Weld	L737 (Wire Ht. 4275)	0.35	0.10	(*) 9.21E+18
IS Long. Weld (1), and (2) (100% and OD 50%)	SA-1585 (Wire Ht. 72445)	0.22	0.54	(*) 1.36E+19

Reactor Vessel Material	Material ID and/or Heat Number	Cu, wt%	Ni, wt%	(IS) Inside Wetted Surface Fluence or (*) clad/base metal n/cm^2 E> 1.0 MeV)
IS Long. Weld (2) (ID 50%)	WF-4 (Wire Ht. 8T1762)	0.19	0.57	(*) 1.36E+19
IS to Lower Shell (LS) Circ. Weld	R3008 (Wire Ht. 0227)	0.187	0.545	(*) 7.67E+19
LS Long. Weld (Both)	WF-4 (Wire Ht. 8T1762)	0.19	0.57	(*) 1.37E+19
Turkey Point Unit 3, 80-Year Fluence (E > 1.0 MeV)				
US Forging to INF Welds	Heat 8T1762	0.19	0.57	(IS) 1.50E+18
	Heat 8T1554B	0.16	0.57	(IS) 1.50E+18
	Heat 71249	0.23	0.59	(IS) 1.50E+18
US Forging to ONF Welds	Heat 8T1762	0.19	0.57	(IS) 1.50E+18
US to IS Circ. Weld	SA-1484 Heat 72442	0.26	0.60	(*) 1.19E+19
IS to LS Circ. Weld	SA-1101 Heat 71249	0.23	0.59	(*) 1.04E+20
LS to Dutchman Circ. Weld	SA-1135 Heat 61782	0.23	0.52	(IS) 1.5E+18
Turkey Point Unit 4, 80-Year Fluence (E > 1.0 MeV)				
US to INF Welds	Heat 8T1762	0.19	0.57	(IS) 1.50E+18
	Heat 8T1554B	0.16	0.57	(IS) 1.50E+18
	Heat 299L44	0.34	0.68	(IS) 1.50E+18
US to ONF Welds	Heat 8T1554B	0.16	0.57	(IS) 1.50E+18
	Heat 299L44	0.34	0.68	(IS) 1.50E+18
US to IS Circ. Weld (ID 67%)	WF-67 Heat 72442	0.26	0.60	(*) 1.21E+19
IS to LS Circ. Weld	SA-1101 Heat 71249	0.23	0.59	(*) 1.03E+20
LS to Dutchman Circ. Weld	SA-1135 Heat 61782	0.23	0.52	(IS) 1.5E+18

Figure 3—1
Reactor Vessel—Oconee Unit 1



* Equivalent Margins Analysis performed for these Linde 80 Welds.

Figure 3—2
Reactor Vessel Shell Longitudinal Welds—Oconee Unit 1

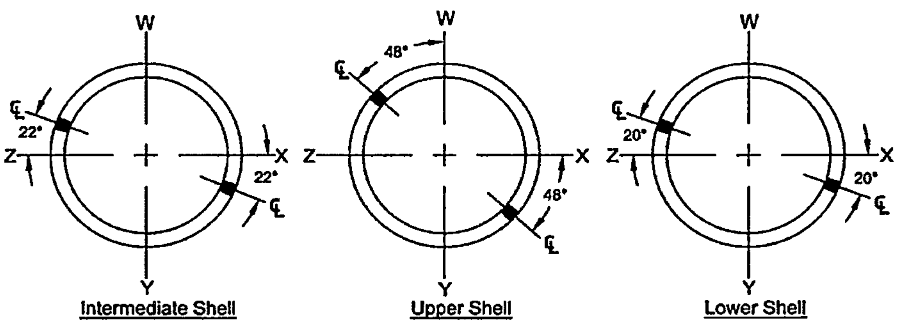
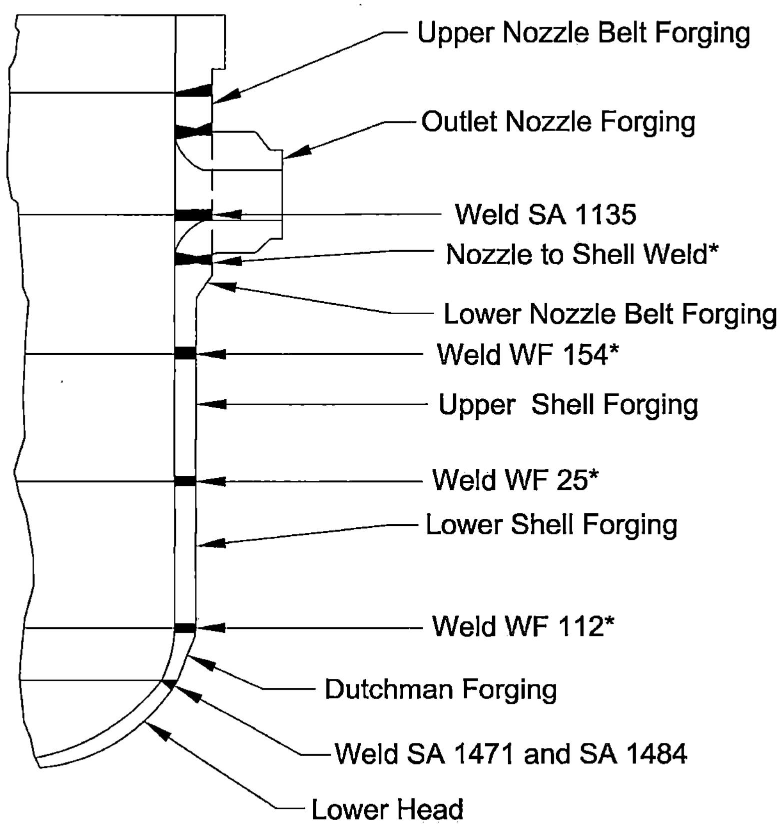
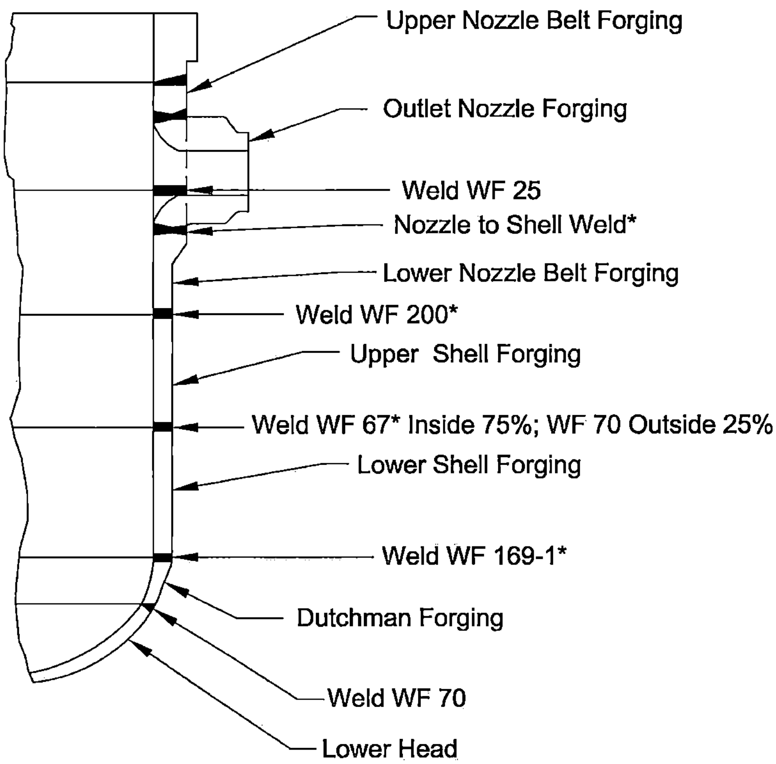


