Enclosure 2

Cooper Nuclear Station Pressure and Temperature Limits Report (PTLR) for 54 Effective Full-Power Years (EFPY) (Non-Proprietary)

Cooper Nuclear Station Docket No. 50-298, DPR-46

Engineering Report No. 2016-042 Rev 1
Page 1 of 31

Engineering Report Cover Sheet

Engineering Report Title:
Cooper Nuclear Station Pressure and Temperature Limits Report (PTLR) for 54 Effective Full-Power Years (EFPY)
(Non-Proprietary)

Engineering Report Type: (3)

New ☐ Revision ☒ Cancelled ☐ Superseded ☐
Superseded by: ________

Revision 1: Removed "Proprietary Brackets" from Table 4 values for Cu and Ni content.

(6)
ECR No. N/A **EC No.** 16-46

(4) **Report Origin:** ☒ CNS ☐ Vendor
Vendor Document No.: N/A

(5) **Quality-Related:** ☒ Yes ☐ No

Prepared by: Tim McClure/ Tim McClure Date: 11-17-16
Responsible Engineer (Print Name/Sign)

Design Verified: Phil Leininger/ Date: 12/1/16
Design Verifier (if required) (Print Name/Sign)

Reviewed by: N/A Date:
Reviewer (Print Name/Sign)

Approved by: Troy Barker/ Troy Barker Date: 12/1/16
Manager (Print Name/Sign)

Table of Contents

Section		Page
1.0	Purpose	3
2.0	Applicability	3
3.0	Methodology	4
4.0	Operating Limits	5
5.0	Discussion	6
6.0	References	12
Figure 1	CNS P-T Curve A (Hydrostatic Pressure and Leak Tests) for 54 EFPY	16
Figure 2	CNS P-T Curve B (Normal Operation – Core Not Critical) for 54 EFPY	17
Figure 3	CNS P-T Curve C (Normal Operation – Core Critical) for 54 EFPY	18
Figure 4	Cooper Feedwater Nozzle Finite Element Model [19]	19
Figure 5	Cooper Core Differential Nozzle Finite Element Model [20]	20
Table 1	CNS Pressure Test (Curve A) P-T Curves for 54 EFPY	21
Table 2	CNS Core Not Critical (Curve B) P-T Curves for 54 EFPY	24
Table 3	CNS Core Critical (Curve C) P-T Curves for 54 EFPY	27
Table 4	CNS ART Calculations for 54 EFPY	30
Appendix A	Cooper Reactor Vessel Materials Surveillance Program	31
Appendix B	BWRVIP-135, Revision 3: BWR Vessel and Internals Integrated Surveillance Program (ISP) Data Source Book and Plant Evaluations, Technical Report No. 3002003144, December 2014 (Non-Proprietary, Pages 1 - 45)	

1.0 Purpose

The purpose of the Cooper Nuclear Station (CNS) Pressure and Temperature Limits Report (PTLR) is to present operating limits relating to:

1. Reactor Coolant System (RCS) Pressure versus Temperature limits during Heatup, Cooldown and Hydrostatic/Class 1 Leak Testing;
2. RCS Heatup and Cooldown rates;
3. RPV head flange boltup 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], and 0900876.401, Revision 0-A, contained within BWROG-TP-11-023-A, Revision 0 [2].

2.0 Applicability

This report is applicable to the CNS RPV for up to 54 Effective Full-Power Years (EFPY).

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

TS RCS Pressure and Temperature (P-T) Limits

TS Surveillance Requirements

3.0 Methodology

The limits in this report were derived as follows:

1. The methodology used is in accordance with Reference [1] and Reference [2], incorporating the NRC Safety Evaluations in References [3] and [4], respectively.
2. The neutron fluence is calculated in accordance with NRC Regulatory Guide 1.190 (RG 1.190) [5], using the RAMA computer code, as documented in Reference [6].
3. The adjusted reference temperature (ART) values for the limiting beltline materials are calculated in accordance with NRC Regulatory Guide 1.99, Revision 2 [7], as documented in Reference [8].
4. The pressure and temperature limits were calculated in accordance with Reference [1], "Pressure – Temperature Limits Report Methodology for Boiling Water Reactors," June 2013, as documented in NPPD Calculation NEDC 07-048, Reference [9].
5. This revision of the pressure and temperature limits is to incorporate the following changes:
   - Update pressure and temperature limits for 54 EFPY.

