DUKE ENERGY®

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

Subject: Brunswick Steam Electric Plant, Unit Nos. 1 and 2
Renewed Facility Operating License Nos. DPR-71 and DPR-62
Docket Nos. 50-325 and 50-324
Pressure and Temperature Limits Report for Unit Nos.1 and 2

Ladies and Gentlemen:

Enclosed are copies of the Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR) for Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2. Duke Energy Progress, LLC (Duke Energy), is providing the enclosed PTLR in accordance with Brunswick Unit 1 and Unit 2 Technical Specifications 5.6.8.c.

This document and the enclosed PTLR do not contain any regulatory commitments.

Please contact me at (910) 832-2568, if there are any questions regarding this submittal.

Sincerely,

Stephen B. Yodersmith
Manager – Nuclear Support Services (Acting)
Brunswick Steam Electric Plant

SBY/mkb

Enclosure:
Brunswick Steam Electric Plant, Unit Nos. 1 and 2
Pressure and Temperature Limits Report

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Brunswick Steam Electric Plant, Unit Nos. 1 and 2
Renewed Facility Operating License Nos. DPR-71 and DPR-62
Docket Nos. 50-325 and 50-324

Brunswick Steam Electric Plant, Unit Nos. 1 and 2
Pressure and Temperature Limits Report

DUKE ENERGY

Facility Code :	BNP
Applicable Facilities :	
Document Number :	0B11-0062
Document Revision Number :	004
Document EC Number :	
Change Reason :	
Document Title :	PRESSURE AND TEMPERATURE LIMITS REPORT FOR 54 EFFECTIVE FULL-POWER YEARS

Bliss, Victor M	Preparer	7/15/2019
Karriker, Scott H	Approver	7/15/2019

Notes :

Calculation Cover Sheet

Pressure and Temperature Limits Report for 54 Effective Full-Power Years

Calculation Number: 0B11-0062 Rev # 004

System: 1005 DSD List: ☐ Yes ☒ No

[BNP, HNP, RNP] Sub-Type: RVI Microfiche Attachment List: ☐ Yes ☒ No

Quality Level Priority E: ☐ Yes ☒ No

☐ All

☒ BNP Unit 0 ☐ CNS Unit ☐ HNP Unit

☐ MNS Unit ☐ ONS Unit ☐ RNP Unit

☐ WLS Unit ☐ LNP Unit ☐ HAR Unit

☐ General Office ☐ Keowee Hydro Station

Originated By	Design Verification Review By	Approved By
Signature	Signature	Signature
	Verification Method 1 ☐ 2 ☐ 3 ☐ Other ☐	
Printed Name Victor Bliss	Printed Name	Printed Name Scott Karriker
Date See digital signature	Date	Date See digital signature

☐ **YES** ☒ **NO** Check Box for Multiple Originators or Design Verifiers (see next page)

For Vendor Calculations:

Vendor: Structural Integrity Associates, Inc. Vendor Document #: 1700147.401.RC ISSUED 10/28/2018

Owners Review By: Victor Bliss Date: See digital signature

Approval By: Scott Karriker Date: See digital signature

List of Affected Pages

Calculation Number:	**0B11-0062**
Revision Number:	**004**

Body of Calculation (including appendices)				**Supporting Documents**				
Rev. #	*Pages Revised*	*Pages Deleted*	*Pages Added*	*Rev. #*	*Type*	*Pages Revised*	*Pages Deleted*	*Pages Added*
000			53					
001	3 (pages 2, 5 & 54)							
002	12 (pages 1, 4, 13, 15, 18, 19, 21, 22, 31, 34, 40, 43)							
003	7 (pages 19 of 47 thru 25 of 47 of Attach. 1)							
004	2 (pages 6 and 8 of Attachment 1)							

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ATTACHED SUPPORTING DOCUMENTS

Revision Summary

Revision	Summary
000	New issue of vendor calculation defining Pressure and Temperature Limits Report (PTLR) for 54 Effective Full-Power Years (EFPY) using the new format for calculations per AD-EG-ALL-1117. Include data and conclusions from 2017 vendor provided calculations as referenced in supporting calculation 0B11-0027 Rev 001. All technical content provided by vendor, Structural Integrity Associates, Inc. This calculation submits the pressure temperature curves for use during the periods of extended operation of Brunswick Nuclear Plant Units 1 and 2 in accordance with the 10 CFR 50.90 License Amendment Process at least one year prior to the expiration of the 32 EFPY P-T Limit Curves that are currently approved in the Technical Specifications.
001	Date Correction: Reference [13] which is discussed in Appendix A of Attachment 1 on page 54 of 0B11-0062 (page 47 of 47 in Attachment 1) has an incorrectly copied date of July 23, 2004. The reference [13] as noted on page 21 of 0B11-0062 (page 14 of 47 in Attachment 1) has the correct date of January 14, 2004. The approving letter from the NRC is dated January 14, 2004. Therefore, the date on page 54 will be changed to January 14, 2004. The Revision Summary and the List of Affected pages are also revised to record the change. Added the vendor calculation number to the cover page.
002	Curve B and Curve C have been updated due to a fracture mechanics analysis for the feedwater nozzle that incorporates revised plant-specific heat transfer and fluid temperature inputs to the finite element thermal analysis, NEDC030633R1(Unit 2) and NEDC030634R1 (Unit 1) which revised SIA 1700147.301R1 and 1700147.302R1.
003	Editorial correction on graph titles. Starting on Figure 4, its graph title was left on the previous page during pagination, and the error followed through on the remaining graphs. The title alignment was corrected.
004	Editorial revision Attachment 1. This revision copies a reference comment currently in section 5 (Discussion) regarding Brunswick's operating guideline limits, which were previous noted in Brunswick's Technical Specifications, and adding that reference comment to section 4 (Operating Limits). The addition is in section 4, page 6 of 47 Attachment 1, noted by a revision line. In addition, the copied comment from section 5, page 8 of 47 Attachment 1, has been modified to note its historical reference (i.e. no longer included in the Technical Specifications). The reference comment is "Normal Operational conditions, the RCS heatup and cooldown rates are **≤ 100˚F/hour** in any 1 hour period."

DOCUMENT INDEXING TABLE

The purpose of this table is to list cross-references

Document Type (e.g., CALC, DWG, Procedure, Tag, Software)	**Document Number** (e.g., Calculation Number, Equipment Tag Number, Procedure Number, Software Name and Version)	**Function** IN for Design Inputs; OUT for Affected Documents	**Relationship to this Calculation** (e.g., Design Input, Assumption Basis, Reference, Document affected by results)
CALC	0B11-0012, Rev. 1	Input	Provides fluence information used in the development of the P-T curves.
CALC	0B21-1029, Rev. 0	Input	Provides instrument uncertainty values used in the development of the P-T curves.
CALC	0B11-0005, Rev. 1	Input	Methodology documented in 0B11-0005, Rev. 1 was used to develop the P-T curves in this calculation package.
CALC	0B11-0027R001	Input	Revision 001 includes additional calculations which developed the final PTLR document 0B11-0062 R000.

1.0 Purpose

The purpose of this calculation package is to document the formulation of Reactor Pressure Vessel (RPV) pressure-temperature curves for the licensing renewal period, i.e. up to 54 effective full power years (EFPY). Curves have been developed for core critical, core not critical, and pressure test conditions. The curves were developed using the same methodology that was used to develop the pressure-temperature curves for extended power uprate (EPU).

This revision submits the pressure temperature curves for use during the periods of extended operation of Brunswick Nuclear Plant Units 1 and 2 in accordance with the 10 CFR 50.90 License Amendment Process at least one year prior to the expiration of the 32 EFPY P-T Limit Curves that are currently approved in the Technical Specifications.

2.0 References

1. Code of Federal Regulations 10 CFR 50, Appendix G.
2. BNP Calculation 0B11-0005, Rev. 1, "Development of RPV Pressure-Temperature Curves For BNP Units 1 & 2 For Up To 32 EFPY of Plant Operation."
3. BNP Calculation 0B21-1029, Rev. 0, "Instrument Uncertainty for RCS Pressure/Temperature Limits Curve."
4. BNP Calculation 0B11-0012, Rev. 1, "Neutron Exposure Evaluations for the Core Shroud and Pressure Vessel Brunswick Units 1 and 2."
5. ASME Boiler and Pressure Vessel Code, Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limits Curves," Section XI, Division 1, Approved February 26, 1999.
6. USNRC Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, (Talk ME 305-4), May 1988.
7. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components, Nonmandatory" Appendix G, "Fracture Toughness Criteria for Protection Against Failure," 1989 Edition, No Addenda.
8. 0B11-0027, Rev. 001, Development of RPV Pressure-Temperature Curves for License Renewal
9. SIA 1700147.301.R1, Feedwater Nozzle Fracture Mechanics Evaluation for Pressure-Temperature Limit Curve Development.
10. SIA 1700147.302.R2, Brunswick Nuclear Plant Unit 1 and 2 Updated P-T Curve Calculation for 54 EFPY

3.0 Body of Calculation

The development of extended power uprate (EPU) pressure-temperature (PT) curves for the Brunswick reactor pressure vessels are documented in calculation package 0B11-0005, Rev. 1 [2]. These curves are valid for up to 32 effective full power years (EFPY) and were approved by the Nuclear Regulatory Commission (NRC) on June 18, 2003.

This calculation package documents the development of RPV PT curves for up to 54 EFPY of plant operation to address the license renewal period of plant operation. The methodology utilized in the development of the curves in this calculation package is the same as utilized for the EPU pressure-temperature curves (see Ref. 2 for discussion of methodology). The curves were developed by inserting the applicable updated adjusted reference temperature values (ART_{NDT}) shown in Attachment 1 into the Ref. 2 spreadsheets. Instrument uncertainty values of 10°F and 15 psig [3] (plus an additional 15 psig for static water head) are also included in the development of the curves. The 54 EFPY limiting material in Unit 1 is plate heat number B8496 which is located in the lower intermediate shell. The limiting material in Unit 2 are the N16 nozzles, heat number Q2Q1VW. See Attachment 2 for a comparison of the beltline region and the N16 nozzles.

The curves were developed in accordance with the 1989 ASME Code Section XI, Appendix G [7], and ASME Code Case N-640 [5], which allows the use of K_{IC} for the allowable material fracture toughness.

4.0 Conclusions

This calculation provides pressure-temperature curves for core critical, core not critical, and pressure test conditions for Unit 1 and Unit 2 for up to 54 effective full power years of operation.

Duke Energy

Brunswick Steam Electric Plant Units 1 and 2

Pressure and Temperature Limits Report (PTLR)
for 54 Effective Full-Power Years (EFPY)

Revision C

Prepared by: see digital signature Date: see digital signature

Victor Bliss
[Program Engineer]

Approved by: see digital signature Date: see digital signature

John Becker
[Program Engineering Manager]

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1.0 Purpose

The purpose of the Brunswick Steam Electric Plant (BSEP) Pressure and Temperature Limits Report (PTLR) is to present operating limits relating to:

1. Reactor Coolant System (RCS) Pressure versus Temperature limits during Heat-up, Cool-down and Hydrostatic/Class 1 Leak Testing;
2. RCS Heat-up and Cool-down rates;
3. RPV head flange bolt-up temperature limits.

This report has been prepared in accordance with the requirements of Licensing Topical Reports SIR-05-044, Revision 1-A, contained within BWROG-TP-11-022-A, Revision 1 [1].

2.0 Applicability

This report is applicable to the BSEP Unit 1 and Unit 2 RPVs for up to 54 Effective Full-Power Years (EFPY).

The following BSEP Technical Specifications (TS) are affected by the information contained in this report:

TS 3.4.9 RCS Pressure/Temperature (P-T) Limits

3.0 Methodology

The limits in this report were derived as follows:

1. The methodology used is in accordance with Reference [1], "Pressure – Temperature Limits Report Methodology for Boiling Water Reactors," August 2013, incorporating the NRC Safety Evaluation in Reference [2].