Figure 3—3
Reactor Vessel—Oconee Unit 2



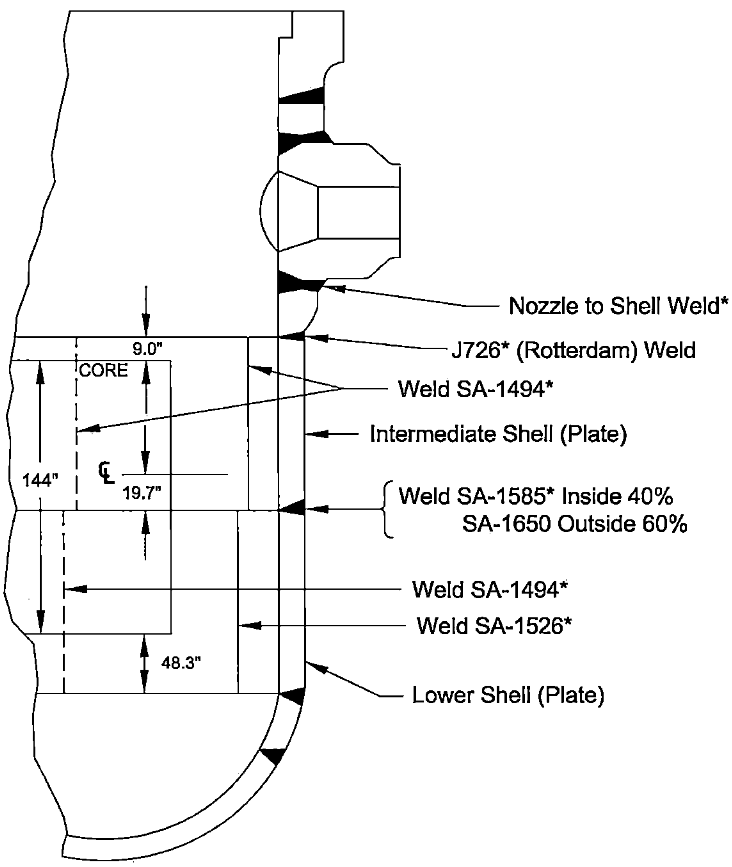
* Equivalent Margins Analysis
performed for these Linde 80 Welds.

Figure 3—4
Reactor Vessel—Oconee Unit 3



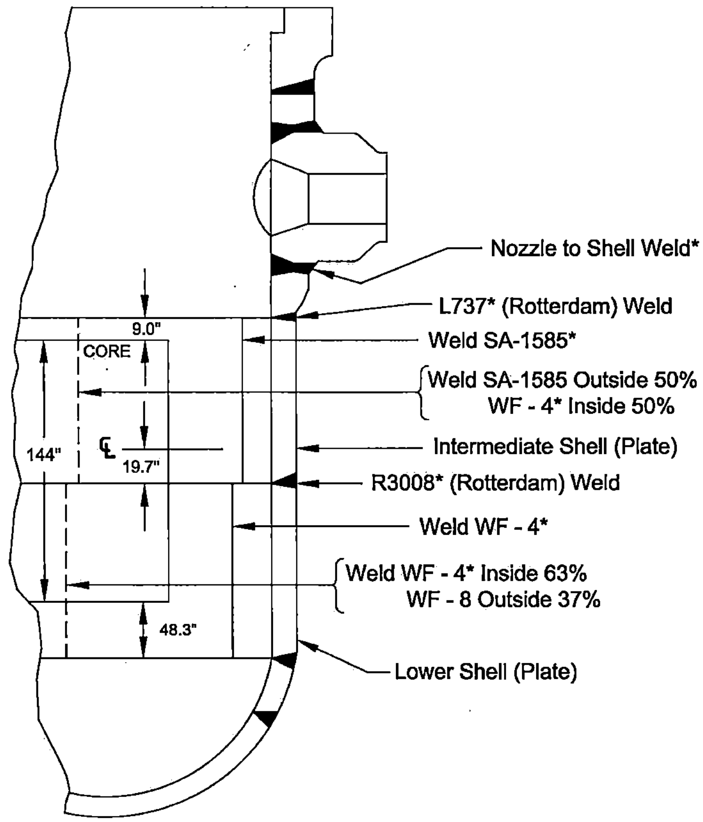
* Equivalent Margins Analysis
performed for these Linde 80 Welds.