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 [10], 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 Cooper Nuclear Station, as documented in Reference [9]. The CNS P-T curves for 54 EFPY are provided in Figures 1 through 3, and a tabulation of the curves is included in Tables 1 through 3. The adjusted reference temperature (ART) tables for the CNS vessel beltline materials are shown in Table 4 for 54 EFPY (Reference [8]). The resulting P-T curves are based on the geometry, design and materials information for the CNS vessel with the following conditions:

- Heatup and Cooldown rate limit during Hydrostatic Class 1 Leak Testing (Figure 1: Curve A): **≤ 25°F/hour**[1] [9].
- Normal Operating Heatup and Cooldown rate limit (Figure 2: Curve B – non-nuclear heating, and Figure 3: Curve C – nuclear heating): **≤ 100°F/hour**[2] [9].

---

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

[2] Interpreted as the temperature change in any 1-hour period is less than or equal to 100°F.

- RPV bottom head coolant temperature to RPV coolant temperature ΔT limit during Recirculation Pump startup: **≤ 145°F.**
- Recirculation loop coolant temperature to RPV coolant temperature ΔT limit during Recirculation Pump startup: **≤ 50°F.**
- RPV flange and adjacent shell temperature limit ≥ **70°F** [9].

To address the NRC condition regarding lowest service temperature in Reference [3, Section 4.0], the minimum temperature is set to 70°F for Curves A and B, which bounds $RT_{NDT,max}$ and the CNS shutdown margin analysis, and 80°F for Curve C, which is equal to $RT_{NDT,max}$ + 60°F. These values are consistent with the minimum temperature limits approved for use by the NRC in Reference [11].

The composite P-T curves are extended below 0 psig to -14.7 psig based on the evaluation documented in Reference [12], which demonstrates that the P-T curves are applicable to negative gauge pressures. A pressure of -14.7 psig bounds the maximum expected vacuum pressure as well as externally applied pressures the RPV may experience. Since the P-T curve calculation methods used do not specifically apply to negative values of pressure, the tabulated results start at 0 psig. However, the minimum analyzed RPV pressure is -14.7 psig

5.0 <u>Discussion</u>

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 [7] 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 CNS vessel plate, weld, and forging materials [8]. This evaluation included the results of two

surveillance capsules for the representative plate material and three surveillance capsules for the representative weld material. 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. However, the fitted CF for the limiting plate (which is based on credible surveillance data) in the CNS vessel bounds the RG 1.99 CF. Therefore, the fitted CF is used for the limiting beltline plate.

The peak RPV ID fluence value of 2.23 x 10^{18} n/cm^2 at 54 EFPY used in the P-T curve evaluation were obtained from Reference [6] and are calculated in accordance with RG 1.190 [5]. These fluence values apply to the limiting beltline lower intermediate shell plate (Heat No. C2307-2). The fluence values for the lower intermediate shell plate are based upon an attenuation factor of 0.72 for a postulated 1/4T flaw. As a result, the 1/4T fluence for 54 EFPY for the limiting lower intermediate shell plate is 1.62 x 10^{18} n/cm^2 for CNS.

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. The water level instrument (WLI) nozzle is located in the lower-intermediate shell beltline plates [9]. The nozzle material is not ferritic, however the effect of the penetration on the adjacent shell is considered according to the methodology in Reference [2]. The RPV ID fluence value of 5.44 x 10^{17} n/cm^2 at 54 EFPY used in the P-T curve evaluation of the WLI nozzle was obtained from Reference [6] and is calculated in accordance with RG 1.190 [5]. This fluence value applies to the limiting WLI nozzle location (Heat No. EV-26067). The fluence value for the WLI nozzle location is based upon an attenuation factor of 0.72 for a postulated 1/4T flaw. As a result, the 1/4T fluence for 54 EFPY for the limiting WLI nozzle location is 3.94 x 10^{17} n/cm^2 for CNS. There are no additional forged or partial penetration nozzles in the extended beltline.

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 cooldown and is in the outer wall during heatup. However, as a conservative simplification, the thermal gradient stresses at the 1/4T location are assumed to be tensile for both heatup and cooldown. 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.

