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

BAW-2178NP
Supplement 1
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Nature of Changes

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

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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
ONF	Outlet Nozzle Forging
ONS	Oconee Nuclear Station
PTN	Turkey Point Plant
RV	Reactor Vessel
RVWG	Reactor Vessel Working Group
SLR	Subsequent License Renewal
Sy	Yield Strength
TSs	Technical Specifications
USE	Upper Shelf Energy

ABSTRACT

This Supplement 1 to BAW-2178PA reports an equivalent margins analysis (EMA) considering Levels C and D 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-2178 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 C & D Service Loads for all reactor vessel shell welds are satisfied. The acceptance criteria for Levels C & D Service Loads for RV transition welds and RV nozzle welds are also satisfied. Consistent with BAW-2178PA, Revision 0, B&WOG J-R Model 4B is used for Linde 80 welds and Rotterdam welds.

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 at a crack extension of 0.1 inches is estimated at 6.4E 19 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 BAW-2192, Supplement 1, 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 BAW-2192, Supplement 1, Appendix A.

1.0 INTRODUCTION

The purpose of Supplement 1 to BAW-2178PA [1] is to report an equivalent margins analysis (EMA) considering Levels C and D 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-2178PA 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).

The EMA utilizes the B&WOG J-integral resistance (J-R) Model 4B reported in BAW-2192PA, Appendix B. Justification for use of Model 4B for 80-year fluence is addressed in BAW-2192, Supplement 1, Appendix A [3].

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). Plants that reference this report must calculate 80-year neutron fluence at reactor vessel weld locations in accordance with the requirements of NUREG-2192 [4], 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 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 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 [5] “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 [19] 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 applicable reactor vessel welds 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, and Section 7.0 lists all references. BAW-2192, Supplement 1, Appendix A [3] provides the technical justification for the use of B&WOG J-R Model 4B for the EMA reported herein.

1.1 *Equivalent Margins Analysis—Analysis of Record*

BAW-2178PA, Revision 00 [1] provided the EMA analysis of record for Levels C and D 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 [6], 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.

Oconee Units 1, 2, and 3

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 [7]. 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 [8]. 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 ONS License Renewal Application [9], which referenced AREVA report BAW-2251-A [6], Appendix B, for the ONS 48 EFPY equivalent margins analysis. The NRC performed an equivalent margins analysis reconciliation for the ONS-1, 2, and 3 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 [7]. 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 [10]. 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) [11]. Specifically, Attachment 4 to Reference [11] 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 [12]. 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 [13]
- 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 [14]. Reference [14] 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 [15]. Reference [15] 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 [5], 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 [16], 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 design transients and selection of material properties are addressed in Sections 3.0 and 4.0.

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 [17] 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 [18]. 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. At the limiting time points in the Level C & D analysis, where cladding effects are included, the temperature of the cladding is well below 600°F, and thus this change does not impact the low upper shelf toughness analysis reported herein.

2.2.1 Acceptance Criteria Levels C and D

ASME Section XI [17], Subarticles K-2300 and K-2400, provide acceptance criteria for Levels C and D Service Conditions. Consistent with BAW-2178PA [1], the evaluations reported herein will utilize acceptance criteria applicable to Level C Service Loadings as summarized below.

a) When evaluating adequacy of the upper shelf toughness for the weld material for Level C Service Loadings, interior semi-elliptical surface flaws with depths up to 1/10 of the base metal wall thickness, plus the cladding thickness, with total depths not exceeding 1 in. (25 mm), and a surface length 6 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 shall be postulated, and toughness properties for the corresponding orientation shall be used. Flaws of various depths, ranging up to the maximum postulated depth, shall be analyzed to determine the most limiting flaw depth. Smaller maximum flaw sizes may be used when justified. Two criteria shall be satisfied:

    1. The applied J-integral shall be less than the J-integral of the material at a ductile flaw extension of 0.10 in. (2.5 mm), using a structural factor of 1 on loading.

    2. Flaw extensions shall be ductile and stable, using a structural factor of 1 on loading.

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

For the Level C and D transients defined in Section 4.3.1 for the Oconee reactor vessels, the evaluations reported herein will conservatively utilize the Level C acceptance criteria given above as a means to treat the transients as a combined Levels C & D event category. The above Level C acceptance criteria will be conservatively imposed on the Level D transients defined in Section 4.3.2 of for the Surry and Turkey Point reactor vessels. In addition, for information purposes only, the acceptance criteria applicable to the Level D Service Loadings as summarized below will be reported for Level D transients.

a) When evaluating adequacy of the upper shelf toughness for Level D Service Loadings, flaws as specified for Level C Service Loadings shall be postulated, and toughness properties for the corresponding orientation shall be used. Flaws of various depths, ranging up to the maximum postulated depth, shall be analyzed to determine the most limiting flaw depth. Flaw extensions shall be ductile and stable, using a factor of safety of 1.0 on loading.

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

c) The extent of stable flaw extension shall be less than or equal to 75% of the vessel wall thickness, and the remaining ligament shall not be subject to tensile instability.

3.0 DESCRIPTION OF OCONEE, SURRY AND TURKEY POINT REACTOR VESSELS

The Oconee, Surry and Turkey Point reactor vessels with 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. 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). 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.

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 analyses of record or a search of ONS, Surry, and Turkey Point reactor vessel fabrication reports.

The dimensions of the reactor vessel shell geometry for the Oconee, Surry and Turkey Point reactor vessels are provided in Table 3-2. Similarly, the dimensions for the reactor vessel nozzle belt region located above the reactor vessel shell course for each of these three groups of reactor vessels are given in Table 3-3.

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)				
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

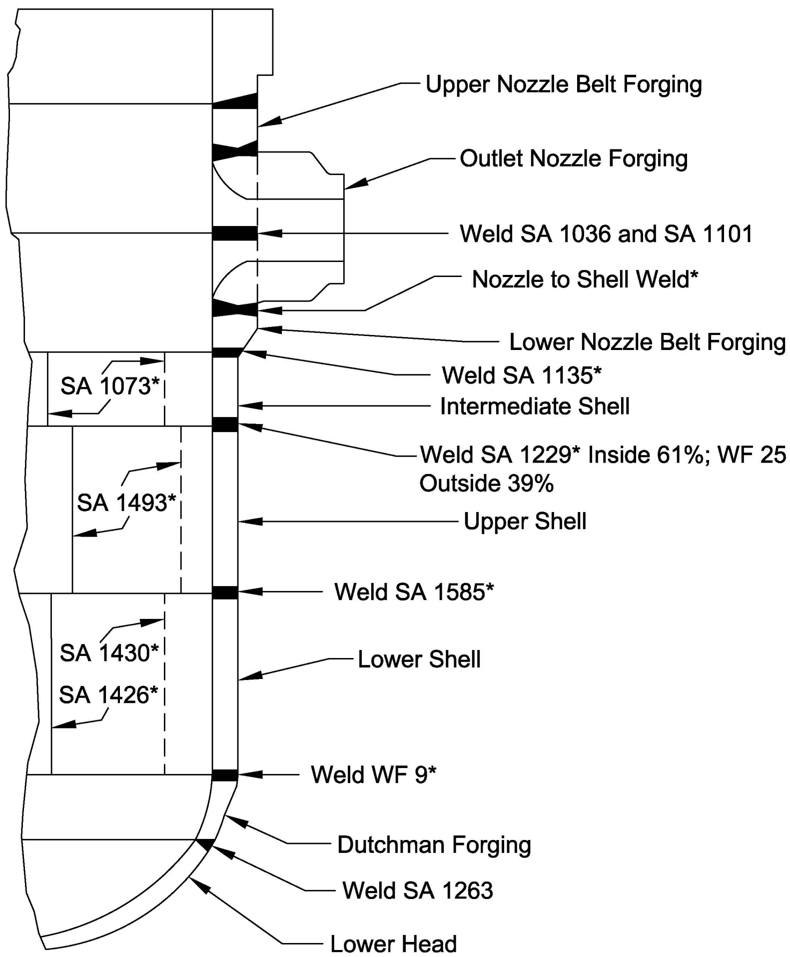
Reactor Vessel Material	Material ID and/or Heat Number	Cu, wt%	Ni, wt%	(IS) Inside Wetted Surface Fluence or (*) clad/base metal n/cm^2 E> 1.0 MeV)
LNB to Inlet Nozzle Forging (INF) Welds	Wire Ht. 72105	0.32	0.58	(IS) 1.50E+18
	Wire Ht. 72102	0.21	0.58	(IS) 1.50E+18
	Wire Ht. 82102 (Note 3)	0.35	1.00	(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)				
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%)	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	(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
IS Long. Weld (2) (ID 50%)	WF-4 (Wire Ht. 8T1762)	0.19	0.57	(*) 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 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