Figure 3—5
Reactor Vessel—Surry Unit 1



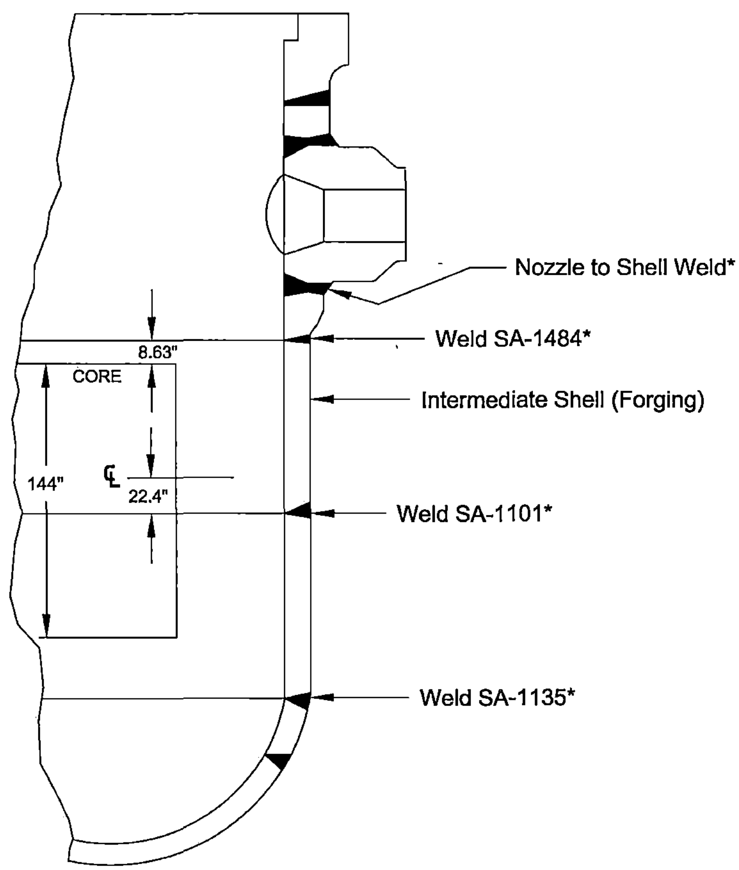
* Equivalent Margins Analysis performed for these Linde 80 and Rotterdam Welds.

Figure 3—6
Reactor Vessel—Surry Unit 2



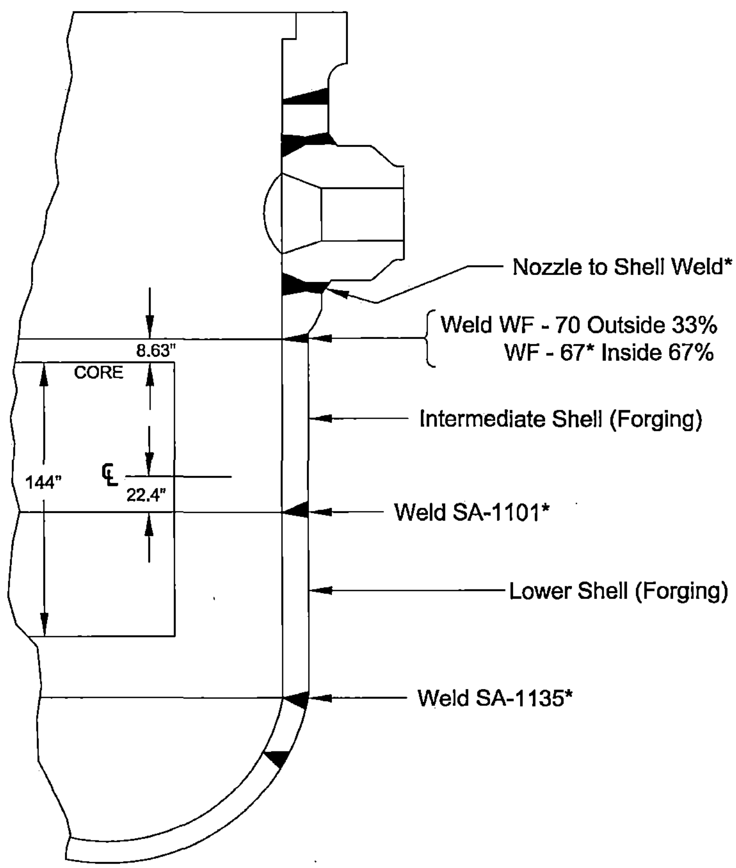
* Equivalent Margins Analysis performed for these Linde 80 and Rotterdam Welds.

Figure 3—7
Reactor Vessel—Turkey Point Unit 3



* Equivalent Margins Analysis performed for these Linde 80 Welds.

Figure 3—8
Reactor Vessel—Turkey Point Unit 4



* Equivalent Margins Analysis performed for these Linde 80 Welds.

4.0 MATERIAL PROPERTIES

4.1 *J-Integral Resistance Model*

The J-integral resistance model for Mn-Mo-Ni/Linde 80 welds in the reactor vessels of the B&WOG RVWG plants were developed using a large J-resistance data base. A detailed description of this model is provided in Appendix B of BAW-2192PA [1], Revision 00. This model was developed using specimens irradiated to 8.45E+18 n/cm^2, and the range of applicability of the model was extended (qualitatively) to approximately 1.90 E+19 n/cm^2 in Appendix B, Figure 3-1, to BAW-2251A [5]. See Appendix A of this report for a discussion of the extension of the range of applicability of the B&WOG J-R model to fluence values expected at 80 years for Oconee, Surry, and Turkey Point. Consistent with BAW-2192PA, Revision 00, this J-R model is used for Linde 80 welds and Rotterdam welds.

The coefficients a, d, and C4 are provided in Table 4-1. As required by ASME Section XI, ASME K-3300, when evaluating the vessel for Levels A, B, and C Service Loadings, the J-integral resistance versus crack-extension curve (J-R curve) shall be a conservative representation of the toughness of the controlling beltline material at upper shelf temperatures in the operating range. As such, the Jd correlation minus 2 standard errors is used for evaluation of Levels A & B service loadings (i.e., equation (1) multiplied by [ ]).

As discussed in Appendix B to BAW-2192PA, the J-R curve was generated from a J-integral database obtained from the same class of material with the same orientation as the applicable reactor vessel materials using correlations for the effects of temperature, chemical composition, and fluence level. Crack extension was by ductile tearing with no cleavage. This complies with the ASME Code, Section XI, K-3300.