For the core not critical curve (Curve B) and the core critical curve (Curve C), the P-T curves specify a coolant heatup and cooldown temperature rate of ≤ 100°F/hour for which the curves are applicable. However, the core not critical and the core critical curves were also developed to bound Service Level A/B RPV thermal transients. For the hydrostatic pressure and leak test curve (Curve A), a coolant heatup and cooldown temperature rate of ≤ 25°F/hour 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 heatup/cooldown rate limits cannot be maintained.

The initial RT_{NDT}, the 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 > 1MeV) are shown in Table 4 for 54 EFPY [8]. Initial RT_{NDT} values were reported in the ART calculation in CNS Amendment 120 [13].

Per Reference [8] and in accordance with Appendix A of Reference [1], the CNS representative weld and plate surveillance materials data were reviewed from the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) [14]. The representative heat of the plate material (C2307-2) in the ISP is the same as the lower intermediate shell plate material in the vessel beltline region of CNS. For plate heat C2307-2, since the scatter in the fitted results is less than 1-sigma (17°F), the margin term (σ_Δ = 17°F) is cut in half for the plate material when calculating the ART. The representative heat of the weld material (20291) in the ISP is not the same as the limiting weld material in the vessel beltline region of CNS. Therefore, CFs from the tables in RG1.99 were used in the determination of the ART values for all CNS beltline materials except for plate heat C2307-2.

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

- ANSYS, Revision 5.3 [15] for the feedwater (FW) nozzle (non-beltline) pressure and thermal down shock stresses.
- Mechanical and PrepPost, Release 11.0 (Service Pack 1) [16] for the development of the generic WLI nozzle stress intensity factors in [2].
- Mechanical APDL and PrepPost, Release 12.1 [17] for the FW nozzle (non-beltline) thermal ramp stresses and the core differential pressure (DP) nozzle (bottom head) pressure stress distribution.

ANSYS finite element analyses were used to develop the stress distributions through the FW, WLI, and core DP nozzles, and these stress distributions were used in the determination of the stress intensity factors for these nozzles [2, 18, 19, 20]. 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 CNS FW nozzle analysis was performed to determine through-wall pressure stress distributions and thermal stress distributions due to bounding thermal transients [18, 19]. Detailed information regarding the analysis can be found in References [18] and [19]. The following inputs were used as input to the finite element analysis:

- With respect to operating conditions, stress distributions were developed for two bounding thermal transients. A thermal shock, which represents the maximum thermal shock for the FW nozzle during normal and upset operating conditions [18], and a thermal ramp were analyzed [19]. Potential leakage past the primary and secondary thermal sleeves is considered in the heat transfer calculations. The thermal down shock of 450°F, which is associated with the turbine roll transient during startup, produces the highest tensile stresses at the 1/4T location. Because operation is along the saturation curve, these stresses are scaled to reflect the worst-case step change due to the available temperature difference. It is recognized that at low temperatures, the available temperature difference is insignificant and could potentially result in a near zero stress distribution. Therefore, a minimum stress distribution is calculated based on the thermal ramp of 100°F/hour, which is associated with the shutdown transient. Therefore, the combination of the thermal down shock and thermal ramp stresses represent the bounding stresses in the FW nozzle associated with 100°F/hour heatup/cooldown limits associated with the P-T curves for the upper vessel FW nozzle region.

- Heat transfer coefficients were given in the CNS FW nozzle design basis stress report and are a function of FW temperature and flow rate. Bounding, or larger, convection coefficients were used in the present P-T curve analysis [18, 19]. Therefore, the heat transfer coefficients used in the analysis bound the actual operating conditions in the FW nozzle at CNS.

- A two-dimensional finite element model of the FW nozzle was constructed (Figure 4). The pressure stresses are multiplied by a factor of 2.5 to account for the 3-D effects [18]. Material properties were taken at 350°F, which is approximately the average temperature for the shutdown transient, from the 1989 ASME Code [23]. The use of temperature independent material properties is consistent with original design basis documents. Use of temperature dependent material properties is expected to have minimal impact on the results of the analysis.