Table 3—2 Reactor Vessel Shell Dimensions

Table 3—3 Reactor Vessel Nozzle Belt Dimensions

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

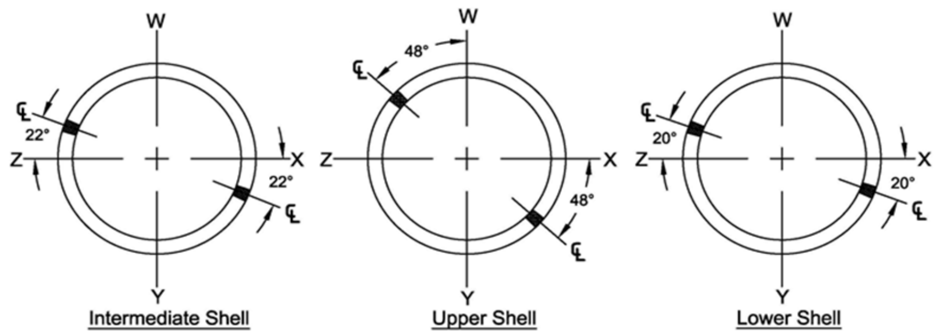
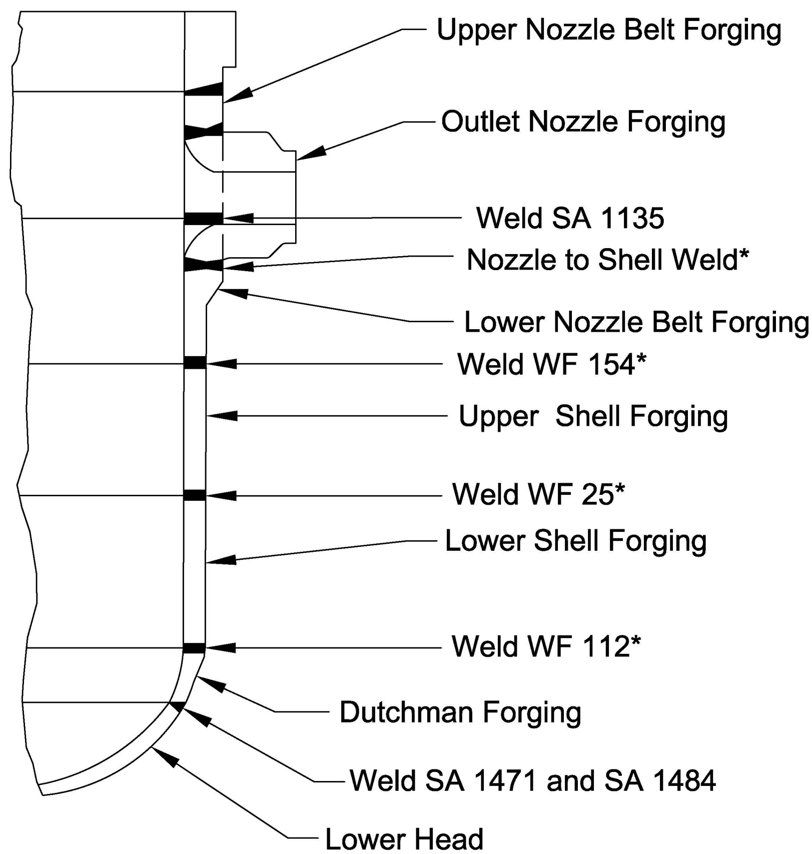
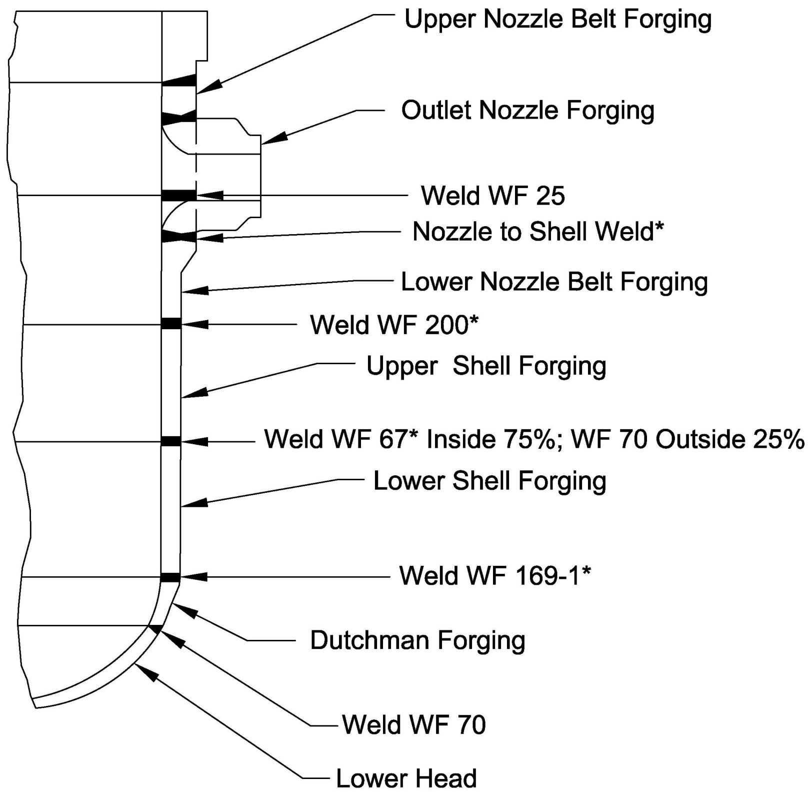