Table 4—1
Parameters in Jd Model 4B

[ ]

4.2 *Mechanical Properties of Weld Metals*

The following subsections provide representative properties for the Oconee, Surry and Turkey Point reactor vessels. The temperature dependent mechanical properties are developed from the 2007 Edition with 2008 Addenda of the ASME Code (Section II, Part D) for the reactor base metal and cladding (the ASME Code does not provide separate mechanical properties for base and weld metal). Both ASME Code minimum and representative irradiated yield strengths are also provided. The mechanical properties such as weld metal yield strengths typically used were the irradiated properties but in some cases the ASME Code minimum properties were conservatively considered. The irradiated material properties used herein are consistent with those used for the plants 60-year license renewal low upper shelf toughness analysis submittals (See Section 1.1 above).

4.2.1 Mechanical Properties for the Oconee Reactor Vessels

The Oconee reactor vessels are fabricated using either A-508 Grade 2 Class 1 (3/4Ni-1/2Mo-1/3Cr-V) forging or A-533 Grade B Class 1 (Mn-1/2Mo-1/2Ni) plate Low Alloy Steel (LAS) materials. The stainless steel cladding material is considered to be fabricated from 18Cr-8Ni. Table 4-2 provides the Young's modulus (E), the mean coefficient of thermal expansion (α), and the yield strength (Sy) for the RV shell regions.

Table 4—2 Mechanical Properties of Oconee RV Shell Materials

	RV Base Metal			Weld Metal
Temp. (°F)	E (ksi)	α (in/in/°F)	Sy (ksi)	Sy (ksi)
70	27800	6.40E-06	50.0	[ ]
200	27100	6.70E-06	47.0	[ ]
300	26700	6.90E-06	45.5	[ ]
400	26200	7.10E-06	44.2	[ ]
500	25700	7.30E-06	43.2	[ ]
600	25100	7.40E-06	42.1	[ ]

For the RV shell and transition regions, the downcomer or steady state cold leg operating condition temperature is [ ] (Table 5-1). For the nozzle-to-shell welds the normal operating steady state condition temperature is [ ]. For the Linde 80 weld materials in the RV shell, transition and the nozzle welds of the Oconee reactor vessels, the measured yield strength value of weld metal SA-1585 is used. The yield strength value for this material is [ ]. This weld metal was used in the Oconee Unit 1 RV shell and represents the lowest yield strength value among the Linde 80 welds tested for three Oconee reactor vessels.

4.2.2 Mechanical Properties for the Surry Reactor Vessels

The Surry reactor vessels are fabricated using A-533 Grade B Class 1 (Mn-1/2Mo-1/2Ni) Low Alloy Steel (LAS) and stainless steel (18Cr-8Ni) cladding materials. Table 4-3 provides the Young's modulus (E), the mean coefficient of thermal expansion (α), and the yield strength (Sy) for the RV shell.

For the Surry reactor vessel shell welds, transition welds, and RV nozzle welds, the normal operating steady state condition cold leg temperature value is [ ] (Table 5-1). The yield strength value for SA-1526 at [ ] corresponding to a value of [ ] was used in the analysis.

Table 4—3 Mechanical Properties of Surry RV Shell Materials

	RV Base Metal			Weld Metal
Temp. (°F)	E (ksi)	α (in/in/°F)	Sy (ksi)	SA-1526 (ksi)
70	29000	7.00E-06	50.0	[ ]
200	28500	7.30E-06	47.0	[ ]
300	28000	7.40E-06	45.5	[ ]
400	27600	7.60E-06	44.2	[ ]
500	27000	7.70E-06	43.2	[ ]
600	26300	7.80E-06	42.1	[ ]

4.2.3 Mechanical Properties for the Turkey Point Reactor Vessels

The Turkey Point reactor vessels are fabricated using A-508 Grade 2 Class 1 (3/4Ni-1/2Mo-1/3Cr-V) Low Alloy Steel (LAS) and stainless steel (18Cr-8Ni) cladding materials. Table 4-4 provides the Young's modulus (E), the mean coefficient of thermal expansion (α), and the yield strength (Sy) for the RV shell regions.

Table 4—4 Mechanical Properties of Turkey Point RV Shell Materials

	RV Base Metal			Weld Metals	
Temp. (°F)	E (ksi)	α (in/in/°F)	Sy (ksi)	SA-1101 (ksi)	SA-1135 (ksi)
70	27800	6.40E-06	50.0	[ ]	[ ]
200	27100	6.70E-06	47.0	[ ]	[ ]
300	26700	6.90E-06	45.5	[ ]	[ ]
400	26200	7.10E-06	44.2	[ ]	[ ]
500	25700	7.30E-06	43.2	[ ]	[ ]
600	25100	7.40E-06	42.1	[ ]	[ ]

For the Turkey Point reactor vessel shell and RV nozzle regions, the normal operating steady state condition cold leg temperature value is [ ] (Table 5-1). Both the Turkey Point Units have SA-1101 and SA-1135 Linde 80 welds in the RV shell regions. The yield strength values for these weld metals at [ ], respectively.

5.0 FRACTURE MECHANICS ANALYSIS

5.1 *Methodology*

In accordance with ASME Section XI, Appendix K [16], Subarticle K-1200, the following analytical procedure was used for Levels A & B Service Loads.

a. The postulated flaws in the reactor vessel shell welds, the transition welds as well as RV nozzle-to-shell welds were postulated in accordance with the acceptance criteria of Subarticle K-2200.

b. Loading conditions at the locations of the postulated flaws were determined for Levels A and B Service Loadings. For Levels A and B Service loadings the equations to calculate the stress intensity factor (SIF) due to pressure and thermal gradients for a given pressure and cooldown rate are given in Article K-4210. Consistent with Section 5 of BAW-2192PA [1], the accumulation pressure is taken as ten percent above the design pressure and the maximum cooldown rate is 100°F/hr. In the area of the nozzle-to-shell weld, applied loadings consist of pressure, thermal, and attached piping reactions.

c. Material properties, including E, α, σ_y, and the J-integral resistance curve (J-R curve), were determined at the locations of the postulated flaws. Young's modulus, mean coefficient of thermal expansion and yield strength are addressed in Section 4.2. The J-R curve is discussed in Section 4.1.

d. The postulated flaws were evaluated in accordance with the acceptance criteria of Article K-2000. Requirements for evaluating the applied J-integral are provided in Subarticle K-3200, and for determining flaw stability in Subarticle K-3400. Subarticle K-3500(a) invokes the procedure provided in Subarticle K-4200 (K-4220) for evaluating the applied J-integral for a specified amount of ductile flaw extension. Three permissible evaluation methods to address flaw stability are described in Subarticle K-3500(b). The evaluation method selected herein is the J-R curve crack driving force diagram procedure described in Subarticle K-4310.