2. The neutron fluence is calculated in accordance with NRC Regulatory Guide 1.190 (RG 1.190) [3] as documented in Reference [4].

3. The adjusted reference temperature (ART) values for the limiting beltline materials are calculated in accordance with NRC Regulatory Guide 1.99, Revision 2 (RG 1.99) [5], as documented in Reference [6].

4. The pressure and temperature limits, which were calculated in accordance with Reference [1], are documented in Reference [7].

5. This revision of the pressure and temperature limits report is to incorporate the following changes:

   - Revision A: Initial issue of PTLR.
   - Revision B: Reference date correction.
   - Revision C: Curve B and Curve C have been updated due to a fracture mechanics analysis for the feedwater nozzle that incorporates revised plant-specific heat transfer and fluid temperature inputs to the finite element thermal analysis.

Changes to the curves, limits, or parameters within this PTLR, based upon new irradiation fluence data of the RPV, or other plant design assumptions in the Updated Final Safety Analysis Report (UFSAR), can be made pursuant to 10 CFR 50.59 [9], provided the above methodologies are utilized. The revised PTLR shall be submitted to the NRC upon issuance.

Changes to the curves, limits, or parameters within this PTLR, based upon new surveillance capsule data of the RPV or other plant design assumption modifications in the UFSAR, cannot be made without prior NRC approval. Such analysis and revisions shall be submitted to the NRC for review prior to incorporation into the PTLR.

4.0 Operating Limits

The pressure-temperature (P-T) curves included in this report represent steam dome pressure versus minimum vessel metal temperature and incorporate the appropriate non-beltline limits and irradiation embrittlement effects in the beltline region.

The operating limits for pressure and temperature are required for three categories of operation: (a) hydrostatic pressure tests and leak tests, referred to as Curve A; (b) core not critical operation, referred to as Curve B; and (c) core critical operation, referred to as Curve C.

Complete P-T curves were developed for 54 EFPY for BSEP, as documented in Reference [7], and are provided in Figure 1 through Figure 3 for BSEP Unit 1 and in Figure 4 through Figure 6 for BSEP Unit 2. A tabulation of the curves is included in Table 1 through Table 3 for BSEP Unit 1 and in Table 4 through Table 6 for BSEP Unit 2. The adjusted reference temperature (ART) tables for 54 EFPY for the BSEP Unit 1 and Unit 2 vessel beltline materials are shown in Table 7 and Table 8, respectively [6].

The resulting P-T curves are based on the geometry, design and materials information for the BSEP Unit 1 and Unit 2 vessels with the following conditions:

- Heat-up/Cool-down rate limit during Hydrostatic Class 1 Leak Testing (Figures 1 and 4: Curve A): **≤ 25°F/hour**[1] [7].
- Normal Operating Heat-up/Cool-down rate limit (Figures 2 and 5: Curve B – non-nuclear heating, and Figures 3 and 6: Curve C – nuclear heating): The Single Relief or Safety Valve (SRV) Blowdown thermal transient event, Event No. 14 from the RPV

[1] Interpreted as the temperature change in any 1-hour period is less than or equal to 25°F.

thermal cycle diagram (TCD), has a maximum cooldown rate of **954°F/hr** and is the limiting Service Level A/B event used in the calculations of Limit Curve B and Curve C. However, for Normal Operational conditions, the RCS heat-up and cool-down rates are ≤ **100˚F/hour** in any 1 hour period.

- RPV bottom head coolant temperature to RPV coolant temperature ΔT limit during Recirculation Pump startup: **≤ 145°F** [1].
- Recirculation loop coolant temperature to RPV coolant temperature ΔT limit during Recirculation Pump startup: **≤ 50°F** [1].
- RPV flange and adjacent shell temperature limit
    - BSEP Unit 1: **> 70°F (Curve A) or ≥ 76˚F (Curves B and C)** [7].
    - BSEP Unit 2: **≥ 70˚F** [7].

Minimum temperature limits are set in accordance with 10CFR50, Appendix G [8, Table 1]. An additional 60°F margin above the requirements in Table 1 of 10CFR50, Appendix G, has been commonly applied in the BWR industry. For the BSEP closure flange material, the minimum temperature would be 76°F for Unit 1 (i.e. $RT_{NDT,max}$ of 16°F + 60°F) and 70°F for Unit 2 (i.e. $RT_{NDT,max}$ of 10°F + 60°F) [7]. For Curves A and B, this 60°F margin is a recommendation. Consequently, for Curves A and B, the minimum temperature for Unit 1 was set to 70°F for consistency with Unit 2 and with past work. These values are consistent with the minimum temperature limits and minimum bolt-up temperature in the current docketed P-T curves [10] (approved by the NRC in Reference [11]). These values also bound the lowest service temperatures (LST) for ferritic non-RPV components of the reactor coolant pressure boundary (RCPB), per the component design specifications [12], thereby addressing the NRC condition in Reference [2, Section 4.0].

5.0 Discussion

The adjusted reference temperature (ART) of the limiting beltline material is used to adjust beltline P-T curves to account for irradiation effects. RG 1.99 [5] provides the methods for determining the ART. The RG 1.99 methods for determining the limiting material and adjusting the P-T curves using ART are discussed in this section.

The vessel beltline copper (Cu) and nickel (Ni) values were obtained from the evaluation of the BSEP vessel plate, weld, and forging materials [6]. The Cu and Ni values were used with Table 1 of RG 1.99 to determine a chemistry factor (CF) per Paragraph 1.1 of RG 1.99 for welds. The Cu and Ni values were used with Table 2 of RG 1.99 to determine a CF per Paragraph 1.1 of RG 1.99 for plates and forgings.

The peak RPV ID fluence values of 3.27 x 10^{18} n/cm^2 for Unit 1 and 3.33 x 10^{18} n/cm^2 for Unit 2 at 54 EFPY used in the P-T curve evaluations were obtained from WCAP-17660-NP [4]. Fluence values in Reference [4] were calculated in accordance with RG 1.190 [3]. These fluence values apply to the limiting beltline lower intermediate shell plates (heat nos. C4487-1 and B8496-1 for Unit 1; C4489 and C4521 for Unit 2). The fluence values for the limiting beltline materials in each unit are based upon an attenuation factor of 0.719 for Unit 1 and 0.720 for Unit 2 for a postulated 1/4T flaw. Consequently, the 1/4T fluence for 54 EFPY for the limiting lower intermediate shell plates are 2.35 x 10^{18} n/cm^2 for BSEP Unit 1 and 2.40 x 10^{18} n/cm^2 for BSEP Unit 2. The limiting values for ART for beltline plates and welds are 129.1°F for Unit 1 and 103.4°F for Unit 2 [6].

The P-T limits are developed to bound all ferritic materials in the RPV, including the consideration of stress levels from structural discontinuities such as nozzles. BSEP Unit 1 and Unit 2 have a set of instrument (N16) nozzles, which are located in the lower intermediate shell beltline plates [14]. The feedwater (FW) nozzle is considered in the evaluation of the non-beltline (upper vessel) region P-T limits.

The instrument (N16) nozzle material at BSEP Units 1 and 2 is a ferritic forged nozzle design, which is welded to the RPV using a full penetration weld rather than the partial penetration nozzle design used in other plants. The effect of the penetration on the adjacent shell is considered in the development of bounding beltline P-T limits as described in Reference [7]. The instrument nozzles have an RPV ID fluence of 1.26 x 10^{18} n/cm^2 for Unit 1 and 1.28 x 10^{18} n/cm^2 for Unit 2 at 54 EFPY, obtained from Reference [4] and calculated in accordance with RG

1.190 [3]. The fluence value for the instrument nozzle location is based upon an attenuation factor of 0.719 for Unit 1 and 0.720 for Unit 2 for a postulated 1/4T flaw in the instrument nozzle blend radius. Consequently, the 1/4T fluence for 54 EFPY for the limiting instrument nozzle location is 9.06 x 10^{17} n/cm^2 for Unit 1 and 9.22 x 10^{17} n/cm^2 for Unit 2. The limiting value for ART for the instrument nozzles is 131.0°F for Unit 1 and 123.4°F for Unit 2 at 54 EFPY [6]. There are no additional forged or partial penetration nozzles in the extended beltline at BSEP Units 1 and 2.

The P-T curves for the core not critical and core critical operating conditions at a given EFPY apply for both the 1/4T and 3/4T locations. When combining pressure and thermal stresses, it is usually necessary to evaluate stresses at the 1/4T location (inside surface flaw) and the 3/4T location (outside surface flaw). This is because the thermal gradient tensile stress of interest is in the inner wall during cool-down and is in the outer wall during heat-up. However, as a conservative simplification, the thermal gradient stresses at the 1/4T location are assumed to be tensile for both heat-up and cool-down. This results in the approach of applying the maximum tensile stresses at the 1/4T location. This approach is conservative because irradiation effects cause the allowable toughness at the 1/4T to be less than that at 3/4T for a given metal temperature. This approach causes no operational difficulties, since the BWR is at steam saturation conditions during normal operation, and for a given pressure, the coolant saturation temperature is well above the P-T curve limiting temperature. Consequently, the material toughness at a given pressure would exceed the allowable toughness.

The core not critical curve (Curve B) and the core critical curve (Curve C) are prepared considering a coolant heat-up and cool-down temperature rate corresponding to the limiting Service Level A/B transient on the RPV thermal cycle diagram, the SRV Blowdown event, which has a maximum cool-down rate during the transient of 954°F/hr, although for the majority of the transient, the cool-down rate is 100°F/hr. P-T curves are developed for normal operating conditions, and historically, Technical Specifications limited operation to ≤100°F/hr. Additionally, for some BWRs, the SRV blowdown event is explicitly classified in the RPV thermal cycles as an Emergency condition. However, this was not the case for the SRV

blowdown event at Brunswick. For conservatism, the SRV blowdown event was selected for evaluation as the limiting Service Level A/B transient. For the hydrostatic pressure and leak test curve (Curve A), a coolant heat-up and cool-down temperature rate of ≤ 25°F/hr must be maintained. The P-T limits and corresponding limits of either Curve A or B may be applied, if necessary, while achieving or recovering from test conditions. So, although Curve A applies during pressure testing, the limits of Curve B may be conservatively used during pressure testing if the pressure test heat-up/cool-down rate limits cannot be maintained.

The initial RT_{NDT}, chemistry (weight-percent copper and nickel), and ART at the 1/4T location for all RPV beltline materials significantly affected by fluence (i.e., fluence > 10^{17} n/cm^2 for E > 1 MeV) are shown in Table 7 and Table 8 for Unit 1 and Unit 2, respectively [6]. Use of initial RT_{NDT} values in the determination of P-T curves for BSEP was approved by the NRC in References [15, 16].

Per Reference [6] and in accordance with Appendix A of Reference [1], the BSEP representative weld and plate surveillance materials data were reviewed from the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) [17]. The representative heat of the plate material for both BSEP Unit 1 and Unit 2 (B0673-1) in the ISP is not the same as the target plate material in BSEP Unit 1 (B8496-1) or BSEP Unit 2 (C4500-2), nor does the representative plate heat exist in the BSEP Unit 1 or Unit 2 beltlines. The representative heat of the weld material for both BSEP Unit 1 and Unit 2 (5P6756) is not the same heat number as the target vessel weld in BSEP Unit 1 (1P4218) or BSEP Unit 2 (S3986), nor does the representative weld heat exist in the BSEP Unit 1 and Unit 2 beltlines. Therefore, for all BSEP Unit 1 and Unit 2 beltline materials, the CF values are calculated using table values from R.G. 1.99, Revision 2, Position 1.1 [5].

The only computer code used in the determination of the BSEP P-T curves was the ANSYS finite element computer program:

- ANSYS, Release 14.5. (w/Service Pack 1) [18] for:

- FW nozzle (non-beltline) through-wall thermal and pressure stress distributions in Reference [19].
- Instrument nozzle thermal and pressure stress distributions in Reference [20].