Figure 3—3 Reactor Vessel—Oconee Unit 2



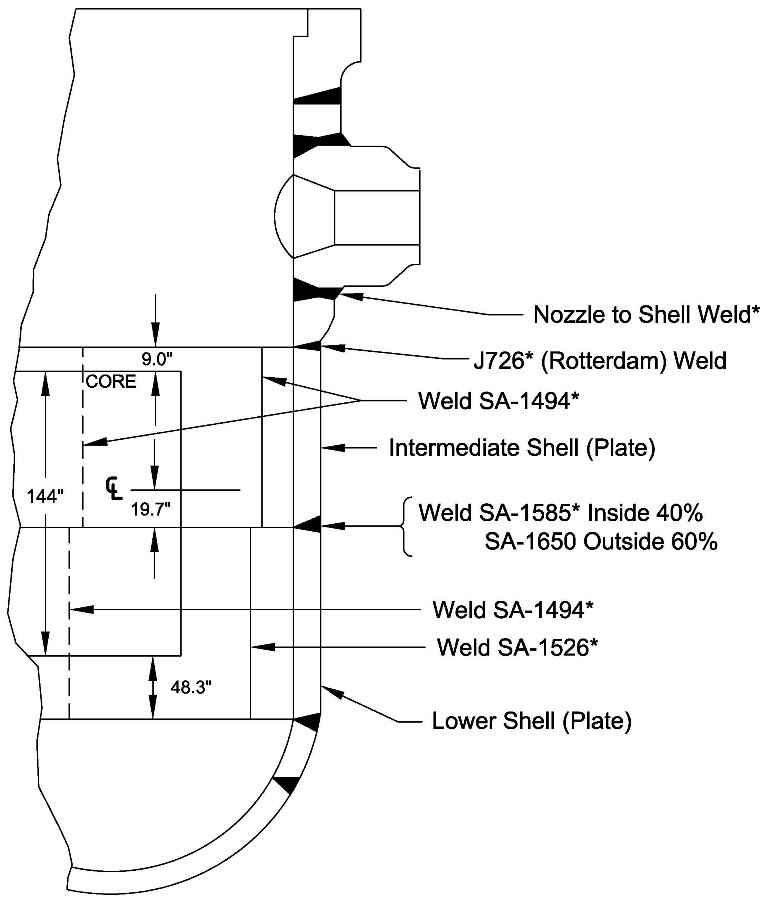
* Equivalent Margins Analysis
performed for these Linde 80 Welds.

Figure 3—4 Reactor Vessel—Oconee Unit 3



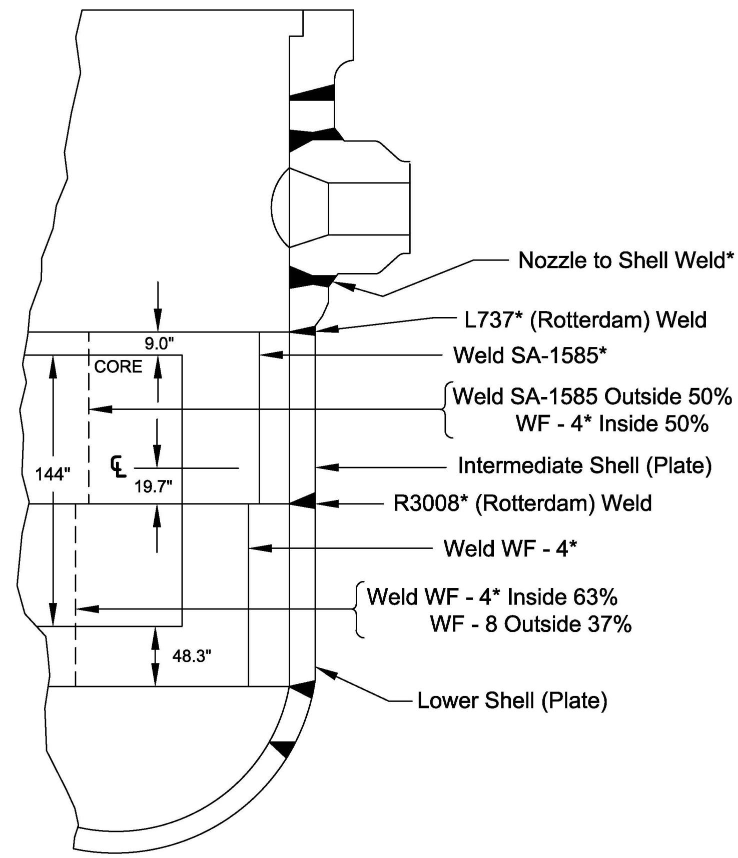
* Equivalent Margins Analysis
performed for these Linde 80 Welds.

Figure 3—5 Reactor Vessel—Surry Unit 1



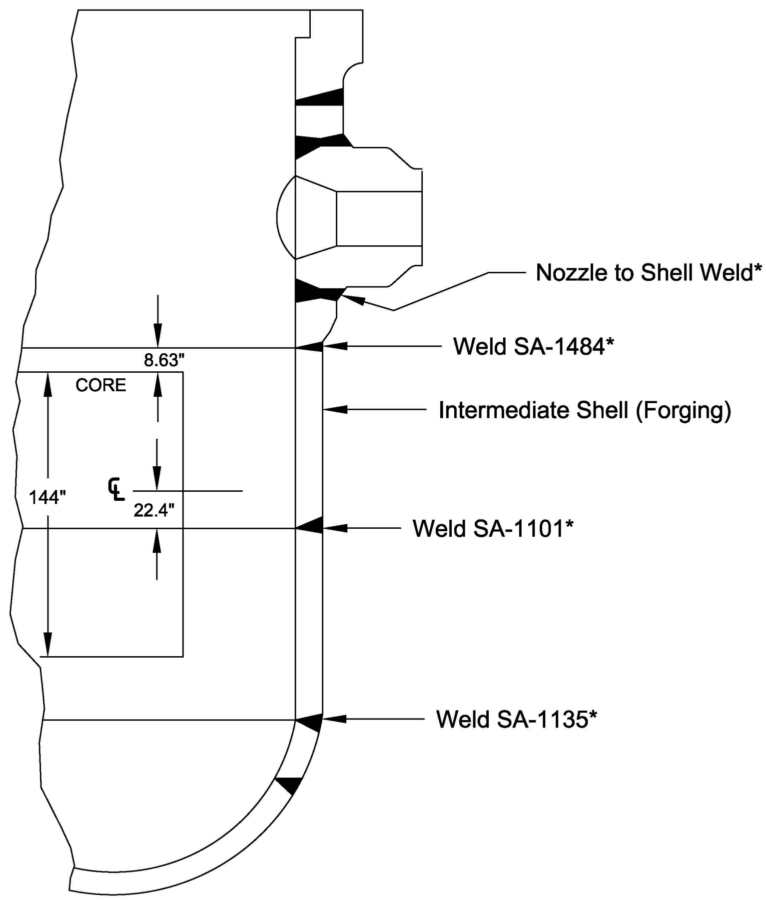
* Equivalent Margins Analysis performed
for these Linde 80 and Rotterdam Welds.

Figure 3—6 Reactor Vessel—Surry Unit 2



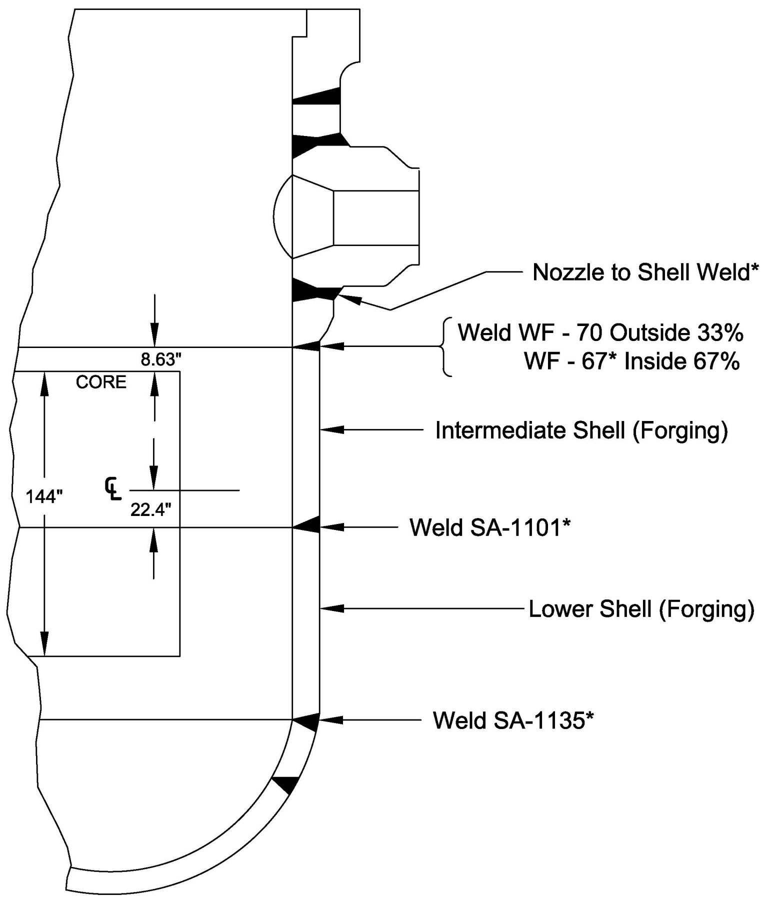
* Equivalent Margins Analysis performed
for these Linde 80 and Rotterdam Welds.