5.2 *Procedure for Evaluating Levels A and B Service Loadings*

For RV shell regions remote from structural discontinuities, the applied *J*-integral is calculated in accordance with Appendix K, Subsubarticle K-4210, 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, as outlined below.

(1) For an axial flaw of depth *a*, the stress intensity factor due to internal pressure is calculated with a structural factor (*SF*) on pressure using the following:

$$K_{Ip} = (SF)p\left(1+\frac{R_i}{t}\right)(\pi a)^{0.5}F_1$$

where

$$F_1 = 0.982 + 1.006\left(\frac{a}{t}\right)^2, \qquad 0.20 \le \left(\frac{a}{t}\right) \le 0.50$$

(2) For a circumferential flaw of depth *a*, the stress intensity factor due to internal pressure is calculated with a structural factor (*SF*) on pressure using the following:

$$K_{Ip} = (SF)p\left(1+\frac{R_i}{2t}\right)(\pi a)^{0.5} F_2$$

where

$$F_2 = 0.885 + 0.233\left(\frac{a}{t}\right) + 0.345\left(\frac{a}{t}\right)^2, \quad 0.20 \le \left(\frac{a}{t}\right) \le 0.50$$

(3) For an axial or circumferential flaw of depth a, the stress intensity factor due to radial thermal gradients are calculated using the following:

$$K_{It} = C_m(CR)t^{2.5}F_3, \qquad 0 \le (CR) \le 100^{\circ}\text{F/hr}$$

where for SA-508, Class 2 or SA-533, Grade B, Class1 steels the material coefficient C_m is defined as:

$$C_m = \frac{E\alpha}{(1-\nu)d} = 0.0051,$$

CR = cooldown rate (°F/hr), and

$$F_3 = 0.1181 + 0.5353\left(\frac{a}{t}\right) - 1.273\left(\frac{a}{t}\right)^2 + 0.6046\left(\frac{a}{t}\right)^3, \quad 0.20 \le \left(\frac{a}{t}\right) \le 0.50$$

(4) The effective flaw depth for small scale yielding, a_e, is calculated using the following:

$$a_e = a + \left(\frac{1}{6\pi}\right)\left[\frac{K_{Ip} + K_{It}}{\sigma_y}\right]^2$$

(5) For an axial flaw of depth a_e, the stress intensity factor due to internal pressure is:

$$K_{Ip}^{'} = (SF)p\left(1+\frac{R_i}{t}\right)(\pi a_e)^{0.5}F_1^{'}$$

where

$$F_1^{'} = 0.982 + 1.006\left(\frac{a_e}{t}\right)^2, \qquad 0.20 \le \left(\frac{a_e}{t}\right) \le 0.50$$

(6) For a circumferential flaw of depth a_e, the stress intensity factor due to internal pressure is:

$$K_{Ip}^{'} = (SF)p\left(1+\frac{R_i}{2t}\right)(\pi a_e)^{0.5}F_2^{'}$$

where

$$F_2^{'} = 0.885 + 0.233\left(\frac{a_e}{t}\right) + 0.345\left(\frac{a_e}{t}\right)^2, \qquad 0.20 \le \left(\frac{a_e}{t}\right) \le 0.50$$

(7) For an axial or circumferential flaw of depth a_e, the stress intensity factor due to radial thermal gradients is:

$$K_{It}^{'} = C_m(CR)t^{2.5}F_3^{'}, \qquad 0 \le (CR) \le 100^{\circ}\mathrm{F/hr}$$

where

$$F_3^{'} = 0.1181 + 0.5353\left(\frac{a_e}{t}\right) - 1.273\left(\frac{a_e}{t}\right)^2 + 0.6046\left(\frac{a_e}{t}\right)^3, \quad 0.20 \le \left(\frac{a_e}{t}\right) \le 0.50$$

(8) The *J*-integral due to applied loads for small scale yielding is calculated using the following:

$$J_1 = 1000 \frac{\left(K'_{Ip} + K'_{It}\right)^2}{E'}$$

where

$$E' = \frac{E}{1-\nu^2}$$

(9) Evaluation of upper-shelf toughness at a flaw extension of 0.10 in. is performed for a flaw depth,

$$a = 0.25t + 0.10\text{in.},$$

using

$$SF = 1.15$$

$$p = P_a$$

where P_a is the accumulation pressure for Levels A and B Service Loadings, such that

$$J_1 < J_{0.1}$$

where

J_1 = the applied J-integral for a safety factor of 1.15 on pressure, and a safety factor of 1.0 on thermal loading

$J_{0.1}$ = the lower bound J-integral resistance at a ductile flaw extension of 0.10 in.

(10) Evaluation of flaw stability is performed through use of a crack driving force diagram procedure, by comparing the slopes of the applied J-integral

curve and the lower bound *J-R* curve. The applied *J*-integral is calculated for a series of flaw depths corresponding to increasing amounts of ductile flaw extension. The applied pressure is the accumulation pressure for Levels A and B Service Loadings, P_a, and the safety factor (*SF*) on pressure is 1.25. Flaw stability at a given applied load is verified when the slope of the applied *J*-integral curve is less than the slope of the *J-R* curve at the point on the *J-R* curve where the two curves intersect.

For the Oconee reactor vessels, the effect of structural discontinuity at the transition welds (i.e., "upper weld" at Unit 1 connecting upper end of the intermediate shell to the lower nozzle belt forging & "lower weld" connecting the lower end of the lower shell to the Dutchman forging) is accounted for by applying a scaling factor of [ ] to the applied J-integral calculated by the above procedure for circumferential welds. This approach is based on a previous finite element analysis of an applicable axisymmetric B&W-designed reactor vessel shell model that included a detailed description of the transition regions. The stresses due to pressure and thermal loads at the nozzle-to-shell welds are obtained from a previous axisymmetric analysis of the outlet nozzle of a B&W-designed plant that is also deemed applicable to the Oconee reactor vessels.