ANSYS finite element analyses were used to develop the stress distributions through the FW and instrument nozzles as well as the vessel shell, and these stress distributions were used in the determination of the stress intensity factors for the FW and instrument nozzles [19, 20] and vessel shell. At the time that each of the analyses above was performed, the ANSYS program was controlled under the vendor's 10 CFR 50 Appendix B [21] Quality Assurance Program for nuclear quality-related work. Benchmarking consistent with NRC GL 83-11, Supplement 1 [22] was performed as a part of the computer program verification by comparing the solutions produced by the computer code to hand calculations for several problems.

The plant-specific BSEP FW nozzle analyses were performed to determine through-wall pressure stress distributions and thermal stress distributions due to bounding thermal transients. Detailed information regarding the analyses can be found in Reference [19]. The following summarizes the development of the thermal and pressure stress intensity factors for the FW nozzle:

- A one-quarter symmetric, three-dimensional (3-D) finite element model (FEM) of the FW nozzle was constructed and is shown in Figure 7. Details of the model and material properties are provided in Reference [19]. Temperature-dependent material properties were based on the ASME Code, Section II, Part D, 2007 Edition through 2008 Addenda [23].

- A single model was developed to bound both units. The only difference between the Unit 1 and Unit 2 FW nozzle geometries is that Unit 1 has a single welded thermal sleeve, while Unit 2 has a single thermal sleeve interference fit to the FW nozzle safe end.

- Heat transfer coefficients were calculated in Reference [19] and are a function of FW temperature and flow rate.

- With respect to operating conditions, the thermal transient which represents the maximum thermal ramp for the regions corresponding to the FW nozzles during normal and upset operating conditions were analyzed [19]. The thermal stress distributions, corresponding to the limiting times presented in Reference [19], along a linear path through the nozzle corner is used. The boundary integral equation/influence function (BIE/IF) methodology presented in Reference [1] is used to calculate the thermal stress intensity factor, K_{It}, due to the thermal stresses by fitting a third order polynomial equation to the path stress distribution for the thermal load case.
- With respect to pressure stress, a unit pressure of 1000 psig was applied to the internal surfaces of the 3-D model in Reference [19]. The pressure stress distribution was taken along a linear path through the nozzle corner. The BIE/IF methodology presented in Reference [1] was used to calculate the applied pressure stress intensity factor, K_{Ip}, by fitting a third order polynomial equation to the path stress distribution for the pressure load case. The resulting K_{Ip} may be linearly scaled to determine the K_{Ip} for various RPV internal pressures.

The plant-specific BSEP instrument nozzle analysis was performed to determine through-wall pressure stress distributions and thermal stress distributions due to bounding thermal transients. Detailed information regarding the analysis can be found in Reference [20]. The following summarizes the development of the thermal and pressure stress intensity factors for the instrument nozzle:

- A one-quarter symmetric, 3-D FEM of the instrument nozzle was constructed and is shown in Figure 8. Temperature-dependent material properties, taken from the ASME Code, Section II, Part D, 2007 Edition through 2008 Addenda [23], were used in the evaluation and are described in Reference [20].
- A single model was developed to bound both units. The instrument nozzle geometry is identical between Unit 1 and Unit 2.
- With respect to operating conditions, the bounding thermal transient for the region corresponding to the instrument nozzles during normal and upset operating conditions

was analyzed [20]. The thermal stress distribution, corresponding to the limiting time in Reference [20], along a linear path through the nozzle corner is used. The BIE/IF methodology presented in Reference [1] was used to calculate the thermal stress intensity factor, K_{It}, due to the thermal stresses by fitting a third order polynomial equation to the path stress distribution for the thermal load case.

- Boundary conditions and heat transfer coefficients used for the thermal analysis are described in Reference [20].
- With respect to pressure stress, a unit pressure of 1000 psig was applied to the internal surfaces of the FEM [20]. The pressure stress distribution was taken along the same path as the thermal stress distribution. The BIE/IF methodology presented in Reference [1] is used to calculate the pressure stress intensity factor, K_{Ip}, by fitting a third order polynomial equation to the path stress distribution for the pressure load case. The resulting K_{Ip} can be linearly scaled to determine the K_{Ip} for various RPV internal pressures

The thermal stress intensity factor for the RPV shell in the beltline region was calculated from the stress distribution output of a plant-specific analysis in Reference [20], using the LEFM solution shown in Equation 2.5.1-6 of Reference [1]. Figure 9 shows the FE model and the path through the beltline RPV shell used to extract the thermal stress distribution. Detailed information regarding the analysis can be found in Reference [20].

6.0 References

1. BWROG-TP-11-022-A, Revision 1, Pressure Temperature Limits Report Methodology for Boiling Water Reactors, August 2013. (ADAMS Accession No. ML13277A557)
2. U.S. NRC Letter to BWROG dated May 16, 2013, "Final Safety Evaluation for Boiling Water Reactor Owners' Group Topical Report BWROG-TP-11-022, Revision 1, November 2011, 'Pressure-Temperature Limits Report Methodology for Boiling Water Reactors'" (TAC NO. ME7649, ADAMS Accession No. ML13277A557).
3. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence", March 2001.
4. Westinghouse Report, WCAP-17660-NP, Revision 0, "Neutron Exposure Evaluations for Core Shroud and Pressure Vessel Brunswick Units 1 and 2," November 2012. SI File No. 1501581.201.
5. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials", May 1988.
6. SI Calculation No. 1501581.301, Revision 0, "Brunswick Nuclear Plant Unit 1 and 2 RPV Beltline ART Evaluation," November 17, 2017.
7. SI Calculation No. 1700147.302, Revision 1, "Brunswick Nuclear Plant Unit 1 and 2 Updated P-T Curve Calculation for 54 EFPY", May 29, 2018.
8. U.S. Code of Federal Regulations, Title 10, Part 50, Appendix G, "Fracture Toughness Requirements," December 12, 2013.
9. U.S. Code of Federal Regulations, Title 10, Part 50, Section 59, "Changes, tests and experiments," August 28, 2007.
10. Attachment 2 to Carolina Power & Light Letter No. BSEP 02-0121, dated June 26, 2002, SI Calculation No. CPL-54Q-303, Revision 1, "Development of Updated P-T Curves For 32 EFPY," May 13, 2002 (ADAMS Accession No. ML021890061 and ML021890087).

11. U.S. NRC License Amendments and Safety Evaluation Report, "Brunswick Steam Electric Plant, Units 1 and 2 – Issuance of Amendment Re: Pressure-Temperature Limit Curves (TAC Nos. MB5579 and MB5580)," dated June 18, 2003 (ADAMS Accession No. ML031690683).

12. Piping Design Specifications:
    a. United Engineers & Constructors Inc. Specification No. 9527-01-248-3, "Specification for Reactor Coolant Pressure Boundary Piping for Carolina Power & Light Company, Brunswick Steam Electric Plant – Units 1 and 2," dated December 2, 1974. SI File No. 1700147.201.
    b. GE Data Sheet No. 22A1295AJ, "Design Requirements for Pressure Integrity of Piping and Equipment Pressure Parts – Data Sheet," Rev. 0. SI File No. 1700147.201.
    c. GE System Design Specification No. 22A1295, "Pressure Integrity of Piping and Equipment Pressure Parts," Rev. 4. SI File No. CPL-40Q-205.

13. Brunswick Steam Electric Plant Unit 1 License Amendment No. 229 and Unit 2 License Amendment No. 257, Issuance of Amendments Regarding the Boiling Water Reactor Vessel and Surveillance Program, dated January 14, 2004 (TAC No. Nos. MC0254 and MC0255, ADAMS Accession No. ML040150192).

14. Chicago Bridge & Iron Company Drawing, Contract No. 68-2471/72, DWG No. 1, Revision 8, "General Plan – Nozzles 18'-4 ID x 69'-1-3/4 Ins Heads Nuclear Reactor Vessel – Gen. Electric Co. for Carolina Power Co. at Southport, North Carolina," October 20, 1971, SI File No. 1501581.201.

15. U.S. NRC Letter, "Closeout for Carolina Power & Light Company's Response to Generic Letter 92-01, Revision 1, Supplement 1 for the Brunswick Steam Electric Plant, Units 1 and 2 (TAC Nos. M92652 and M92653)," December 23, 1996, SI File No. 1700147.201. (ADAMS Public Legacy Library Accession No. 9612260127)

16. U.S. NRC Letter, "Closeout of TAC Nos. MA1182 and MA1183 – Response to the Requests for Additional Information to Generic Letter 92-01, Revision 1, Supplement 1,

'Reactor Vessel Structural Integrity.' for Brunswick Steam Electric Plant, Units 1 and 2," August 5, 1999, SI File No. 1700147.201. (ADAMS Public Legacy Library Accession No. 9908120115)

17. *BWRVIP-135, Revision 3: BWR Vessel and Internals Project, Integrated Surveillance Program (ISP) Data Source Book and Plant Evaluations*. EPRI, Palo Alto, CA: 2014. 3002003144. **EPRI PROPRIETARY INFORMATION**.
18. ANSYS Mechanical APDL, Release 14.5 (w/Service Pack 1 UP20120918), ANSYS, Inc., September 2012.
19. SI Calculation No. 1700147.301, Revision 1, "Feedwater Nozzle Fracture Mechanics Evaluation for Pressure-Temperature Limit Curve Development," May 29, 2018.
20. SI Calculation No. 1501581.303, Revision 0, "Water Level Instrument Nozzle and Vessel Fracture Mechanics Evaluation for Pressure-Temperature Limit Curve Development," November 17, 2017.
21. U.S. Code of Federal Regulations, Title 10, Energy, Part 50, Appendix B, "Quality Assurance for Nuclear Power Plants and Fuel Reprocessing Plants".
22. U.S. Nuclear Regulatory Commission, Generic Letter 83-11, Supplement 1, "License Qualification for Performing Safety Analyses", June 24, 1999.
23. ASME Boiler and Pressure Vessel Code, Section II, Part D, Material Properties, 2007 Edition with Addenda through 2008.
24. ASME Boiler and Pressure Vessel Code, Section II, Part D, Material Properties, 2001 Edition with Addenda through 2003.
25. U.S. Code of Federal Regulations, Title 10, Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," January 31, 2008.
26. *BWRVIP-86, Revision 1-A: BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan*. EPRI, Palo Alto, CA: 2012. 1025144. **EPRI PROPRIETARY INFORMATION.**

27. CP&L Summary Report No. SR-BSEP1-1005-001, "Brunswick Steam Electric Plant Unit 1 Reactor Pressure Vessel Surveillance Program, First Capsule Removal After 8 Fuel Cycles, Test Results and Projections," August 1994. SI File No. CPL-35Q-203.

28. Enclosure 1 to CP&L Letter BSEP 97-0051 dated February 25, 1997, "Analysis of the 300 Deg Capsule from the Carolina Power and Light Company, Brunswick Unit 2 Reactor Vessel Radiation Surveillance Program," Westinghouse Report No. WCAP-14774, November 1996. (ADAMS Accession No. 9703040242). SI File No. 1501581.202.