The plant-specific CNS core DP nozzle analysis was performed to determine a through-wall pressure stress distribution [20]. Detailed information regarding the analysis can be found in Reference [20]. The following inputs were used as input to the finite element analysis:

- No thermal transients were analyzed as part of the plant-specific core DP nozzle evaluation. Thermal stresses were addressed generically as specified in [1] with the use of a stress concentration factor of 3.0 to account for the discontinuity in the bottom head.
- A two-dimensional finite element model of the core DP nozzle was constructed (Figure 5). Material properties were taken at 325°F from the vessel stress report [20]. The use of temperature independent material properties is consistent with original design basis documents. Use of temperature dependent material properties is expected to have minimal impact on the results of the analysis.

6.0 References

1. BWROG-TP-11-022-A, Revision 1, Pressure Temperature Limits Report Methodology for Boiling Water Reactors, June 2013.
2. BWROG-TP-11-023-A, Revision 0, Linear Elastic Fracture Mechanics Evaluation of General Electric Boiling Water Reactor Water Level Instrument Nozzles for Pressure-Temperature Curve Evaluations, May 2013.
3. 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, ML13277A557).
4. U.S. NRC Letter to BWROG dated March 14, 2013, "Final Safety Evaluation for Boiling Water Reactor Owners" Group Topical Report BWROG-TP-11-023, Revision 0, November 2011, 'Linear Elastic Fracture Mechanics Evaluation of General Electric Boiling Water reactor Water Level Instrument Nozzles for Pressure-Temperature Curve Evaluations'" (TAC NO. ME7650, ML13183A017)
5. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence", March 2001.
6. Cooper Nuclear Station Calculation NEDC 07-032, Revision 3, "CNS Review of TransWare Calculations NPP-FLU-003-R-002, Revision 0, NPP-FLU-003-R-004, and NPP-FLU-003-R-005, Reactor Pressure Vessel Fluence Evaluation", April 2013, that incorporated TransWare Enterprises Report No. NPP-FLU-003-R-005, Revision 0, "Non-Proprietary Version of Cooper Nuclear Station Reactor Pressure Vessel Fluence Evaluation," January 2011.
7. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials", May 1988.

8. Cooper Nuclear Station Calculation NEDC 07-045, Revision 3, September 2016, "Review of SIA Calculation 1100445.301, Proprietary and Non-Proprietary Versions, ΔRT_{NDT} and ART Evaluation," dated July 2010.
9. Cooper Nuclear Station Calculation, NEDC 07-048, Revision 7, September 2016, "Review of SIA Calculation 1400473.302 Cooper Updated P-T Curve Calculation for 54 EFPY", dated December 2015.
10. U.S. Code of Federal Regulations, Title 10, Part 50, Section 59, "Changes, tests and experiments," August 28, 2007.
11. Cooper Nuclear Station Amendment 245 as approved by the NRC on February 22, 2013. (ML13032A526)
12. Cooper Nuclear Station Calculation NEDC 16-024, Revision 0, September 2016, "Review of SIA Calculation 1100473.301 Cooper Vacuum Assessment", Revision 0 dated December 2015.
13. Cooper Nuclear Station Amendment 120 as approved by the NRC on April 26, 1988. (ML021360424)
14. 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. SI File No. BWRVIP-135P. **EPRI PROPRIETARY INFORMATION**.
15. ANSYS, Revision 5.3, ANSYS Inc., October 1996.
16. ANSYS Mechanical and PrepPost, Release 11.0 (w/ Service Pack 1), ANSYS, Inc., August 2007.
17. ANSYS Mechanical APDL and PrepPost, Release 12.1 x64, ANSYS, Inc., November 2009.
18. Cooper Nuclear Station Calculation No. NEDC99-020, "Review of Structural Integrity Report SIR-99-069 and Calculations No. NPPD-13Q-301, NPPD-13Q-302, NPPD-13-Q-

303," specifically Structural Integrity Associates Calculation No. NPPD-13Q-302, Revision 1, "Feedwater Nozzle Stress Analysis," June 1999.

19. Cooper Nuclear Station Calculation No. NEDC99-020, Structural Integrity Associates Calculation No. 1100445.302, Revision 0, "Finite Element Stress Analysis of Cooper RPV Feedwater Nozzle," June 2011.

20. Cooper Nuclear Station Calculation, NEDC 16-025, "Review of SIA Calculation 1100445.304 Core Differential Pressure Nozzle Finite Element Model and Stress Analysis" dated July 2011.

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 III, Division 1, Appendices, 1989 Edition.