For both the Surry and Turkey Point reactor vessels, the applied J-integrals at the nozzle-to-shell welds and the upper transition welds were determined using stresses from a detailed three-dimensional finite element analysis. Path line stresses were used to determine applied J-integrals that included a plastic zone correction to account for small scale yielding. Based on the results of analysis performed for Oconee it was deemed that the effects of structural discontinuities at the lower transition welds need not be explicitly addressed.

5.3 *Evaluation for Flaw Extension*

The applied J-integrals for the RV shell welds, the RV transition welds, and the RV nozzle welds are calculated as discussed in Section 5.2.

5.3.1 Reactor Vessel Shell Welds

The basic reactor vessel shell geometry and design pressure along with operating condition temperature information for each of the Oconee, Surry and Turkey Point reactor vessels (also referred to as three groups of reactor vessels) is provided in Table 5-1. Initial flaw depths equal to ¼ of the vessel wall thickness are analyzed for Levels A and B service loadings following the procedure outlined in Section 5.2 and evaluated for acceptance based on values for the J-integral resistance of the material from the Linde 80 J-R model discussed in Section 4.1. For each of the three groups of reactor vessels, calculations are initially carried out to identify the controlling weld such that subsequent detailed low upper shelf toughness flaw evaluations can be performed using the controlling weld.

The results of the plant specific flaw evaluations for each of the RV shell welds of the three groups of reactor vessels are presented in Table 5-2. From the results of the evaluation in Table 5-2, the controlling RV shell welds for each of the three groups of reactor vessels can be observed. The controlling welds are determined by noting the minimum ratio of the material J-resistance ($J_{0.1}$) to the applied J-integral (J_1) (also referred to as "margin") for each of the three groups of reactor vessels.

For the Oconee reactor vessels, the controlling RV shell weld is the SA-1073 longitudinal weld of Oconee Unit 1. This weld is located in the intermediate shell. The minimum ratio of material J-resistance to applied J-integral ($J_{0.1}/J_1$) or margin is [ ], which is higher than the minimum acceptable value of 1.0.

For the Surry reactor vessels, the controlling RV shell weld is the SA-1526 longitudinal weld of Surry Unit 1. This weld is located in the lower shell. The minimum margin ($J_{0.1}/J_1$) is [ ], which is higher than the minimum acceptable value of 1.0.

For the Turkey Point reactor vessels, the controlling RV shell weld is the circumferential weld SA-1101 which is located between the intermediate and lower shell courses of both Units 3 and 4 reactor vessels. The minimum ratio of material J-resistance to applied J-integral ($J_{0.1}/J_1$) is **[ ]** , which is significantly higher than the minimum acceptable value of 1.0.

5.3.2 Reactor Vessel Transition Welds and RV Nozzle Welds

The reactor vessel nozzle welds are located in the substantially thicker cylindrical shell section (reinforced to account for the inlet/outlet RV nozzle openings and typically referred to as the nozzle belt), located above the reactor vessel shell welds. The reactor vessel nozzle belt dimensions are reported in Table 5-3.

For the Oconee reactor vessels there is only one upper transition weld: SA-1135 for Oconee Unit 1 (see Figures 3-1 through 3-4). This circumferential weld is addressed in Table 5-2. The calculated ratio of $J_{0.1}/J_1$ is **[ ]** . The flaw evaluations for the Oconee lower transition welds are summarized in Table 5-4. The minimum ratio of material J-resistance to applied J-integral ($J_{0.1}/J_1$) is **[ ]** , which is significantly higher than the minimum acceptable value of 1.0. The flaw evaluation of the RV nozzle-to-shell weld are summarized in Table 5-5. The applied stress intensity factor is conservatively calculated for this location in terms of tension and bending loads using a flat plate solution by Newman and Raju. The calculated $J_{0.1}/J_1$ ratios are **[ ]** and **[ ]** , respectively for flaws oriented in the axial and circumferential direction with respect to the RV, which are higher than the minimum acceptable value of 1.0.

For the Surry reactor vessels upper RV transition and RV nozzle weld regions, the J-applied results with the safety factor of 1.15 on applied pressure are compared against the lower bound J-integral resistance at a ductile flaw extension of 0.1 inches ($J_{0.1}$) in Table 5-6. The most limiting ratio or margin ($J_{0.1}/J_1$) is [ ] due to postulating an axial flaw in the longitudinal weld SA-1585 located near the base of the taper transition section. Weld SA-1585 has the highest copper content of any longitudinal weld in the intermediate shells at Surry. The fluence value corresponding to the base of the taper transition section was utilized. The lower transition welds are located approximately four feet below the bottom of the core and are predicted to receive less than the threshold fluence value of 1.0E17 n/cm^2 at 80-years As such, these lower transition welds were not evaluated in this report.

For the Turkey Point reactor vessels upper RV transition and RV nozzle weld regions, the J-applied results with the safety factor of 1.15 on applied pressure is compared against the lower bound J-integral resistance at a ductile flaw extension of 0.1 inches ($J_{0.1}$) in Table 5-7. The limiting item in this case is the inlet nozzle with a margin of [ ]. For the lower transition weld SA-1135, it is noted that this circumferential weld is located at the thickness transition between the cylindrical shell and the thinner lower head. This location was additionally evaluated using the dimensions applicable to the lower head [ ] since this will result in higher pressure stresses but lower thermal stresses when compared against the thicker RV cylindrical shell. For evaluation of the SA-1135 weld, the margin reduces from [ ] (considering the thicker RV shell as shown in Table 5-2) to [ ] (as given in Table 5-8) when considering the thinner lower head. Since the reduced margin is still significantly larger than 1.0, this simplified analytical approach is deemed to be an acceptable means of addressing the structural discontinuity at the lower transition weld.

5.4 *Evaluation for Flaw Stability*

The flaw stability analysis is performed by calculating the applied J-integrals for various amounts of flaw extension with a safety factor (on pressure) of 1.25. The resulting applied J-integral curve can then be compared against the lower bound J-R curve for the weld metal. It is noted that applied J-integrals are also calculated with a safety factor on pressure of 1.15 for illustration of the $J_{0.1}/J_1$ margin with respect to the lower bound J-R curve at a flaw extension of 0.1 inch.