Figure 1: BSEP Unit 1 P-T Curve A (Hydrostatic Pressure and Leak Tests) for 54 EFPY

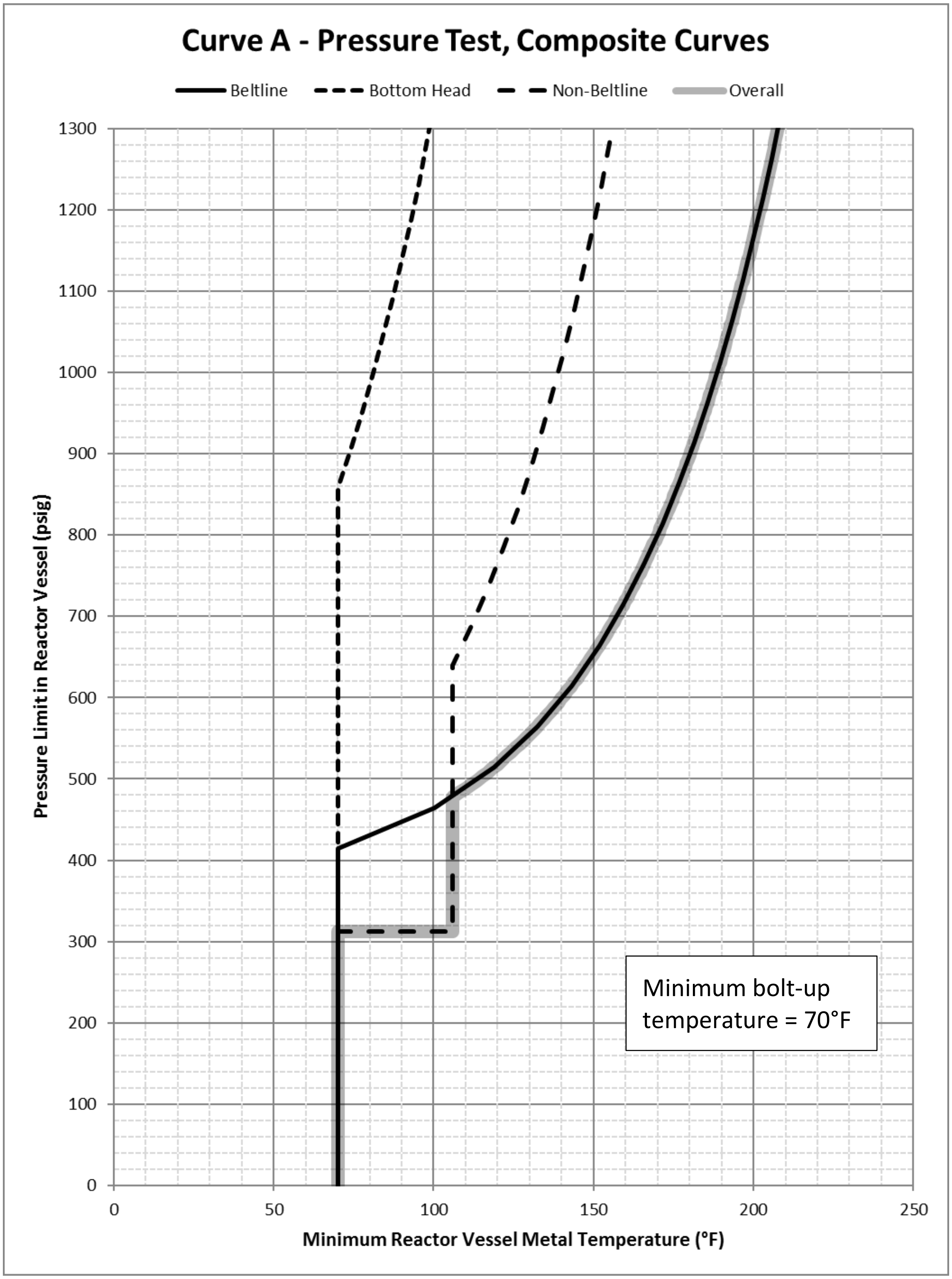


Figure 2: BSEP Unit 1 P-T Curve B (Normal Operation – Core Not Critical) for 54 EFPY

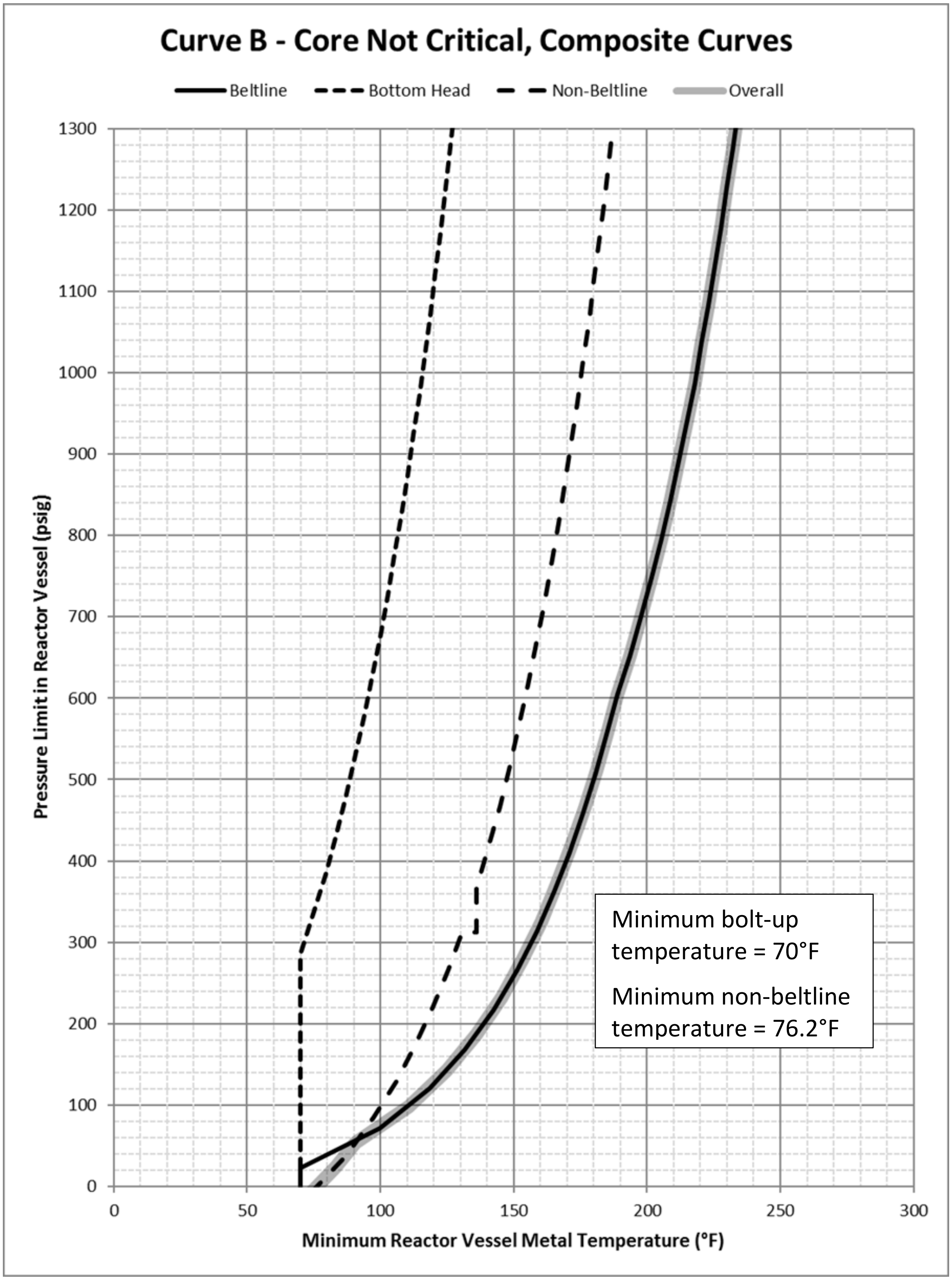


Figure 3: BSEP Unit 1 P-T Curve C (Normal Operation – Core Critical) for 54 EFPY

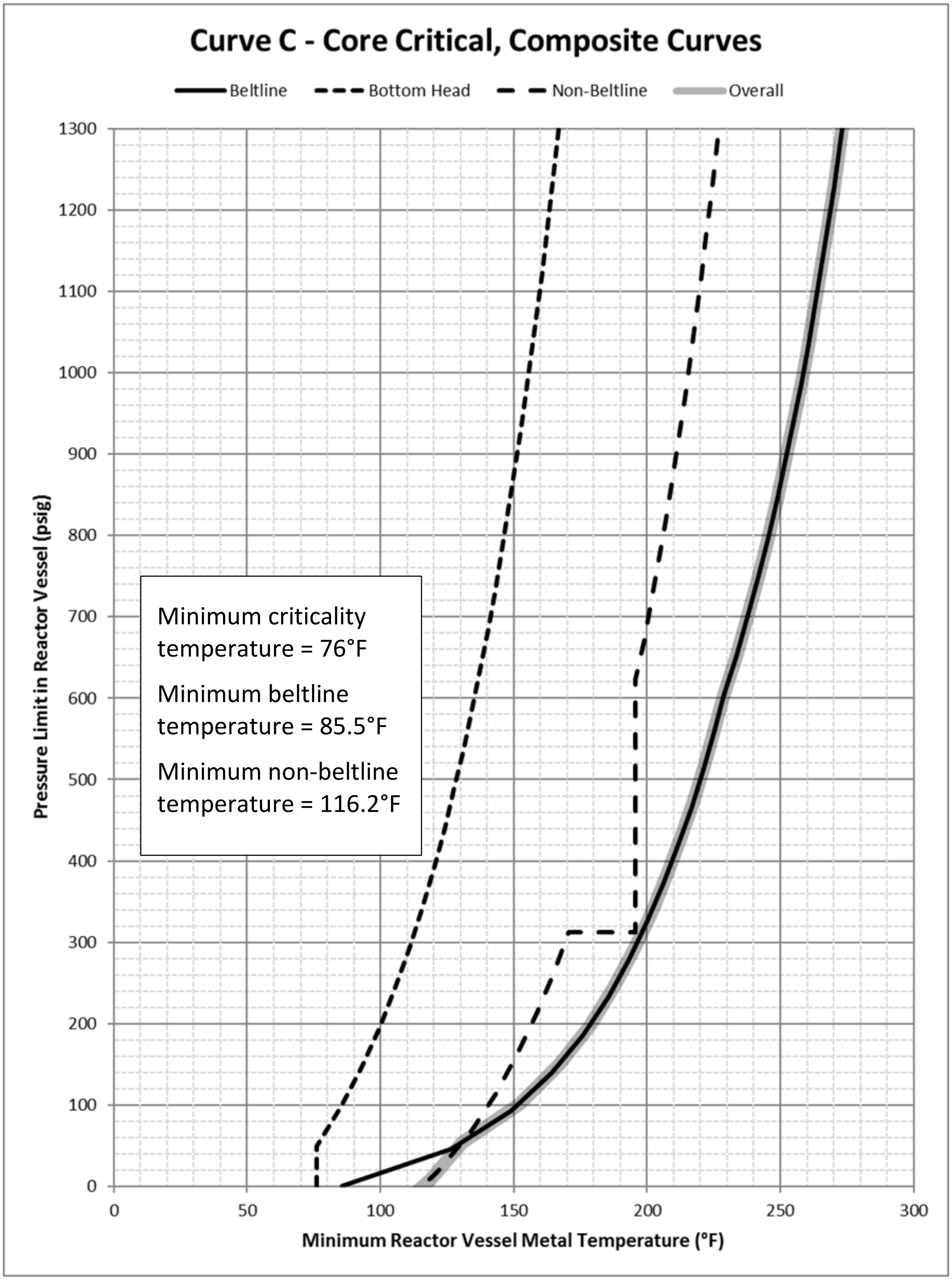


Figure 4: BSEP Unit 2 P-T Curve A (Hydrostatic Pressure and Leak Tests) for 54 EFPY