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 AND LEVELS C&D SERVICE LOADINGS

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 model (J-R model) data base. A detailed description of this model is provided in Appendix B of BAW-2192PA [2], Revision 0. 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 [6]. See Appendix A of BAW-2192P, Revision 0, Supplement 1, 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 reactor vessels, Surry reactor vessels, and Turkey Point reactor vessels. Consistent with BAW-2178PA, 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. When evaluating the vessel for Level D Service Loadings, the J-R curve shall be a best estimate 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 Level C Service Loadings (i.e., equation (1) multiplied by [ ]) while the unaltered Jd correlation would be used to evaluate Level D Service Loadings.

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 using correlations for effects of temperature, chemical composition, and fluence level. Crack extension was by ductile tearing with no cleavage. This complies with ASME 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 III) for the reactor base metal and cladding (the ASME Code does not provide separate mechanical properties for base and weld metal). 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. At the limiting time points in the Levels C & D analysis where cladding effects are included the temperature of the cladding is well below 600°F, and thus this change does not impact the EMA analyses summarized herein.

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 60-year license renewal low upper shelf toughness analysis submittals created for the plants (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 base metal and weld material and the E and α properties for the RV cladding material.

Table 4—2 Mechanical Properties of Oconee RV Materials

	RV Base Metal			Weld Metal	Cladding	
Temp. (°F)	E (ksi)	α (in/in/°F)	Sy (ksi)	Sy (ksi)	E (ksi)	α (in/in/°F)
70	27800	6.40E-06	50.0	[ ]	28300	8.50E-06
200	27100	6.70E-06	47.0	[ ]	27500	8.90E-06
300	26700	6.90E-06	45.5	[ ]	27000	9.20E-06
400	26200	7.10E-06	44.2	[ ]	26400	9.50E-06
500	25700	7.30E-06	43.2	[ ]	25900	9.70E-06
556	25364	7.36E-06	42.6	[ ]	25564	9.76E-06
600	25100	7.40E-06	42.1	[ ]	25300	9.80E-06

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 base metal and weld material and the E and α properties for the RV cladding material.

Table 4—3 Mechanical Properties of Surry RV Materials

	RV Base Metal			Weld Metal	Cladding	
Temp. (°F)	E (ksi)	α (in/in/°F)	Sy (ksi)	SA-1526 (ksi)	E (ksi)	α (in/in/°F)
70	29000	7.00E-06	50.0	[ ]	28300	8.50E-06
200	28500	7.30E-06	47.0	[ ]	27500	8.90E-06
300	28000	7.40E-06	45.5	[ ]	27000	9.20E-06
400	27600	7.60E-06	44.2	[ ]	26400	9.50E-06
500	27000	7.70E-06	43.2	[ ]	25900	9.70E-06
600	26300	7.80E-06	42.1	[ ]	25300	9.80E-06

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 base metal and weld material and the E and α properties for the RV cladding material.

Table 4—4 Mechanical Properties of Turkey Point RV Materials

	RV Base Metal			Weld Metals		Cladding	
Temp. (°F)	E (ksi)	α (in/in/°F)	Sy (ksi)	SA-1101 (ksi)	SA-1135 (ksi)	E (ksi)	α (in/in/°F)
70	27800	6.40E-06	50.0	[ ]	[ ]	28300	8.50E-06
200	27100	6.70E-06	47.0	[ ]	[ ]	27500	8.90E-06
300	26700	6.90E-06	45.5	[ ]	[ ]	27000	9.20E-06
400	26200	7.10E-06	44.2	[ ]	[ ]	26400	9.50E-06
500	25700	7.30E-06	43.2	[ ]	[ ]	25900	9.70E-06
600	25100	7.40E-06	42.1	[ ]	[ ]	25300	9.80E-06

4.3 *Levels C and D Service Loadings*

4.3.1 Oconee

Levels C and D Service Loadings were developed for the Oconee reactor vessels considering the transients listed below.

Level C: Stuck Open Turbine Bypass Valve (SOTBV)

Level D: Design Basis Steam Line Break (DB-SLB)

Steam Line Break (ALT-SLB)

Core Flood Line Break (CFLB)

Hot Leg Large Break Loss of Coolant Accident (HL-LOCA)

The pressure and temperature transients except for the CFLB transient are shown in Figure 4-1 and Figure 4-2, respectively. The CFLB is shown separately in Figure 4-3 and Figure 4-4. 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.

4.3.2 Surry

Levels C and D Service Loadings were developed for the Surry reactor vessels for the two steam line break transients identified below.

Level D: Steam Line Break (SM-0979)
Steam Line Break (SSDC 1.3 SLB)

The pressure and temperature steam line break transients for the Surry reactor vessels are illustrated in Figure 4-5.

4.3.3 Turkey Point

Levels C and D Service Loadings were developed for the Turkey Point reactor vessels for the two steam line break transients listed below. The Turkey Point Safety Injection (SI) pump shutoff head will limit RCS pressure increase to 1660 psi.

Level D: Steam Line Break (SLB Without Offsite Power)
Steam Line Break (SSDC 1.3 SLB)

The pressure and temperature steam line break transients for the Turkey Point reactor vessels are depicted in Figure 4-6.

Figure 4—1 Oconee Levels C and D Transients (except CFLB) – Reactor Coolant Pressure versus Time

Figure 4—2 Oconee Levels C and D Transients (except CFLB) – Reactor Coolant Temperature versus Time

[ ]

Figure 4—3 Oconee CFLB Transient – Reactor Coolant Pressure Versus Time

Figure 4—4 Oconee CFLB Transient – Reactor Coolant Temperature versus Time

Figure 4—5 Surry Steam Line Break Transients

Figure 4—6 Turkey Point Steam Line Break Transients

5.0 FRACTURE MECHANICS ANALYSIS

5.1 *Methodology*

In accordance with ASME Section XI, Appendix K [17], Subarticle K-1200, the following analytical procedure was used for Levels C & D Service Loadings:

a) 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 Subarticles K-2300 and K-2400.

b) Loading conditions at the locations of the postulated flaws were determined for Levels C and D Service Loadings.

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 by calculating the applied J-integral according to the procedure provided by Subarticle K-5210. The applied J-integral was then evaluated to satisfy the criteria for flaw extension in Subarticle K-5220 and flaw stability in Subarticle K-5300.

5.2 *Procedure for Evaluating Levels C and D Service Loadings*

The evaluation for the Levels C and D service loadings is performed as follows:

1 For each transient described in Section 4.3, calculate stress intensity factors for a semi-elliptical flaw of depth up to $^1/_{10}$ 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 for each of these transients is compared to the K_{Jc} upper-shelf toughness curve of the weld material. The transient for which the applied K_I most closely approaches the K_{Jc} curve is chosen as the limiting transient, and the critical time in the limiting transient selected for further evaluation occurs at the point where K_I most closely approaches the K_{Jc} 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. The pressure at instability, P_l, is given in K-5300, Appendix K of ASME Section XI for both axial and circumferential flaws.