5.4.1 Reactor Vessel Shell Welds

For the Oconee reactor vessels, the applied J-integral values of the controlling weld (SA-1073) is calculated for various amounts of flaw extension with safety factors (on pressure) of 1.15 and 1.25 in Table 5-9. The corresponding mean and lower bound *J-R* curve values are calculated and shown in Table 5-10. The resulting J-applied curves are compared against the lower bound *J-R* curve for the SA-1073 material in Figure 5-1. An evaluation line at a flaw extension of 0.1 inch is included to confirm the results of Table 5-2 by showing the margin between the applied J-integral with the safety factor of 1.15 and the lower bound J-integral resistance of the material. The requirement for ductile and stable crack growth is demonstrated by Figure 5-1 since the slope of the applied J-integral curve for a safety factor of 1.25 is less than slope of the lower bound *J-R* curve at the point where the two curves intersect.

The applied J-integral values for the controlling weld of the Surry reactor vessels (SA-1526) is similarly calculated and shown in Table 5-11 with the corresponding mean and lower bound *J-R* curve values shown in Table 5-12. The resulting J-applied curves are then compared against the lower bound J-R curve for this material in Figure 5-2. An evaluation line at a flaw extension of 0.1 inch is included to confirm the results of Table 5-2 by showing the margin between the applied J-integral with the safety factor of 1.15 and the lower bound J-integral resistance of the material. The requirement for ductile and stable crack growth is demonstrated by Figure 5-2 since the slope of the applied J-integral curve for a safety factor of 1.25 is less than slope of the lower bound *J-R* curve at the point where the two curves intersect.

Similarly, for the controlling SA-1101 weld material of Turkey Point Units 3 and 4, the applied J-integral values are calculated and shown in Table 5-13 with the corresponding mean and lower bound *J-R* curve values shown in Table 5-14. The resulting J-applied curves are then compared against the lower bound J-R curve for this material in Figure 5-3. An evaluation line at a flaw extension of 0.1 inch is included to confirm the results of Table 5-2 by showing the margin between the applied J-integral with the safety factor of 1.15 and the lower bound J-integral resistance of the material. The requirement for ductile and stable crack growth is demonstrated by Figure 5-3 since the slope of the applied J-integral curve for a safety factor of 1.25 is less than slope of the lower bound *J-R* curve at the point where the two curves intersect.

5.4.2 Reactor Vessel Transition Welds and RV Nozzle Welds

As discussed in Section 5.3.2, the controlling weld for the Oconee RV transition welds and the RV nozzle welds is the RV nozzle-to-shell weld. The results of the flaw evaluation of this weld is provided in Table 5-5 with a margin of [ ] and [ ], respectively for stresses in the circumferential (applicable for axial flaw) and longitudinal directions (applicable for circumferential flaw). The margins and flaw stability considering both flaw orientations are shown in Figure 5-4. Both the mean and the lower bounding *J-R* curves are illustrated in Figure 5-4. An evaluation line at a flaw extension of 0.1 inch is included to confirm the results of Table 5-5 by showing the margin between the applied J-integral with the safety factor of 1.15 and the lower bound J-integral resistance of the material. The requirement for ductile and stable crack growth is demonstrated by Figure 5-4 since the slope of the applied J-integral curve for a safety factor of 1.25 is less than slope of the lower bound *J-R* curve at the point where the two curves intersect.

For the Surry reactor vessels, the controlling weld and flaw orientation is the axial flaw in the longitudinal weld SA-1585 of Surry Unit 2 as discussed previously in Section 5.3.2. The applied J-integral for this axial flaw with a safety factor of 1.25 on pressure at various flaw extensions is plotted with the lower bound J-resistance curve (mean *J-R* curve provided for information only) in Figure 5-5. The slope of the applied J-integral is less than the slope of the lower bound J-resistance curve at the point of intersection, which demonstrates that the flaw is stable as required by ASME Section XI, Appendix K.

As discussed in Section 5.3.2 and shown in Table 5-7, the limiting location for the Turkey Point reactor vessels is the inlet nozzle with a margin of [ ]. The applied J-integral for the inlet nozzle with a safety factor of 1.25 on pressure at various flaw extensions is plotted with the lower bound J-resistance curve in Figure 5-6. The slope of the applied J-integral is less than the slope of the lower bound J-resistance curve at the point of intersection, which demonstrates that the flaw is stable as required by ASME Section XI, Appendix K.

Table 5—1
Reactor Vessel Shell Dimensions and Operating Conditions

[ ]

Table 5—2 Plant Specific Flaw Evaluation Summary for RV Shell Regions

[ ]

Table 5—3 Reactor Vessel Nozzle Belt Dimensions

[ ]

Table 5—4 Flaw Evaluation Summary of Lower Transition Welds of Oconee Units

[ ]

Table 5—5 Flaw Evaluation Summary of Oconee RV Nozzle-to-Shell Weld

[ ]

Table 5—6 Flaw Evaluation Summary of Surry Upper Transition and RV Nozzle-to-Shell Welds

[ ]

Table 5—7 Flaw Evaluation Summary of Turkey Point Upper Transition and RV Nozzle-to-Shell Welds

[ ]

Table 5—8 Flaw Evaluation Summary of Turkey Point Lower Transition Welds

[ ]

Table 5—9 Applied J-Integral versus Flaw Extensions of Oconee Controlling RV Shell Weld (SA-1073)

Table 5—10 Mean & Lower Bound J-R Curve Values for Oconee Controlling RV Shell Weld (SA-1073)

[ ]

Table 5—11 Applied J-Integral versus Flaw Extensions of Surry Controlling RV Shell Weld (SA-1526)

[ ]

Table 5—12 Mean & Lower Bound J-R Curve Values for Surry Controlling RV Shell Weld (SA-1526)

[ ]

Table 5—13 Applied J-Integral versus Flaw Extensions of Turkey Point Controlling RV Shell Weld (SA-1101)

[ ]

Table 5—14 Mean & Lower Bound J-R Curve Values for Turkey Point Controlling RV Shell Weld (SA-1101)

[ ]

Figure 5—1
J-Integral versus Flaw Extension for ONS -1, 2, 3 Controlling Reactor Vessel Shell Weld (SA-1073)

[

]

Figure 5—2
J-Integral versus Flaw Extension for Surry 1 and 2 Controlling Reactor Vessel Shell Weld (SA-1526)

[ ]

Figure 5—3
J-Integral Flaw Extension for Turkey Point Controlling Reactor Vessel Shell weld (SA-1101)

[ ]

Figure 5—4 J-Integral versus Flaw Extension for Oconee RV Nozzle-to-Shell Weld

[

]

Figure 5—5 J-Integral versus Flaw Extension for Surry Axial Weld (SA-1585) Near Transition

Figure 5—6 J-Integral versus Flaw Extension for Turkey Point Inlet Nozzle

[ ]

6.0 SUMMARY AND CONCLUSIONS

6.1 *Reactor Vessel Shell Welds*

The ASME Section XI, acceptance criteria for Levels A & B Service Loads for all reactor vessel shell welds are satisfied. The results of the limiting welds for Oconee Units 1, 2, and 3, Surry Units 1 and 2, and Turkey Point Units 3 and 4 are reported below.