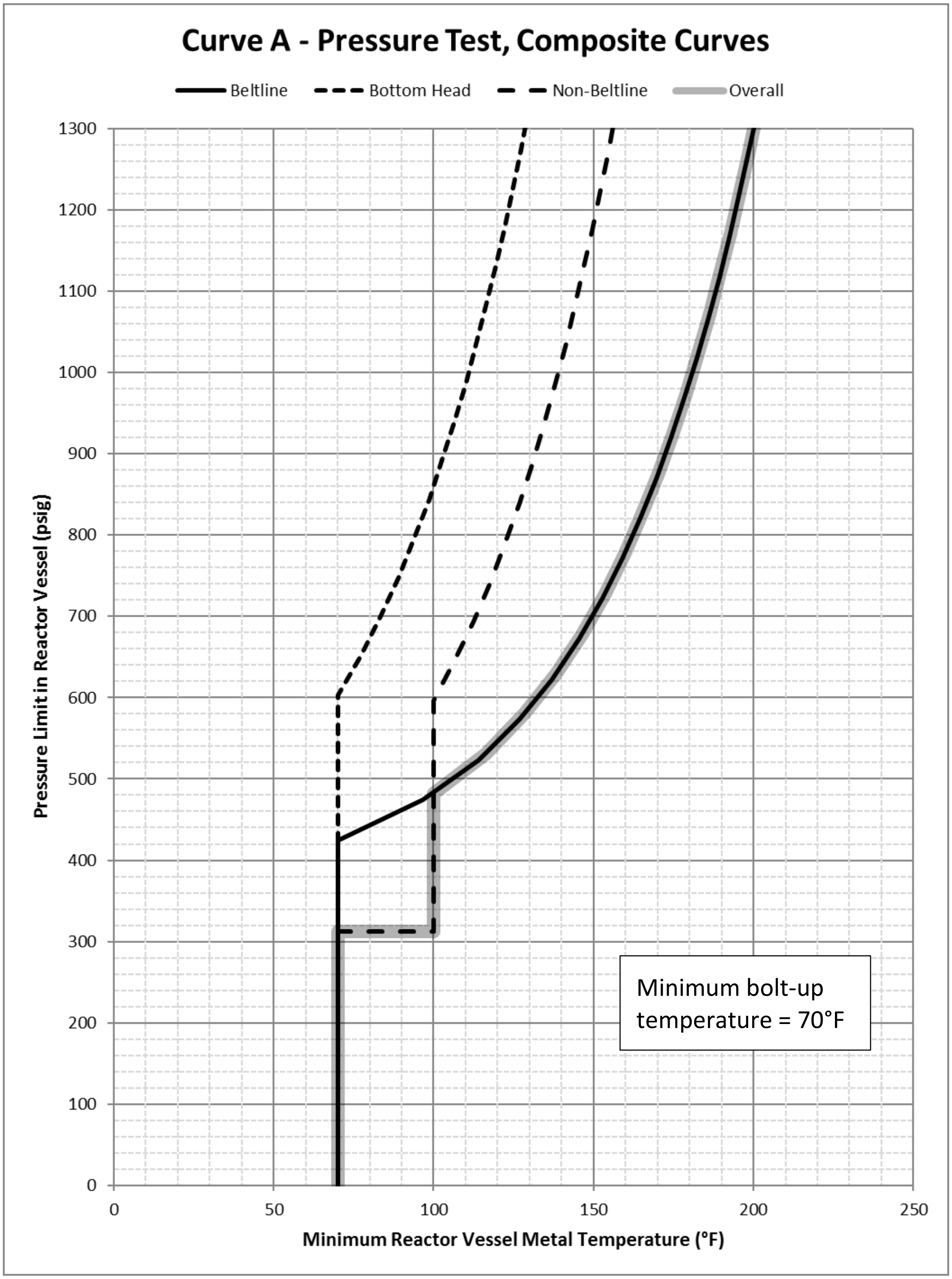


Figure 5: BSEP Unit 2 P-T Curve B (Normal Operation – Core Not Critical) for 54 EFPY

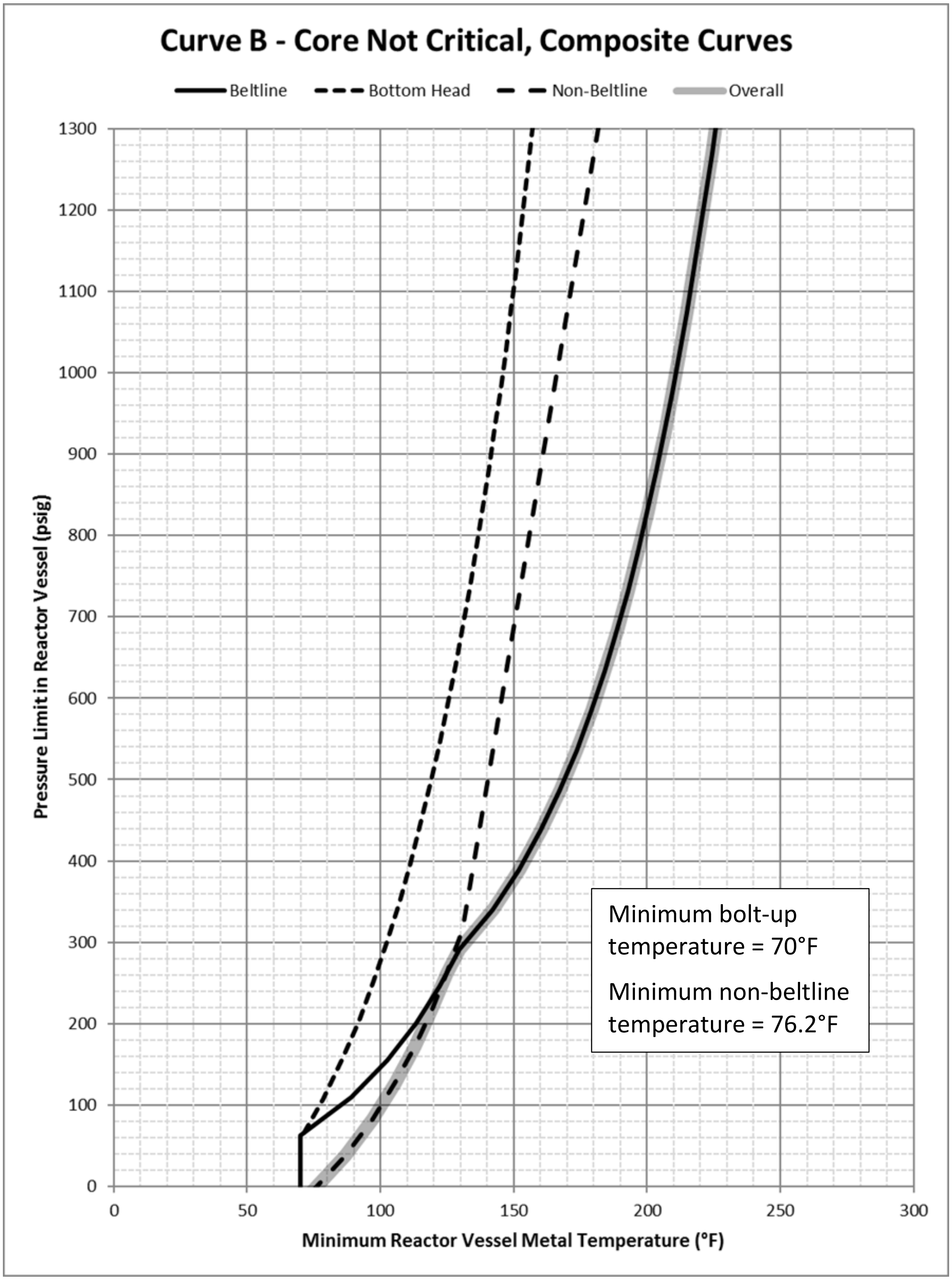


Figure 6: BSEP Unit 2 P-T Curve C (Normal Operation – Core Critical) for 54 EFPY

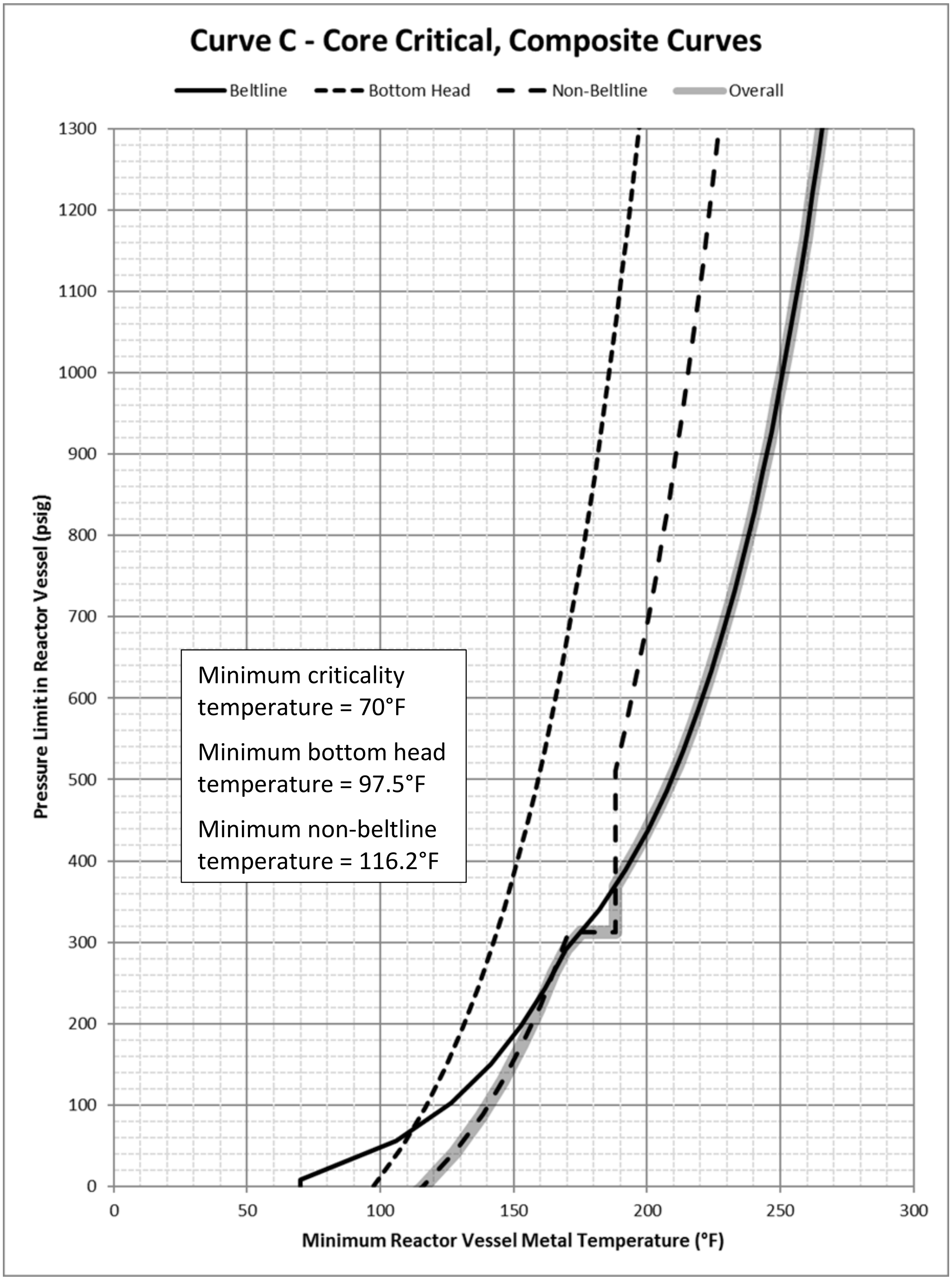
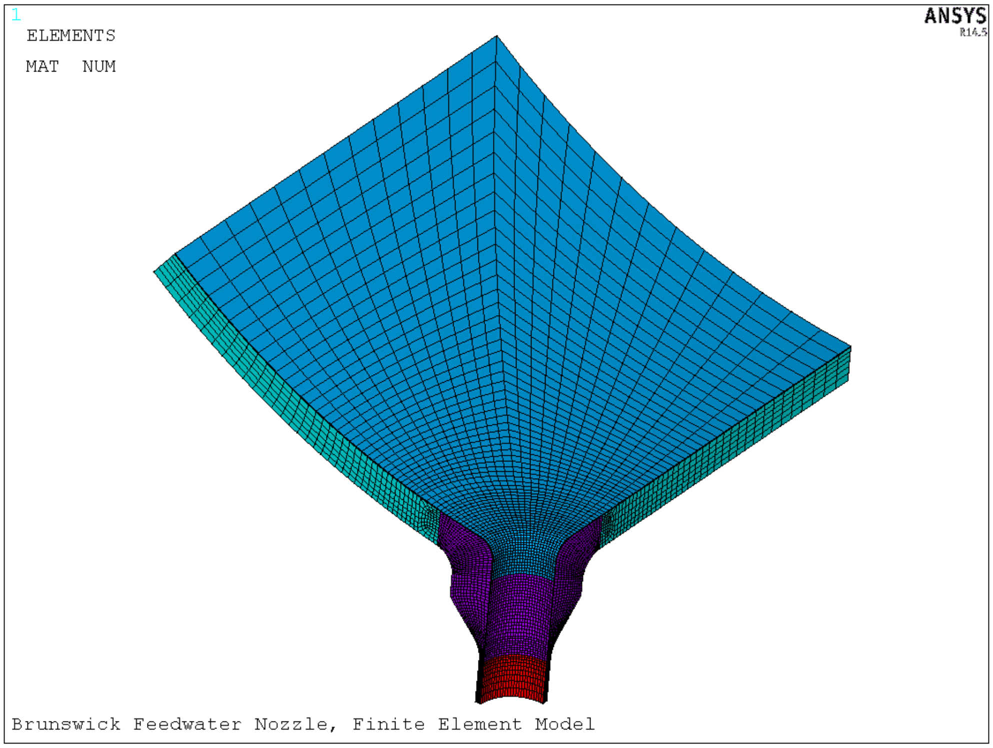