5.2.1 Processing of Transient Time-History Data

5.2.1.1 Oconee

Transient input descriptions in the form of pressure and temperature time-histories were processed for analysis of the shell, transition, and nozzle regions of the Oconee reactor vessels using the one-dimensional finite element shell thermal model, closed-form expressions for through-wall thermal and pressure stresses, and closed-form stress intensity factor models embodied in the AREVA Inc. computer code PCRIT. Applied J-integrals under plane strain conditions were calculated from stress intensity factors adjusted for the effects of small scale yielding.

5.2.1.2 Surry

For the Surry reactor vessels, the applied J-integrals at the nozzle to shell welds and the upper transition weld were determined from three-dimensional finite element analysis. The through-thickness path line stresses were subsequently used to calculate stress intensity factors and the applied J-integrals were determined based on consideration of small scale yielding. For the controlling reactor vessel shell weld, stress intensity factors were calculated using the one-dimensional, finite element thermal and closed form stress models and linear elastic fracture mechanics methodology of the PCRIT computer code.

5.2.1.3 Turkey Point

For the Turkey Point reactor vessels, the applied J-integrals at the nozzle to shell welds and the upper transition weld were also determined from three-dimensional finite element analysis. The analysis approach was similar to that described above for the Surry reactor vessels.

5.2.2 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 4.1. 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 fracture 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.

5.2.3 Cladding Effects

[ ]

5.3 *Evaluation for Levels C and D Service Loadings*

The type of analysis models and computer code used to evaluate the RV shell welds, the RV transition welds and the RV nozzle welds for Levels C & D service loads are addressed in Sections 5.2.1.1 through 5.2.1.3 for the Oconee, Surry, and Turkey Point plants, respectively. Sections 4.3.1 through 4.3.3 address the specific types of transient events analyzed for each of these three groups of reactor vessels. The applied J-integral for the RV shell welds, the RV transition and nozzle welds, due to these Levels C & D transient events, is calculated and evaluated as discussed in Section 5.2.

The transition region toughness and upper shelf toughness are discussed in Section 5.2.2. Transition region toughness is obtained from the ASME Section XI equation for crack initiation,

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

using the applicable RTNDT value for a flaw depth of 1/10th the wall thickness, where:

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

Upper shelf toughness K_{Jc} is derived from the J-integral resistance model of Section 4.1 for a flaw depth of 1/10th the wall thickness, a crack extension of 0.10 inch, and the applicable fluence value at the crack tip:

$$K_{Jc} = \sqrt{\frac{J_{0.1}E}{1000(1-\nu^2)}}$$

where

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

Using the above equations, the transition and upper shelf toughness values as a function of temperature are determined for the controlling weld and Levels C and D service loading conditions.

5.3.1 Reactor Vessel Shell Welds

5.3.1.1 Oconee

The controlling weld for the Oconee reactor vessels was determined in the evaluation of the Levels A and B service loadings to be SA-1073. This limiting weld with a postulation of 1/10 of the base metal thickness is used to evaluate the Levels C and D service loadings. The stress intensity factor K_I calculated by the PCRIT code is the sum of thermal, residual, and pressure terms. PCRIT analysis runs are made for each of the identified Levels C and D transients from Section 4.3.1 (Figure 4—1 through Figure 4—4).

Figure 5-1 shows the variation of the applied stress intensity factor, K_I, for these transient cases as a function of the crack tip temperature. This figure also shows the transition region toughness K_{Ic} curve and the mean and lower bound upper-shelf toughness K_{Jc} curves with crack tip temperature. The K_{Ic} curve is determined using the Adjusted Reference Temperature (ART) value at 1/10th of the wall thickness for the limiting weld SA-1073, which at 80-years is [ ]. The symbols on the K_I curves for each of the transient cases indicate points in time at which PCRIT solutions are available. The Hot Leg Large Break LOCA is identified as the limiting transient since it most closely approaches the K_{Jc} limit of the weld. All subsequent analysis was therefore based on evaluation of this transient case. In the upper-shelf toughness range, the HL-LOCA K_I curve is closest to the lower bound K_{Jc} curve at 5.5 minutes into the transient. This time is selected as the critical time in the transient at which to perform the flaw evaluation for Levels C and D Service Loadings.

The additional stress intensity factor attributable to the cladding, K_{Iclad}, at 5.5 minutes (limiting time point) into the HL-LOCA transient is determined to be [ ] at a flaw depth corresponding to 1/10th of the wall thickness.

Applied J-Integrals are calculated for the controlling weld for various flaw depths in Table 5—1 using stress intensity factors from PCRIT for the Hot Leg Large Break LOCA (at 5.5 min.) and adding [ ] to account for cladding effects. Stress intensity factors are converted to *J*-integrals by the plane strain relationship,

$$J_{\text{applied}}(a) = 1000 \frac{K_{I\text{total}}^2(a)}{E}(1-\nu^2)$$

Since the RV shell weld for the Oconee reactor vessels are [ ] inches thick as given in Table 3—2, the initial flaw depth of 1/10 of the wall thickness is [ ] inches. The flaw extension is calculating by subtracting this depth from the built-in PCRIT flaw depths. The results along with the mean and lower bound J-R curves developed in Table 5-2 are plotted in Figure 5—2. An evaluation line is used at a flaw extension of 0.10 in. to show that the applied *J*-integral is less than the lower bound *J*-integral of the material, as required by ASME Section XI Appendix K.

The applied J-integral at a flaw extension of 0.1 inch is determined to be [ ] as reported below Table 5—1. The associated material J-resistance ($J_{0.1}$) to the applied J-integral (J_1) ratio (also referred to as "margin") using the lower bound and mean J-R curve values (from Table 5—2) for Levels C and D conditions, respectively, are also shown below Table 5—1. The margin for Level C Service Loading is [ ] and the margin for Level D Service Loading is [ ].

The requirements for ductile and stable crack growth are demonstrated by Figure 5—2 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.

Referring to Figure 5—2, 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 curve intersects the mean *J-R* curve at a flaw extension that is only a small fraction of the wall thickness (less than 1%).

The last requirement for Level D Conditions is that the internal pressure *p* shall be less than P_l, the internal pressure at tensile instability of the remaining ligament. The calculations for P_l were determined for an axial flaw for various flaw depths up to 1.01 inches. The lowest P_l value is in excess of 6000 psi, which is far greater than any operating or accident condition pressure. The remaining ligament is therefore not subject to tensile instability.

5.3.1.2 Surry

For the Surry reactor vessels, the controlling weld was identified to be the SA-1526 RV shell weld based on the results of Levels A and B Service Loadings. This controlling weld is therefore evaluated at a flaw depth of 1/10 the base metal thickness for Levels C and D Service Loadings. The two main steam line break (SLB) transients (SSDC 1.3 SLB & SM-0979) identified in Section 4.3.2 and illustrated in Figure 4—5 are evaluated. The analysis is performed using the PCRIT Code and performed similarly to that described above for the Oconee reactor vessels.

Figure 5-3 shows the variation of the applied stress intensity factor, K_I, for these two transient cases as a function of the crack tip temperature. This figure also shows the transition region toughness K_{Ic} curve and the mean and lower bound upper-shelf toughness K_{Jc} curves with crack tip temperature. The K_{Ic} curve is determined using the Adjusted Reference Temperature (ART) value at 1/10th of the wall thickness for the limiting weld SA-1526, which at 80-years is [ ]. The symbols on the K_I curves for each of the two transient cases indicate points in time at which PCRIT solutions are available. The SSDC SLB is identified as the limiting transient since it most closely approaches the K_{Jc} limit of the weld. All subsequent analysis was therefore based on evaluation of this transient case. In the upper-shelf toughness range, the SSDC SLB K_I curve is closest to the lower bound K_{Jc} curve at 10.0 minutes into the transient. This time is selected as the critical time in the transient at which to perform the flaw evaluation for Levels C and D Service Loadings.