<u>**Oconee Units 1, 2, and 3**</u>

- The limiting RV shell weld is Oconee Unit 1 axial weld SA-1073. With factors of safety of 1.15 on pressure and 1.0 on thermal loading, the applied J-integral (J1) is less than the J-integral of the material at a ductile flaw extension of 0.10 in. (J0.1)—(Figure 5-1). The ratio J0.1/J1 = [ ] is greater than the required value of 1.0.
- 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 (Figure 5-1).

<u>**Surry Units 1 and 2**</u>

- The limiting RV shell weld is Surry Unit 1 axial weld SA-1526. With factors of safety of 1.15 on pressure and 1.0 on thermal loading, the applied J-integral (J1) is less than the J-integral of the material at a ductile flaw extension of 0.10 in. (J0.1)—(Figure 5-2). The ratio J0.1/J1 = [ ] is greater than the required value of 1.0.
- 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 (Figure 5-2).

Turkey Point Units 3 and 4

- The limiting RV shell weld is TP 3 and 4 circumferential weld SA-1101. With factors of safety of 1.15 on pressure and 1.0 on thermal loading, the applied J-integral (J1) is less than the J-integral of the material at a ductile flaw extension of 0.10 in. (J0.1)—(Figure 5-3). The ratio J0.1/J1 = [ ] is greater than the required value of 1.0.
- 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 (Figure 5-3).

6.2 *Reactor Vessel Transition Welds and RV Nozzle Welds*

The acceptance criteria for Levels A & B Service Loads for RV transition welds and RV nozzle welds are satisfied. The results of the limiting weld considering transition welds and RV nozzle welds (inlet and outlet) for Oconee, Surry, and Turkey Point are reported below.

Oconee Units 1, 2, and 3

- The limiting weld for the Oconee Units is the RV outlet nozzle-to-shell weld. With factors of safety of 1.15 on pressure and 1.0 on thermal loading, the applied J-integral (J1) is less than the J-integral of the material at a ductile flaw extension of 0.10 in. (J0.1)—(Figure 5-4). The ratio J0.1/J1 = [ ] and 1.61, respectively, for axial and circumferential flaws.
- 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 (Figure 5-4).

Surry Units 1 and 2

- The limiting weld for the Surry Units 1 and 2 is the longitudinal weld SA-1585 near the base of the transition section. With factors of safety of 1.15 on pressure and 1.0 on thermal loading, the applied J-integral (J1) is less than the J-integral of the material at a ductile flaw extension of 0.10 in. (J0.1)—(Figure 5-5). The ratio J0.1/J1 = [ ] is greater than the required value of 1.0. This is due to postulating an axial flaw in the longitudinal weld of Surry Unit 2 while using the highest copper content of any intersecting longitudinal weld in the intermediate shell.
- 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 (Figure 5-5).

Turkey Point Units 3 and 4

- The limiting weld for the Turkey Point Units 3 and 4 RV transition welds (upper and lower) and the RV nozzle welds is the RV inlet nozzle-to-shell weld. With factors of safety of 1.15 on pressure and 1.0 on thermal loading, the applied J-integral (J1) is less than the J-integral of the material at a ductile flaw extension of 0.10 in. (J0.1)—(Figure 5-6). The ratio J0.1/J1 = [ ] is greater than the required value of 1.0.
- 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 (Figure 5-6).

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8.0 CERTIFICATION

This report is an accurate description of the low upper-shelf toughness fracture analysis of ONS Units 1, 2 and 3, Surry Units 1 and 2, and Turkey Point Units 3 and 4vessels.

Ashok Nana/Mark Rinckel
Component Analysis and Fracture Mechanics Unit/Nuclear Analysis Unit

This report has been reviewed and is an accurate description of the low upper-shelf toughness fracture analysis of reactor vessels of ONS Units 1, 2 and 3, Surry Units 1 and 2, and Turkey Point Units 3 and 4.

D.E. Killian
Component Analysis and Fracture Mechanics Unit

Verification of independent review.

David Cofflin
Component Analysis and Fracture Mechanics Unit

This report is approved for release.

Danielle ~~Paige~~ Blair Page MAL 12/15/17
NSSS Project Management

APPENDIX A
B&WOG J-R MODEL-DATA ANALYSIS AND EMPIRICAL MODEL DEVELOPMENT

A.1 Background

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Table A—1 Model 4B, Range of Test Data

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Figure A—1 BAW-2251A, Appendix B, Figure 3-1

A.2 New B&WOG J–Δa Data and Comparison to B&WOG J-R Model

Figure A—2 Jd (0.1) vs Fluence B&WOG J-R Model 4B and New Test Data (Normalized to Standard Conditions)

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A.3 New B&WOG J-R Model

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Table A—3 Jd Model Coefficients (Models 4B, 5B, and 6B)

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Figure A—3 Original and New Data and Model Fit Normalized at Standardized Conditions vs Δa

Figure A—4 Original and New Data and Model Fit Normalized at Standard Conditions vs Fluence

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Figure A—5 Model 6B Residuals vs Fitted Values

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Figure A—6 Model 6B Standardized Residuals vs Fitted Values

Figure A—7 Normal Q-Q Plot of Standardized Residuals

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A.4 Model 6B Reconciliation to EMA Results Presented in Section 6.0

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Figure A—8 Comparison of Models 4B, 5B, and 6B at Standard Conditions

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Table A—4 EMA Reconciliation for Limiting RV Shell Welds— Models 4B and 6B

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A.5 Summary and Conclusions