Figure 7: BSEP Feedwater Nozzle 3-D Finite Element Model [19]



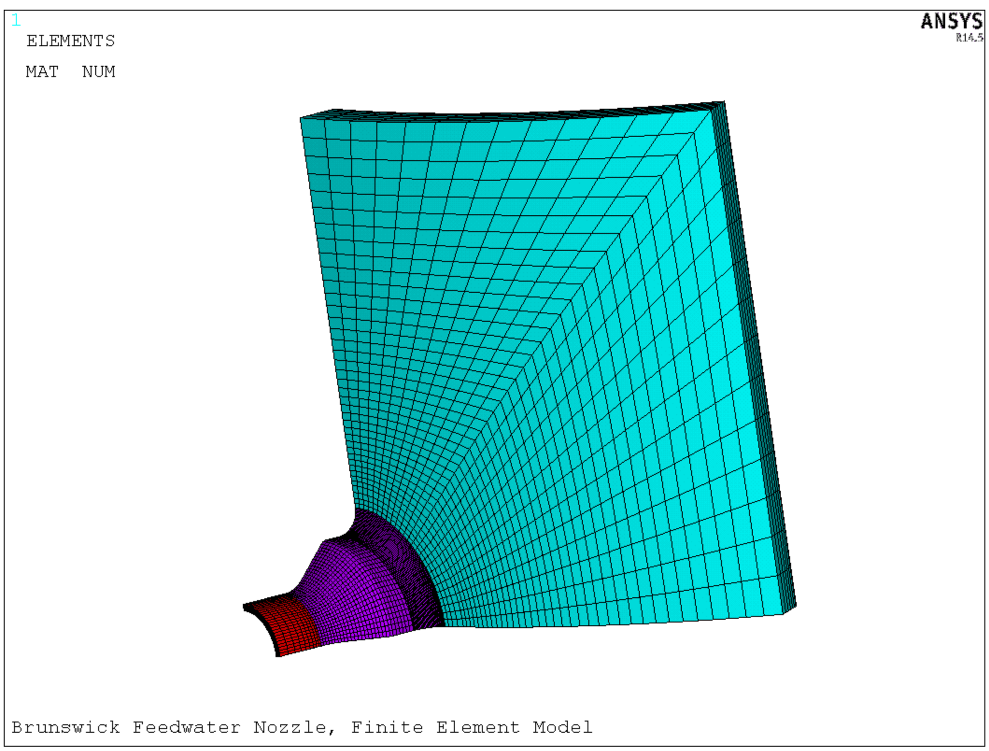
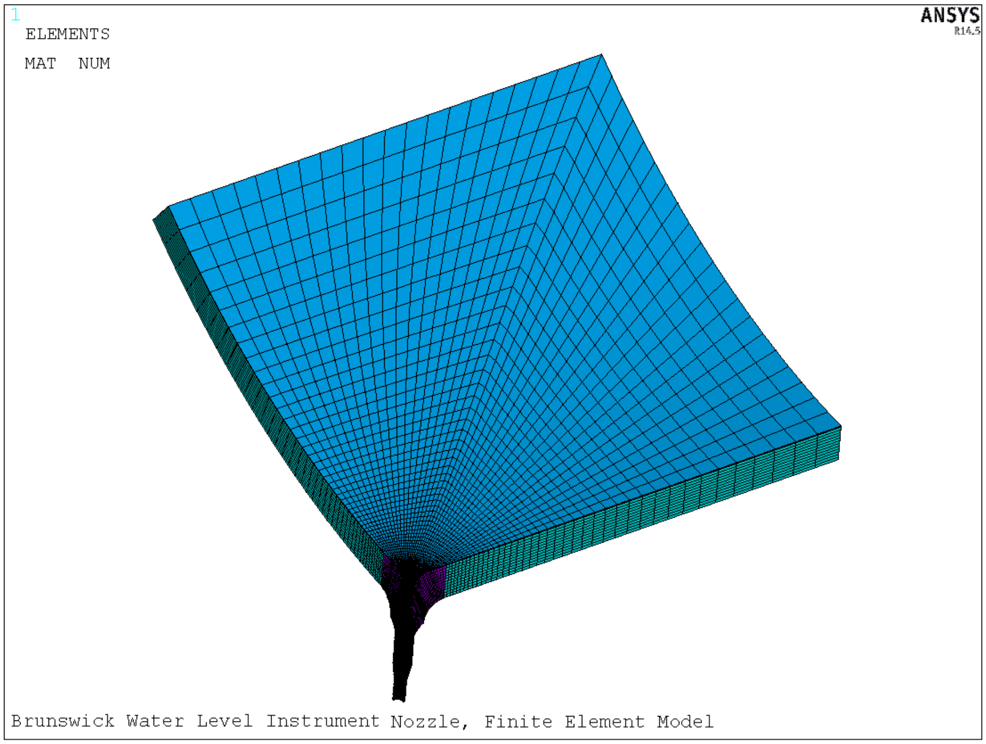


Figure 8: BSEP Instrument Nozzle Finite Element Model [20]



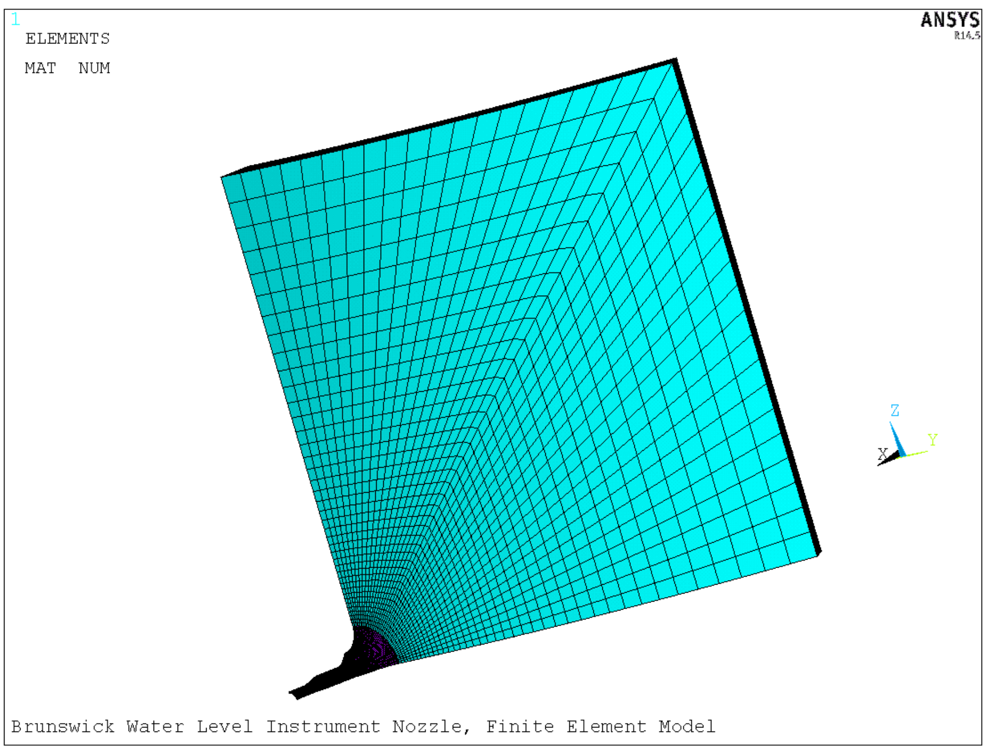


Figure 9: Vessel Path [20]

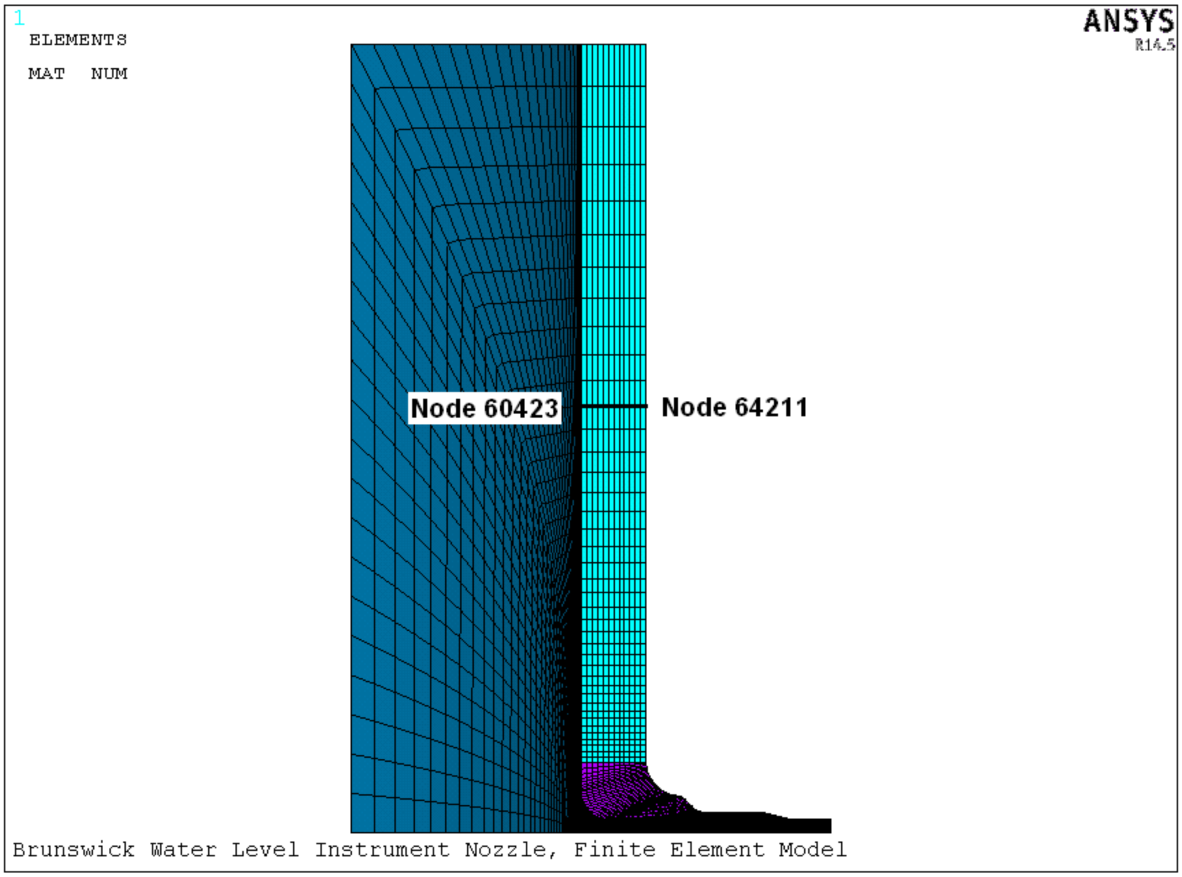


Table 1: BSEP Unit 1 Pressure Test (Curve A) P-T Curves for 54 EFPY

Beltline Region

Curve A - Pressure Test	
P-T Curve Temperature *°F*	**P-T Curve Pressure** *psi*
70.0	0.0
70.0	414.7
100.2	464.6
118.8	514.6
132.4	564.5
143.1	614.4
151.9	664.3
159.3	714.3
165.8	764.2
171.6	814.1
176.7	864.0
181.4	914.0
185.7	963.9
189.6	1013.8
193.3	1063.7
196.7	1113.7
199.8	1163.6
202.8	1213.5
205.7	1263.4
208.3	1313.4