The additional stress intensity factor attributable to the cladding, K_{Iclad}, at 10.0 minutes (limiting time point) into the SSDC SLB transient is determined to be [ ] at a flaw depth corresponding to 1/10th of the wall thickness.

Applied J-Integrals are calculated for the controlling weld for various flaw depths in Table 5—3using stress intensity factors from PCRIT for the SDC SLB (at 10.0 min.) and adding [ ] to account for cladding effects. Stress intensity factors are converted to *J*-integrals by the previously reported plane strain relationship.

Since the RV shell weld for the Surry reactor vessels is [ ] inch thick as given in Table 3-2, the initial flaw depth of 1/10 of the wall thickness is [ ] inches. The flaw extension is calculating by subtracting this depth from the built-in PCRIT flaw depths. The results along with the mean and lower bound J-R curves developed in Table 5—4, are plotted in Figure 5—4. An evaluation line is used at a flaw extension of 0.10 in. to show that the applied *J*-integral is less than the lower bound *J*-integral of the material, as required by ASME Section XI, Appendix K.

The applied J-integral at a flaw extension of 0.1 inch is determined to be [ ] as reported at the base of Table 5-3. The associated margin using the lower bound and mean J-R curve values (from Table 5-4) for Levels C and D conditions, respectively are also shown at the base of Table 5-3. The margin for Level C Service Loading is [ ] and the margin for Level D Service Loading is [ ]. In accordance with BAW-2192, Supplement 1, Appendix A [3], the margin for Level C loading using J-R Model 6B is [ ].

The requirements for ductile and stable crack growth are demonstrated by Figure 5—4 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.

Referring to Figure 5—4, 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 curve intersects the mean *J-R* curve at a flaw extension that is only a small fraction of the wall thickness.

The last requirement for Level D Conditions is that the internal pressure *p* shall be less than P_I, the internal pressure at tensile instability of the remaining ligament. The calculations for P_I were determined for an axial flaw with various flaw depths up to 1.26 inches. The P_I values calculated are in excess of 9000 psi, which is far greater than any expected pressure. The remaining ligament is therefore not subject to tensile instability.

5.3.1.3 Turkey Point

For the Turkey Point reactor vessels, the controlling weld was identified to be the SA-1101 RV shell weld based on the results of Levels A and B Service Loadings. This controlling weld is therefore evaluated at a flaw depth of 1/10 the base metal thickness for Levels C and D Service Loadings. The two main steam line break transients (SSDC 1.3 SLB & SLB without Offsite Power) identified in Section 4.3.3 and illustrated in Figure 4-6 are evaluated. The analysis is performed using the PCRIT Code and performed in a similar manner to that described above for the Oconee and Surry reactor vessels.

Figure 5—5 shows the variation of the applied stress intensity factor, K_I, for these two transient cases as a function of the crack tip temperature. This figure also shows the transition region toughness K_{Ic} curve and the mean and lower bound upper-shelf toughness K_{Jc} curves with crack tip temperature. The K_{Ic} curve is determined using the Adjusted Reference Temperature (ART) value at 1/10th of the wall thickness for the limiting weld SA-1101, which at 80-years is [ ] . The symbols on the K_I curves for each of the two transient cases indicate points in time at which PCRIT solutions are available. The SSDC SLB is identified as the limiting transient since it most closely approaches the K_{Jc} limit of the weld. All subsequent analysis was therefore based on evaluation of this transient case. In the upper-shelf toughness range, the SSDC SLB K_I curve is closest to the lower bound K_{Jc} curve at 6.5 minutes into the transient. This time is selected as the critical time in the transient at which to perform the flaw evaluation for Levels C and D Service Loadings.

The additional stress intensity factor attributable to the cladding, K_{Iclad}, at 6.5 minutes (limiting time point) into the SSDC SLB transient is determined to be **[ ]** at a flaw depth corresponding to 1/10th of the wall thickness. Applied J-Integrals are calculated for the controlling weld for various flaw depths in Table 5—5 using stress intensity factors from PCRIT for the SSDC SLB (at 6.5 min.) and adding **[ ]** to account for cladding effects. Stress intensity factors are converted to *J*-integrals by the previously reported plane strain relationship.

Since the RV shell weld for the Turkey Point reactor vessels is **[ ]** inch thick as given in Table 3—2, the initial flaw depth of 1/10 of the wall thickness is **[ ]** inches. The flaw extension is calculating by subtracting this depth from the built-in PCRIT flaw depths. The results along with the mean and lower bound J-R curves developed in Table 5-6 are plotted in Figure 5—6. An evaluation line is used at a flaw extension of 0.10 in. to show that the applied *J*-integral is less than the lower bound *J*-integral of the material, as required by Appendix K.

The applied J-integral at a flaw extension of 0.1 inch is determined to be **[ ]** as reported at the base of Table 5—5. The associated material J-resistance ($J_{0.1}$) to the applied J-integral (J_1) ratio can be determined using the lower bound and mean J-R curve values, from Table 5-6, for Levels C and D conditions, respectively. The margin for Level C Service Loading is **[ ]** and the margin for Level D Service Loading is **[ ]**

The requirements for ductile and stable crack growth are demonstrated by Figure 5-6 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.

Referring to Figure 5—6, 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 curve intersects the mean *J-R* curve at a flaw extension that is only a small fraction of the wall thickness (less than 1%).

The last requirement for Level D Conditions is that the internal pressure p shall be less than P_l, the internal pressure at tensile instability of the remaining ligament. The calculations for P_l were determined for a circumferential flaw. An additional check is performed for circumferential flaws to ensure that internal pressure does not exceed 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 is satisfied by

$$P_{\text{instability}} \leq 1.07\ \sigma_o \frac{t}{R_i}.$$

To demonstrate that the remaining ligament does not exceed the pressure at instability a conservative flaw depth equal to 1/10th of the wall thickness plus 0.1 inch is used. Although the internal pressure at tensile instability is calculated to be 12, 530 psi, the validity check on hoop stress requires that the internal pressure not exceed 6240 psi, which is still much greater than any anticipated accident condition pressure. Therefore, the remaining ligament is not subject to tensile instability.

5.3.2 Reactor Vessel Transition Welds and RV Nozzle Welds

The reactor vessel upper and lower transition welds are located above and below the reactor core, respectively (see Figures 3-1 through 3-8). The RV nozzle welds are located above the upper transition weld in the substantially thicker cylindrical section (reinforced to account for the inlet/outlet RV nozzle openings). The reactor vessel nozzle belt dimensions are reported in Table 3-3.

5.3.2.1 Oconee

For the Oconee reactor vessels there is only one upper transition weld: SA-1135 for ONS-1 (see Figures 3-1 through 3-4). This circumferential weld along with the lower transition welds in each of the Oconee Units reactor vessels are bounded by the limiting RV shell weld SA-1073 that is addressed in Section 5.3.1. As such only the RV nozzles to shell welds are addressed herein.

A flaw depth of 1/10 of the base metal wall thickness, plus the cladding thickness with total depth not to exceed 1 inch is used to evaluate the Levels C and D Service Loadings.