Table 1: BSEP Unit 1 Pressure Test (Curve A) P-T Curves for 54 EFPY (continued)

Bottom Head Region

Curve A - Pressure Test	
P-T Curve Temperature *°F*	**P-T Curve Pressure** *psi*
70.0	0.0
70.0	858.3
74.0	905.2
77.6	952.2
81.1	999.2
84.3	1046.2
87.3	1093.2
90.1	1140.2
92.8	1187.1
95.4	1234.1
97.8	1281.1
100.1	1328.1

Table 1: BSEP Unit 1 Pressure Test (Curve A) P-T Curves for 54 EFPY (continued)

Non-Beltline Region

Curve A - Pressure Test	
P-T Curve Temperature *°F*	**P-T Curve Pressure** *psi*
70.0	0.0
70.0	312.6
106.0	312.6
106.0	640.3
111.9	688.9
117.2	737.4
122.0	786.0
126.4	834.6
130.4	883.1
134.2	931.7
137.6	980.2
140.9	1028.8
143.9	1077.4
146.8	1125.9
149.5	1174.5
152.1	1223.1
154.5	1271.6
156.9	1320.2

Table 2: BSEP Unit 1 Core Not Critical (Curve B) P-T Curves for 54 EFPY

Beltline Region

Curve B - Core Not Critical	
P-T Curve Temperature *°F*	**P-T Curve Pressure** *psi*
70.0	0.0
70.0	23.1
99.7	71.5
118.3	120.0
131.8	168.4
142.4	216.9
151.1	265.3
158.6	313.7
165.0	362.2
170.8	410.6
175.9	459.1
180.6	507.5
184.8	556.0
188.8	604.4
193.5	652.3
197.8	700.3
201.7	748.2
205.4	796.1
208.8	844.0
212.0	892.0
215.0	939.9
217.8	987.8
220.5	1035.8
223.1	1083.7
225.5	1131.6
227.8	1179.6
230.0	1227.5
232.1	1275.4
234.2	1323.3

Table 2: BSEP Unit 1 Core Not Critical (Curve B) P-T Curves for 54 EFPY (continued)

Bottom Head Region

Curve B - Core Not Critical

P-T Curve Temperature *°F*	**P-T Curve Pressure** *psi*
70.0	0.0
70.0	285.6
74.9	334.7
79.3	383.9
83.4	433.0
87.2	482.1
90.7	531.3
94.0	580.4
97.1	629.5
100.0	678.7
102.7	727.8
105.3	776.9
107.8	826.0
110.2	875.2
112.4	924.3
114.6	973.4
116.6	1022.6
118.6	1071.7
120.5	1120.8
122.4	1170.0
124.1	1219.1
125.8	1268.2
127.5	1317.3

Table 2: BSEP Unit 1 Core Not Critical (Curve B) P-T Curves for 54 EFPY (continued)

Non-Beltline Region

Curve B - Core Not Critical

P-T Curve Temperature °F	P-T Curve Pressure psi
76.2	0.0
87.1	38.7
97.5	84.4
106.1	130.0
113.4	175.7
119.8	221.3
125.5	267.0
130.5	312.6
136.0	312.6
136.0	367.1
140.5	416.9
144.7	466.7
148.5	516.6
152.1	566.4
155.4	616.2
158.5	666.0
161.4	715.9
164.2	765.7
166.8	815.5
169.3	865.4
171.7	915.2
174.0	965.0
176.1	1014.9
178.2	1064.7
180.2	1114.5
182.1	1164.4
184.0	1214.2
185.7	1264.0
187.5	1313.8

Table 3: BSEP Unit 1 Core Critical (Curve C) P-T Curves for 54 EFPY

Beltline Region

Curve C - Core Critical

P-T Curve Temperature *°F*	**P-T Curve Pressure** *psi*
85.5	0.0
126.6	46.5
148.8	93.0
164.1	139.5
175.9	186.0
185.4	232.5
193.3	279.0
200.2	325.4
206.2	371.9
211.6	418.4
216.5	464.9
220.9	511.4
225.0	557.9
228.8	604.4
233.5	652.3
237.8	700.3
241.7	748.2
245.4	796.1
248.8	844.0
252.0	892.0
255.0	939.9
257.8	987.8
260.5	1035.8
263.1	1083.7
265.5	1131.6
267.8	1179.6
270.0	1227.5
272.1	1275.4
274.2	1323.3

Table 3: BSEP Unit 1 Core Critical (Curve C) P-T Curves for 54 EFPY (continued)

Bottom Head Region

Curve C - Core Critical

P-T Curve Temperature *°F*	P-T Curve Pressure *psi*
76.0	0.0
76.0	49.7
85.2	98.5
92.9	147.4
99.6	196.2
105.6	245.0
110.8	293.8
115.6	342.6
120.0	391.4
124.0	440.3
127.7	489.1
131.2	537.9
134.4	586.7
137.5	635.5
140.3	684.3
143.0	733.1
145.6	782.0
148.0	830.8
150.4	879.6
152.6	928.4
154.7	977.2
156.8	1026.0
158.7	1074.8
160.6	1123.7
162.5	1172.5
164.2	1221.3
165.9	1270.1
167.5	1318.9

Table 3: BSEP Unit 1 Core Critical (Curve C) P-T Curves for 54 EFPY (continued)

Non-Beltline Region

Curve C - Core Critical

P-T Curve Temperature *°F*	P-T Curve Pressure *psi*
116.2	0.0
121.2	16.6
133.5	65.9
143.5	115.3
151.7	164.6
158.8	213.9
165.0	263.3
170.5	312.6
195.8	312.6
195.8	622.8
198.9	672.3
201.8	721.7
204.5	771.2
207.1	820.7
209.5	870.2
211.9	919.7
214.1	969.2
216.3	1018.7
218.3	1068.1
220.3	1117.6
222.2	1167.1
224.0	1216.6
225.8	1266.1
227.5	1315.6

Table 4: BSEP Unit 2 Pressure Test (Curve A) P-T Curves for 54 EFPY

Beltline Region

Curve A - Pressure Test	
P-T Curve Temperature *°F*	**P-T Curve Pressure** *psi*
70.0	0.0
70.0	424.6
96.7	474.1
114.0	523.6
126.8	573.1
137.0	622.6
145.5	672.1
152.7	721.6
159.1	771.1
164.7	820.6
169.7	870.1
174.3	919.6
178.5	969.0
182.4	1018.5
186.0	1068.0
189.3	1117.5
192.5	1167.0
195.4	1216.5
198.2	1266.0
200.8	1315.5

Table 4: BSEP Unit 2 Pressure Test (Curve A) P-T Curves for 54 EFPY (continued)

Bottom Head Region

Curve A - Pressure Test

P-T Curve Temperature *°F*	P-T Curve Pressure *psi*
70.0	0.0
70.0	604.6
77.1	652.6
83.3	700.5
88.9	748.4
93.9	796.3
98.4	844.2
102.6	892.2
106.4	940.1
110.0	988.0
113.3	1035.9
116.4	1083.8
119.3	1131.7
122.1	1179.7
124.7	1227.6
127.2	1275.5
129.6	1323.4

Table 4: BSEP Unit 2 Pressure Test (Curve A) P-T Curves for 54 EFPY (continued)

Non-Beltline Region

Curve A - Pressure Test

P-T Curve Temperature *°F*	**P-T Curve Pressure** *psi*
70.0	0.0
70.0	312.6
100.0	312.6
100.0	596.7
106.6	645.1
112.5	693.4
117.7	741.7
122.4	790.0
126.7	838.3
130.7	886.6
134.4	934.9
137.8	983.2
141.1	1031.6
144.1	1079.9
146.9	1128.2
149.6	1176.5
152.2	1224.8
154.6	1273.1
156.9	1321.4

Table 5: BSEP Unit 2 Core Not Critical (Curve B) P-T Curves for 54 EFPY

Beltline Region

Curve B - Core Not Critical

P-T Curve Temperature *°F*	**P-T Curve Pressure** *psi*
70.0	0.0
70.0	63.1
88.9	108.7
102.5	154.4
113.2	200.0
122.1	245.7
129.6	291.3
142.0	340.2
151.9	389.1
160.2	438.0
167.3	486.9
173.6	535.8
179.1	584.8
184.1	633.7
188.6	682.6
192.8	731.5
196.6	780.4
200.2	829.3
203.5	878.2
206.6	927.1
209.6	976.1
212.3	1025.0
215.0	1073.9
217.5	1122.8
219.8	1171.7
222.1	1220.6
224.3	1269.5
226.3	1318.4

Table 5: BSEP Unit 2 Core Not Critical (Curve B) P-T Curves for 54 EFPY (continued)

Bottom Head Region

Curve B - Core Not Critical

P-T Curve Temperature *°F*	P-T Curve Pressure *psi*
70.0	0.0
70.0	58.4
78.4	106.9
85.6	155.4
91.9	204.0
97.5	252.5
102.5	301.1
107.1	349.6
111.3	398.1
115.2	446.7
118.7	495.2
122.1	543.7
125.2	592.3
128.2	640.8
131.0	689.3
133.6	737.9
136.1	786.4
138.5	835.0
140.8	883.5
143.0	932.0
145.0	980.6
147.0	1029.1
149.0	1077.6
150.8	1126.2
152.6	1174.7
154.3	1223.2
156.0	1271.8
157.6	1320.3

Table 5: BSEP Unit 2 Core Not Critical (Curve B) P-T Curves for 54 EFPY (continued)

Non-Beltline Region

Curve B - Core Not Critical

P-T Curve Temperature *°F*	P-T Curve Pressure *psi*
76.2	0.0
86.9	38.0
97.2	82.9
105.7	127.9
113.0	172.8
119.3	217.7
124.9	262.6
130.0	307.6
130.5	312.6
130.5	312.6
134.0	346.4
138.0	388.5
142.0	434.2
146.0	483.6
150.0	537.1
154.0	595.1
158.0	657.9
162.0	726.0
166.0	799.7
170.0	879.6
174.0	966.1
178.0	1059.8
182.0	1161.4
186.0	1271.4
190.0	1390.5

Table 6: BSEP Unit 2 Core Critical (Curve C) P-T Curves for 54 EFPY

Beltline Region

Curve C - Core Critical	
P-T Curve Temperature *°F*	**P-T Curve Pressure** *psi*
70.0	0.0
70.0	8.3
106.0	55.4
126.7	102.6
141.3	149.8
152.6	196.9
161.8	244.1
169.6	291.3
182.0	340.2
191.9	389.1
200.2	438.0
207.3	486.9
213.6	535.8
219.1	584.8
224.1	633.7
228.6	682.6
232.8	731.5
236.6	780.4
240.2	829.3
243.5	878.2
246.6	927.1
249.6	976.1
252.3	1025.0
255.0	1073.9
257.5	1122.8
259.8	1171.7
262.1	1220.6
264.3	1269.5
266.3	1318.4

Table 6: BSEP Unit 2 Core Critical (Curve C) P-T Curves for 54 EFPY (continued)

Bottom Head Region

Curve C - Core Critical

P-T Curve Temperature *°F*	**P-T Curve Pressure** *psi*
97.5	0.0
108.2	48.8
116.9	97.7
124.4	146.5
130.9	195.4
136.6	244.2
141.8	293.1
146.4	341.9
150.7	390.8
154.6	439.6
158.3	488.4
161.7	537.3
164.8	586.1
167.8	635.0
170.7	683.8
173.3	732.7
175.9	781.5
178.3	830.3
180.6	879.2
182.8	928.0
184.9	976.9
186.9	1025.7
188.9	1074.6
190.7	1123.4
192.5	1172.3
194.3	1221.1
196.0	1269.9
197.6	1318.8

Table 6: BSEP Unit 2 Core Critical (Curve C) P-T Curves for 54 EFPY (continued)

Non-Beltline Region

Curve C - Core Critical

P-T Curve Temperature *°F*	P-T Curve Pressure *psi*
116.2	0.0
128.3	43.7
138.3	88.5
146.6	133.3
153.8	178.1
160.0	223.0
165.5	267.8
170.5	312.6
188.1	312.6
188.1	511.2
191.5	559.0
194.8	606.8
197.8	654.6
200.6	702.4
203.3	750.2
205.9	798.0
208.3	845.8
210.7	893.6
212.9	941.5
215.0	989.3
217.1	1037.1
219.0	1084.9
220.9	1132.7
222.7	1180.5
224.5	1228.3
226.2	1276.1
227.8	1323.9

Table 7: BSEP Unit 1 ART Table for 54 EFPY

	Description	ID No.	Heat No.	Lot No.	Initial RT_{NDT} (°F)	Chemistry		Chemistry Factor (°F)	Adjustments For 1/4t			
									ΔRT_{NDT}	Margin Terms		ART
						Cu (wt %)	Ni (wt %)		(°F)	σ_Δ (°F)	σ_i (°F)	(°F)
Plates	Lower Shell	201	C4535-2	-	34	0.12	0.58	83	45.6	17.0	0.0	113.6
	Lower Shell	251	C4550-1	-	10	0.11	0.60	74	40.9	17.0	0.0	84.9
	Lower Int. Shell	301	C4487-1	-	10	0.12	0.56	82	50.0	17.0	0.0	94.0
	Lower Int. Shell	351	B8496-1	-	10	0.19	0.58	140	85.1	17.0	0.0	129.1
	Upper Int. Shell	401	C4510-2	-	22	0.35	0.58	210	26.4	13.2	1.0	74.9
	Upper Int. Shell	451	C4515-2	-	10	0.35	0.53	203	25.6	12.8	2.0	61.5
Welds	Lower Int. Vertical	F1 F2	S3986	3876 Run 934	10	0.05	0.96	68	35.0	17.5	0.0	80.0
	Lower Vertical	G1 G2	S3986	3876 Run 934	10	0.05	0.96	68	30.9	15.5	0.0	71.8
	Upper Int. to Lower Int. Girth	EF	S3986	3876 Run 934	10	0.05	0.96	68	8.6	4.3	0.0	27.1
	Lower to Lower Int. Girth	FG	1P4218	3929 Run 989	-50	0.06	0.87	82	47.0	23.5	0.0	43.9
Nozzles	Nozzle N16A and N16B	-	Q2Q1VW	247P-4A 247P-4B	48	0.16	0.82	123	49.0	17.0	0.0	131.0
Nozzle Welds	Nozzle N16A and N16B	-	977987	-	-50	0.03	1.04	41	16.3	8.1	0.0	-17.4
		-	650x006	J807A27A	-50	0.03	0.96	41	16.3	8.1	0.0	-17.4

	Location		Wall Thickness (in.)		Fluence at ID	Attenuation, 1/4t	Fluence @ 1/4t	Fluence Factor, FF
			Full	1/4t	(n/cm²)	$e^{-0.24x}$	(n/cm²)	$f^{(0.28-0.10\log f)}$
Plates	Lower Shell	201	5.496	1.374	2.59E+18	0.719	1.86E+18	0.553
	Lower Shell	251	5.496	1.374	2.59E+18	0.719	1.86E+18	0.553
	Lower Int. Shell	301	5.496	1.374	3.27E+18	0.719	2.35E+18	0.609
	Lower Int. Shell	351	5.496	1.374	3.27E+18	0.719	2.35E+18	0.609
	Upper Int. Shell	401	5.496	1.374	1.71E+17	0.719	1.23E+17	0.126
	Upper Int. Shell	451	5.496	1.374	1.71E+17	0.719	1.23E+17	0.126
Welds	Lower Int. Vertical	F1F2	5.496	1.374	2.20E+18	0.719	1.58E+18	0.515
	Lower Vertical	G1 G2	5.496	1.374	1.67E+18	0.719	1.20E+18	0.454
	Upper Int. to Lower Int. Girth	EF	5.496	1.374	1.71E+17	0.719	1.23E+17	0.126
	Lower to Lower Int. Girth	FG	5.496	1.374	2.82E+18	0.719	2.03E+18	0.573
Nozzles	Nozzle N16A and N16B	-	5.496	1.374	1.26E+18	0.719	9.06E+17	0.397
Nozzle Welds	Nozzle N16A and N16B	-	5.496	1.374	1.26E+18	0.719	9.06E+17	0.397

Table 8: BSEP Unit 2 ART Table for 54 EFPY

	Description	ID No.	Heat No.	Lot No.	Initial RT_{NDT} (°F)	Chemistry Cu (wt %)	Chemistry Ni (wt %)	Chemistry Factor (°F)	Adjustments For 1/4t: ΔRT_{NDT} (°F)	Margin Terms σ_Δ (°F)	Margin Terms σ_i (°F)	ΔRT_{NDT} (°F)
Plates	Lower Shell	201	C4500	-	10	0.15	0.54	107	59.4	17.0	0.0	103.4
	Lower Shell	251	C4550	-	10	0.11	0.60	74	41.2	17.0	0.0	85.2
	Lower Int. Shell	301	C4489	-	10	0.12	0.60	83	50.9	17.0	0.0	94.9
	Lower Int. Shell	351	C4521	-	10	0.12	0.57	82	50.6	17.0	0.0	94.6
	Upper Int. Shell	401	C4854-2	-	10	0.35	0.56	207	26.4	13.2	0.0	62.8
	Upper Int. Shell	451	C4862-2	-	10	0.35	0.58	210	26.7	13.4	0.0	63.5
Welds	Lower Int. Vertical	F1 F2	S-3986	3876 Run 934	10	0.05	0.96	68	34.8	17.4	0.0	79.5
	Lower Vertical	G1 G2	S-3986	3876 Run 934	10	0.05	0.96	68	30.7	15.3	0.0	71.3
	Upper Int. to Lower Int. Girth	EF	S-3986	3876 Run 934	10	0.05	0.96	68	8.7	4.3	0.0	27.3
	Lower to Lower Int. Girth	FG	3P4000	3932 Run 989	-50	0.02	0.90	27	15.6	7.8	0.0	-18.7
Nozzles	Nozzle N16A and N16B	-	Q2Q1VW	247P-3A 247P-3B	40	0.16	0.82	123	49.4	17.0	0.0	123.4
Nozzle Welds	Nozzle N16A and N16B	-	01R496		-50	0.02	0.92	27	10.8	5.4	0.0	-28.4
		-	82D913		-50	0.03	0.80	41	16.4	8.2	0.0	-17.1

	Location		Wall Thickness (in.) Full	Wall Thickness (in.) 1/4t	Fluence at ID (n/cm^2)	Attenuation, 1/4t $e^{-0.24x}$	Fluence @ 1/4t (n/cm^2)	Fluence Factor, FF $f^{(0.28-0.10\log f)}$
Plates	Lower Shell	201	5.466	1.367	2.63E+18	0.720	1.89E+18	0.557
	Lower Shell	251	5.466	1.367	2.63E+18	0.720	1.89E+18	0.557
	Lower Int. Shell	301	5.466	1.367	3.33E+18	0.720	2.40E+18	0.614
	Lower Int. Shell	351	5.466	1.367	3.33E+18	0.720	2.40E+18	0.614
	Upper Int. Shell	401	5.466	1.367	1.74E+17	0.720	1.25E+17	0.128
	Upper Int. Shell	451	5.466	1.367	1.74E+17	0.720	1.25E+17	0.128
Welds	Lower Int. Vertical	F1F2	5.466	1.367	2.16E+18	0.720	1.56E+18	0.511
	Lower Vertical	G1 G2	5.466	1.367	1.64E+18	0.720	1.18E+18	0.451
	Upper Int. to Lower Int. Girth	EF	5.466	1.367	1.74E+17	0.720	1.25E+17	0.128
	Lower to Lower Int. Girth	FG	5.466	1.367	2.89E+18	0.720	2.08E+18	0.579
Nozzles	Nozzle N16A and N16B	-	5.466	1.367	1.28E+18	0.720	9.22E+17	0.401
Nozzle Welds	Nozzle N16A and N16B	-	5.466	1.367	1.28E+18	0.720	9.22E+17	0.401

Table 9: Nozzle Stress Intensity Factors

Nozzle	Applied Pressure, $K_{Ip\text{-}app}$	Thermal, K_{It}
Feedwater	73.74	50.05
Instrument (N16)	55.42	15.13

K_I in units of ksi-in$^{0.5}$

Appendix A

BRUNSWICK REACTOR VESSEL MATERIALS SURVEILLANCE PROGRAM

In accordance with 10 CFR 50, Appendix H, Reactor Vessel Material Surveillance Program Requirements [25], one surveillance capsule has been removed and tested from each of the BSEP Unit 1 and Unit 2 reactor vessels. The first BSEP Unit 1 capsule was removed in Summer 1993, at 8.67 EFPY [26, 27] and the first BSEP Unit 2 capsule was removed in Spring 1996, at 10.9 EFPY [26, 28]. The surveillance capsules contained flux wires for neutron fluence measurement, Charpy V-Notch impact test specimens and uniaxial tensile test specimens fabricated using materials from the vessel materials within the core beltline region. The methods and results of testing are presented in References [27, 28]. In BSEP Units 1 and 2, there are two remaining capsules in each vessel which will remain in place to serve as backup surveillance material for the BWRVIP program, or as otherwise needed.

BSEP has replaced the original RPV material surveillance program with the BWRVIP ISP [25]. BSEP is currently committed to use the BWRVIP ISP, and has made a licensing commitment to use the ISP for BSEP during the period of extended operation. The BWRVIP ISP meets the requirements of 10 CFR 50, Appendix H, for Integrated Surveillance Programs, and has been approved by NRC. BSEP committed to use the ISP in place of its existing surveillance programs in the license amendment issued by the NRC regarding the implementation of the BWRVIP ISP, dated January 14, 2004 [13]. Under the ISP, no further capsules are scheduled for removal from the BSEP Unit 1 and 2 vessels. Representative surveillance capsule materials for the BSEP Unit 1 and 2 limiting beltline weld are contained in the River Bend and Supplemental Surveillance Program (SSP) Capsules C, F, and H. Representative materials for the BSEP Unit 1 and 2 limiting beltline plate are in the Duane Arnold and SSP-F surveillance capsules. No further SSP capsules are scheduled for withdrawal. The next River Bend surveillance capsule is scheduled to be withdrawn and tested under the ISP in approximately 2025, and the next Duane Arnold surveillance capsule is scheduled for withdrawal and testing in approximately 2027 [25].