The flaw evaluation at the RV nozzle-to-shell intersection are provided in Table 5-7. The applied stress intensity factor is conservatively calculated using the flat plate solution by Newman and Raju. The calculated ratios of $J_{0.1}/J_1$ or margins are [ ] and [ ] for flaws oriented in the axial and circumferential directions with respect to the RV, which are higher than the minimum acceptable value of 1.0. The limiting Hot Leg LOCA transient was evaluated using the PCRIT Code as used for the evaluation of the RV shell welds in Section 5.3.1. The maximum 1/10 wall thickness ART value for the RV nozzles is [ ]. Figure 5-7 shows the applied stress intensity factor, K_{Iappl} the transition toughness, K_{Ic}, and the upper-shelf toughness, K_{Jc} as a function of temperature. The symbols on the K_I curves indicate points in time at which PCRIT solutions are available. In the upper-shelf toughness range, the HL-LOCA K_I curve is closest to the lower bound K_{Jc} curve at 6.8 minutes into the transient. This time is selected as the critical time in the transient at which to perform the flaw evaluation for Levels C and D Service Loadings.

Applied *J*-integrals are calculated for the controlling weld for various flaw depths using stress intensity factors from PCRIT for the Hot Leg Large Break LOCA (at 6.8 min.) and adding [ ] to account for cladding effects. Stress intensity factors are converted to *J*-integrals by the plane strain relationship discussed previously. An initial flaw depth of 1.0 inch ([ ]) is used per the ASME Section XI, Appendix K guideline. Flaw extension from this depth is calculated by subtracting [ ] inch from the built-in PCRIT flaw depths. The results along with the mean and lower bound *J-R* curve are plotted in Figure 5-8. An evaluation line is used at a flaw extension of 0.10 in. to show that the applied *J*-integral is less than the lower bound *J*-integral of the material, as required by Appendix K. The requirements for ductile and stable crack growth are demonstrated in Figure 5-8 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.

Referring to Figure 5-8, 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 curve intersects the mean *J-R* curve at a flaw extension that is only a small fraction of the wall thickness (less than 1%).

The last requirement for Level D Conditions is that the internal pressure p shall be less than P_I, the internal pressure at tensile instability of the remaining ligament. The calculations for P_I were determined for an axial flaw for flaw depths up to 1.2 inches. The lowest value determined was in excess of 9000 psi, which is far greater than any accident condition pressure; therefore the remaining ligament is not subject to tensile instability.

5.3.2.2 Surry

For the Surry reactor vessels the applied J-integrals for the nozzle to shell and upper transition welds were evaluated for Levels C and D Service Loadings. Both transients shown in Figure 4-5 are evaluated. The bounding results with safety factor of 1.0 on the applied pressure are compared with the lower bound J-integral resistance at a ductile flaw extension of 0.1 inches in Table 5-8. The outlet nozzle is seen to be limiting and has a margin of [ ]. The applied J-integral vs crack tip temperature for each transient (SSDC SLB and SM-0979 SLB) is plotted in Figure 5-9 for the outlet nozzle, along with the temperature dependent mean and lower bound $J_{0.1}$ curves. As can be seen all points of the transient remain below the lower bound $J_{0.1}$. Additionally, Figure 5-9 shows the K_{Ic} fracture toughness using an RT_{NDT} of [ ] (lowest of nozzle to shell welds maximizes the upper shelf range), converted to an equivalent J using $K_{Ic}^2/(E/(1-v^2))$; the intersection of this curve with the $J_{0.1}$ curves establishes the upper shelf temperature range. The applied J-integral at the limiting time point at various flaw extensions is plotted with the lower bound J-resistance curve in Figure 5-10; 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.

5.3.2.3 Turkey Point

For the Turkey Point reactor vessels the applied J-integrals for the nozzle to shell and upper transition welds were evaluated for Levels C and D Service Loadings. Both transients shown in Figure 4-6 are evaluated. For the SSDC transient the maximum pressure was capped at 1660 psi due to the shutoff head of the safety injection pumps. The bounding results with safety factor of 1.0 on the applied pressure are compared with the lower bound J-integral resistance at a ductile flaw extension of 0.1 inches in Table 5-9. The outlet nozzle is seen to be limiting and has a margin of [ ]. The applied J-integral vs crack tip temperature for each transient, the SSDC SLB transient and SLB without offsite power transient (also referred to as AIS transient), is plotted in Figure 5-11 for the outlet nozzle, along with the temperature dependent mean and lower bound $J_{0.1}$ curves. As can be seen all points of the transient remain below the lower bound $J_{0.1}$. Additionally, Figure 5-11 shows the K_{Ic} fracture toughness using an RT_{NDT} of [ ], converted to an equivalent J using $K_{Ic}^2/(E/(1-v^2))$; the intersection of this curve with the $J_{0.1}$ curves establishes the upper shelf temperature range. The applied J-integral at the limiting time point at various flaw extensions is plotted with the lower bound J-resistance curve in Figure 5-12; 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 Oconee-J-Integral vs. Flaw Extension for Levels C and D Service Loadings

Table 5—2 ONS-J-R Curves for Evaluation of Levels C and D Service Loadings

Table 5—3 Surry J-Integral versus Flaw Extension for Levels C & D Service Loadings

Table 5—4 Surry J-R Curves for Evaluation of Levels C & D Service Loadings

Table 5—5 Turkey Point J-Integral versus Flaw Extension for Levels C & D Service Loadings

Table 5—6 Turkey Point-J-R Curves for Evaluation of Levels C & D Service Loadings

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Table 5—7 Flaw Evaluation Summary of Oconee RV Nozzle-to-Shell Weld

Table 5—8 Surry-Levels C & D Results for Nozzle to Shell and Upper Transition Welds

Table 5—9 Turkey Point-Levels C & D Results for Nozzle to Shell and Upper Transition Welds

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Figure 5—1 Oconee--KIc, KJc (mean and lower bound), and Applied KI for all Levels C and D transients

Figure 5—2 Oconee-J-Integral vs. Flaw Extension for Levels C & D Service Loadings

Figure 5—3 Surry-KI versus Crack Tip Temperature for Levels C & D Service Loadings

Figure 5—4 Surry-J-Integral versus Flaw Extension for Levels C & D Service Loadings

Figure 5—5 Turkey Point-KI versus Crack Tip Temperature for Levels C & D Service Loadings

Figure 5—6 Turkey Point-J-Integral versus Flaw Extension for Levels C & D Service Loadings

Figure 5—7 Oconee-KIc, KJc (mean and lower bound), and Applied KI for all Level C and D transients

Figure 5—8 Oconee J-Integral vs. Flaw Extension for Levels C & D Service Loadings

Figure 5—9 Surry- Level C & D Applied J Integral vs Crack Tip Temperature for the Outlet Nozzle to Shell Weld

Figure 5—10 Surry-Levels C & D Applied J Integral vs Crack Extension for the Outlet Nozzle to Shell Weld

Figure 5—11 Turkey Point- Levels C & D Applied J Integral vs Crack Tip Temperature for the Outlet Nozzle to Shell Weld

[ ]

Figure 5—12 Turkey Point- Levels C & D Applied J Integral vs Crack Extension for the Outlet Nozzle to Shell Weld

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6.0 SUMMARY AND CONCLUSIONS

6.1 *Reactor Vessel Shell Welds*

The ASME Section XI, acceptance criteria for Levels C & D Service Loads for all reactor vessel shell welds are satisfied. ASME Section XI, Appendix K, Level C acceptance criteria (Subarticle K-2300), relative to the ratio of applied J-integral to J-integral of the material and use of the lower bound J-integral resistance curve, were conservatively imposed on the Level D transients evaluated in Section 5.0, although ASME Section XI, Subarticle K-2400(b) permits use of a best estimate J-integral resistance curve for Level D Service Loadings. 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.

6.1.1 Oconee Units 1, 2, and 3

The limiting weld among the Oconee Reactor Vessel shell welds is ONS-1 longitudinal weld SA-1073. The limiting transient for Level C & D service loads is the HL-LOCA.

- With a factor of safety of 1.0 on loading, the applied J-integral (J_1) for the limiting reactor vessel shell weld (ONS-1, SA-1073) is less than the lower bound J-integral of the material at a ductile flaw extension of 0.10 inch ($J_{0.1}$) with a ratio $J_{0.1}/J_1$ of **[ ]**, which is greater than the required value of 1.0. Using the mean J-R curve permitted by Subarticle K-2400 for this Service Level D transient, the ratio $J_{0.1}/J_1$ is **[ ]**.

- With a factor of safety of 1.0 on loading, flaw extensions are ductile and stable for the limiting reactor vessel shell weld (SA-1073) 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.

- For weld SA-1073 it was demonstrated that flaw growth is 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.

6.1.2 Surry Units 1 and 2

The limiting weld among the Surry reactor vessel shell welds is Surry Unit 1 longitudinal weld SA-1526. The limiting transient for Level C & D service Loads is the SSDC 1.3 steam line break.

- With a factor of safety of 1.0 on loading, the applied J-integral (J_1) for the limiting reactor vessel shell weld (Surry Unit 1, SA-1526) is less than the lower bound J-integral of the material at a ductile flaw extension of 0.10 inch ($J_{0.1}$) with a ratio $J_{0.1}/J_1$ of [ ], which is greater than the required value of 1.0. Using the mean J-R curve permitted by Subarticle K-2400 for this Service Level D transient, the ratio $J_{0.1}/J_1$ is [ ]. In accordance with BAW-2192, Supplement 1, Appendix A [3], the margin for Level C loading using J-R Model 6B is [ ].
- With a factor of safety of 1.0 on loading, flaw extensions are ductile and stable for the limiting reactor vessel shell weld (SA-1526) 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.
- For weld SA-1526 it was demonstrated that flaw growth is 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.

6.1.3 Turkey Point Units 3 and 4

The limiting weld among the Turkey Point reactor vessel shell welds is Turkey Point Units 3 and 4 circumferential weld SA-1101. The limiting transient for Level C & D service loads is the SSDC 1.3 steam line break.

- With a factor of safety of 1.0 on loading, the applied J-integral (J_1) for the limiting reactor vessel shell weld (SA-1101) is less than the lower bound J-integral of the material at a ductile flaw extension of 0.10 inch ($J_{0.1}$) with a ratio $J_{0.1}/J_1$ = **[ ]** = **[ ]**, which is greater than the required value of 1.0. Using the mean J-R curve permitted by Subarticle K-2400 for this Service Level D transient, the ratio $J_{0.1}/J_1$ is **[ ]**.
- With a factor of safety of 1.0 on loading, flaw extensions are ductile and stable for the limiting reactor vessel shell weld (SA-1101) 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.
- For weld SA-1101 it was demonstrated that flaw growth is 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.

6.2 *Reactor Vessel Transition Welds and RV Nozzle Welds*

The ASME Section XI, acceptance criteria for Levels C & D Service Loads for all reactor vessel transition welds and reactor vessel nozzle welds are satisfied. ASME Section XI, Appendix K, Level C acceptance criteria (Subarticle K-2300), relative to the ratio of applied J-integral to J-integral of the material and use of the lower bound J-integral resistance curve, were conservatively imposed on the Level D transients evaluated in Section 5.0, although ASME Section XI, Subarticle K-2400(b) permits use of a best estimate J-integral resistance curve for Level D Service Loadings. 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.

6.2.1 Oconee Units 1, 2, and 3

The Oconee transitions welds are as follows: upper weld (SA-1135) for ONS-1 only, and lower weld connecting lower end of the lower shell to the Dutchman Forging (WF-9 for ONS-1, WF-112 for ONS-2, and WF-169-1 for ONS-3). The transition welds, when considering the effects of structural discontinuities and Levels A and B service loadings, have a significantly higher margin ($J_{0.1}/J_{I}$) when compared to the limiting reactor vessel shell weld SA-1073. Therefore, the transition welds were not evaluated for Levels C and D Service Loadings. The RV inlet and outlet nozzle-to-shell welds were evaluated considering the bounding outlet nozzles and applicable Levels C and D Service Loadings. The limiting transient for Level C & D service Loads is the HL-LOCA.

- With a factor of safety of 1.0 on loading, the applied J-integral (J_1) for the bounding RV nozzle-to-shell weld is less than the lower bound J-integral of the material at a ductile flaw extension of 0.10 inch ($J_{0.1}$) with a ratio $J_{0.1}/J_1$ of [ ], which is greater than the required value of 1.0.
- With a factor of safety of 1.0 on loading, flaw extensions are ductile and stable for the bounding RV nozzle-to-shell weld.
- For the bounding RV nozzle-to-shell weld it was demonstrated that flaw growth is. 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.

6.2.2 Surry Units 1 and 2

The upper transition weld and RV inlet and outlet nozzle-to-shell welds were evaluated for Levels C and D Service Loadings. The limiting transient for Level C & D service loads is the SSDC 1.3 steam line break.

- With a factor of safety of 1.0 on loading, the applied J-integral (J_1) for the RV nozzle-to-shell welds and upper transition weld are less than the lower bound J-integral of the material at a ductile flaw extension of 0.10 inch ($J_{0.1}$) with the following ratios for $J_{0.1}/J_1$: [ ] for the RV outlet nozzle-to-shell weld, [ ] for the RV inlet nozzle-to-shell weld, and [ ] for the upper transition weld (i.e. axial flaw in longitudinal weld SA-1585 at the intersection with circumferential weld L737).
- With a factor of safety of 1.0 on loading, flaw extensions are ductile and stable for the limiting RV outlet nozzle-to-shell weld (i.e., limiting location considering RV nozzle-to-shell welds and upper transition weld).
- For the RV outlet nozzle-to-shell weld it was demonstrated that flaw growth is stable at much less than 75% of the vessel wall thickness. Tensile instability was not explicitly calculated but because this section of the reactor vessel is thicker compared to the RV shell welds, it is considered to be bounded by the RV shell location.

6.2.3 Turkey Point Units 3 and 4

The upper transition weld and RV inlet and outlet nozzle-to-shell welds were evaluated for Levels C and D Service Loadings. The limiting transient for Level C & D service loads is the SSDC 1.3 steam line break.

- With a factor of safety of 1.0 on loading, the applied J-integral (J_1) for the RV nozzle-to-shell welds and upper transition weld are less than the lower bound J-integral of the material at a ductile flaw extension of 0.10 inch ($J_{0.1}$) with the following ratios for $J_{0.1}/J_1$: [ ] for the RV outlet nozzle-to-shell weld, [ ] for the RV inlet nozzle-to-shell weld, and [ ] for the upper transition weld.
- With a factor of safety of 1.0 on loading, flaw extensions are ductile and stable for the limiting RV outlet nozzle-to-shell weld (i.e., limiting location considering RV nozzle-to-shell welds and upper transition weld).

- For the RV outlet nozzle-to-shell weld it was demonstrated that flaw growth is stable at much less than 75% of the vessel wall thickness. Tensile instability was not explicitly calculated but because this section of the reactor vessel is thicker compared to the RV shell welds, it is considered to be bounded by the RV shell location.

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8.0 CERTIFICATION

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

Ashok Nana/Mark Rinckel
Component Analysis & 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 Oconee Units 1, 2 and 3, Surry Units 1 and 2, and Turkey Point Units 3 and 4.

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Verification of independent review.

David Cofflin
Component Analysis & Fracture
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This report is approved for release.

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NSSS Project Management