24. U.S. Code of Federal Regulations, Title 10, Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," January 31, 2008.

25. Letter NLS2002104 dated December 31, 2002, "License Amendment Request to Adopt an Integrated Reactor Vessel Material Surveillance Program, Cooper Nuclear Station, NRC Docket No. 50-298, DPR-46", from M.T. Coyle (NPPD) to U.S. Nuclear Regulatory Commission, ADAMS Accession No. ML030080070, SI File No. 1400473.202.

26. Cooper Nuclear Station Amendment 201 as approved by the NRC on October 23, 2003. (ML033090607)

27. 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.**

28. Cooper Nuclear Station Amendment 256, Cooper Nuclear Station as approved by the NRC on July 25,2016 (ML16158A022)

29. U. S. Nuclear Regulatory Commission, Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits", January 31, 1996.

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

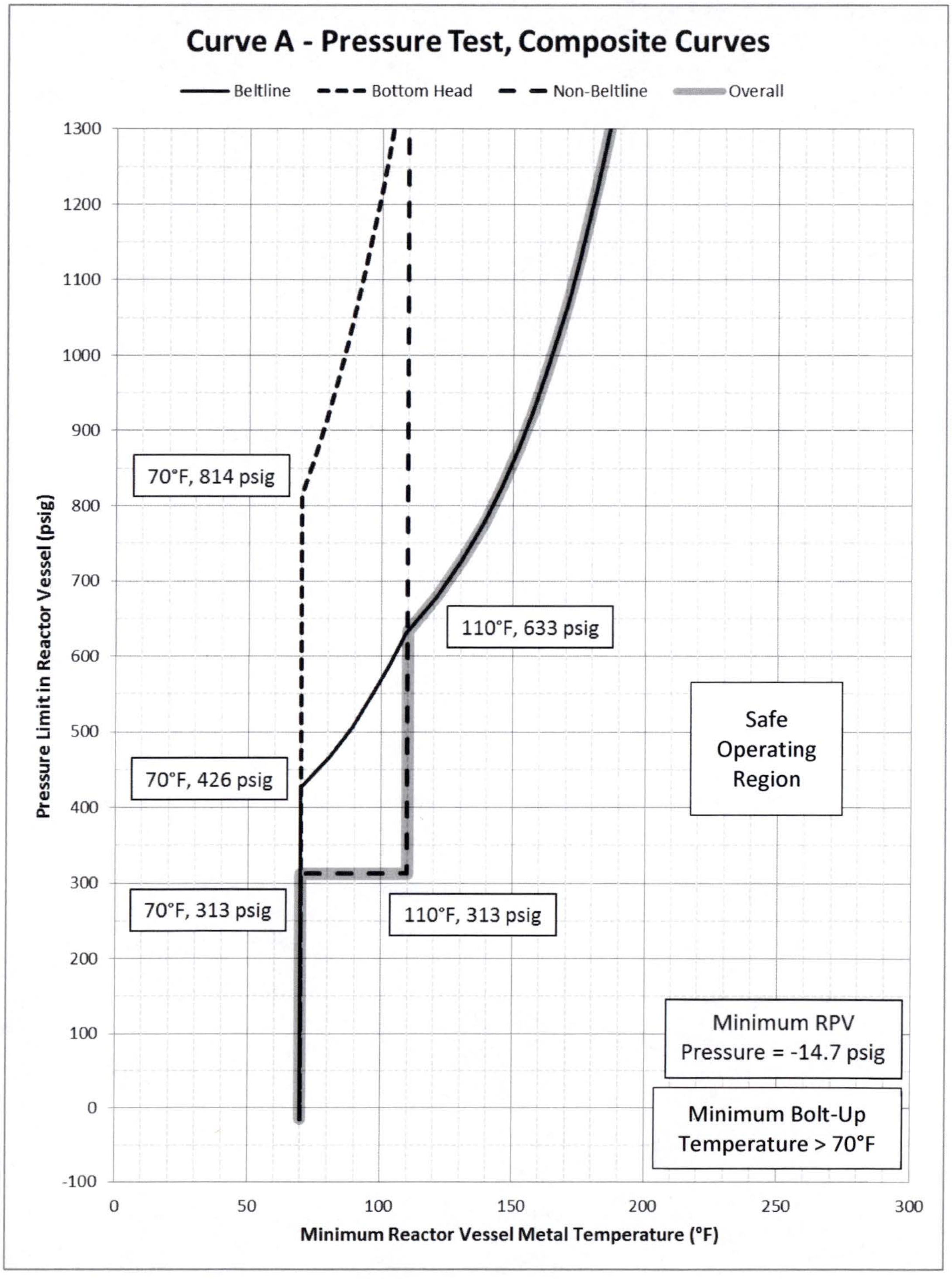

Figure 2: CNS P-T Curve B (Normal Operation – Core Not Critical) for 54 EFPY

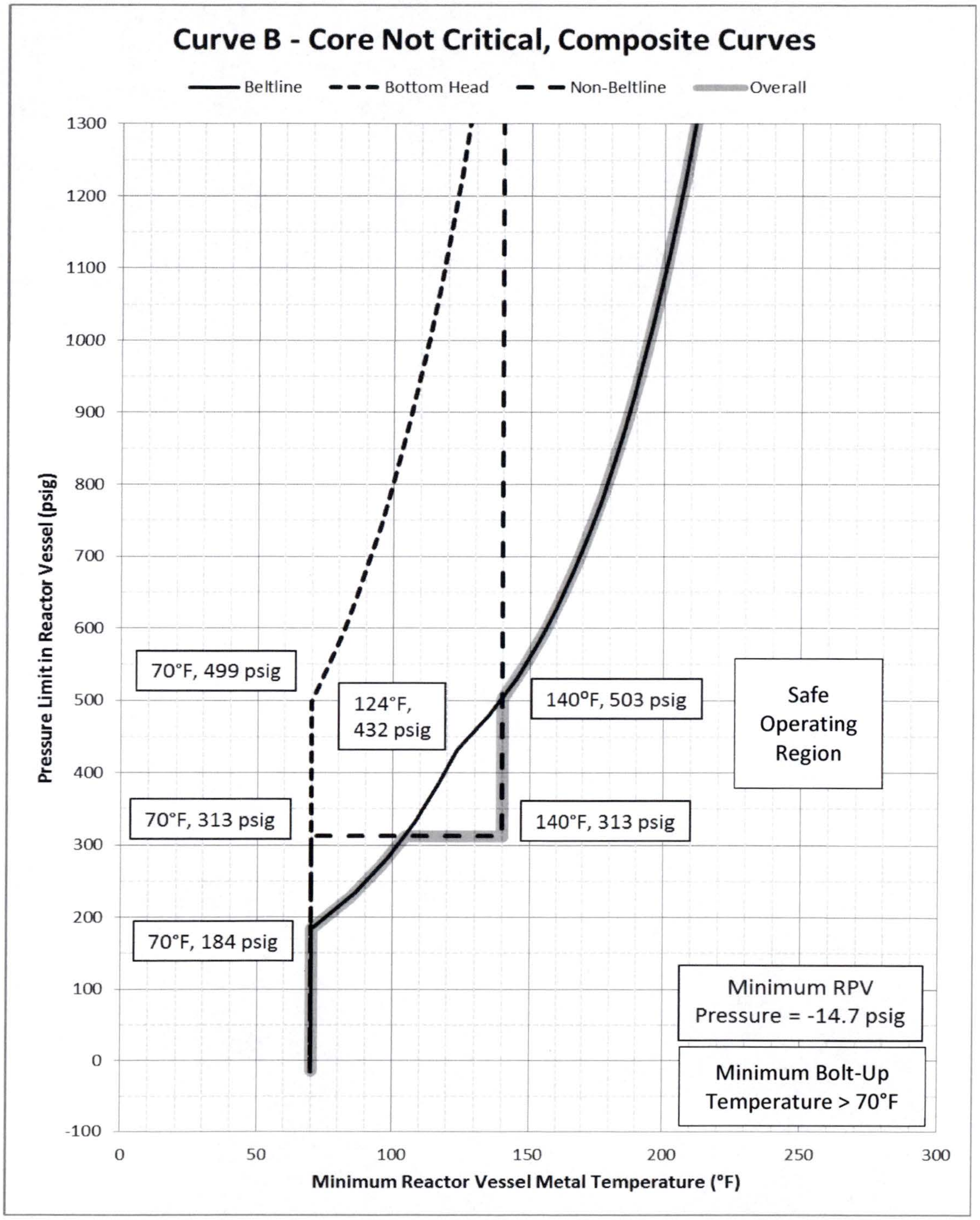

Figure 3: CNS P-T Curve C (Normal Operation – Core Critical) for 54 EFPY

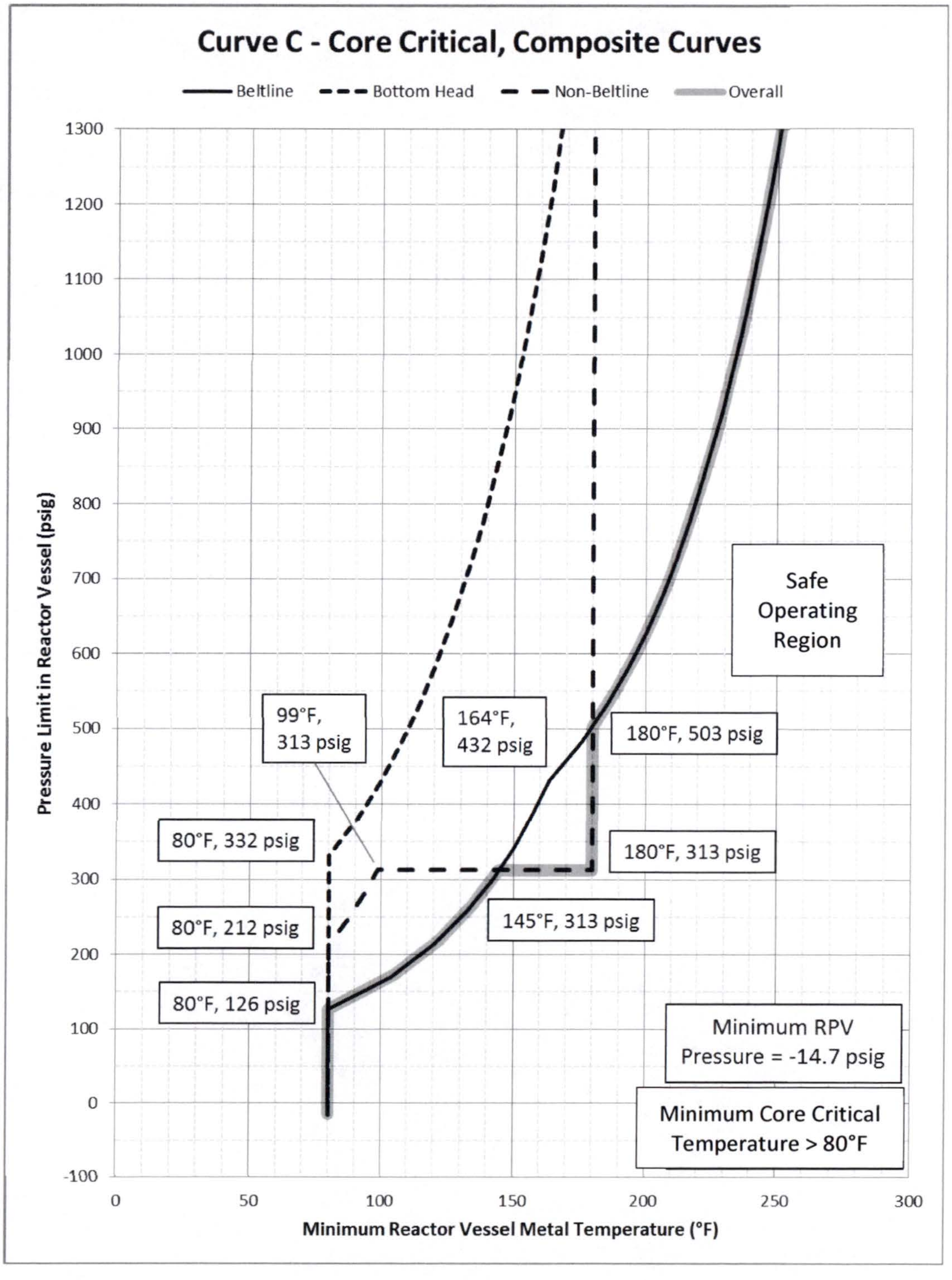

Figure 4: Cooper Feedwater Nozzle Finite Element Model [19]

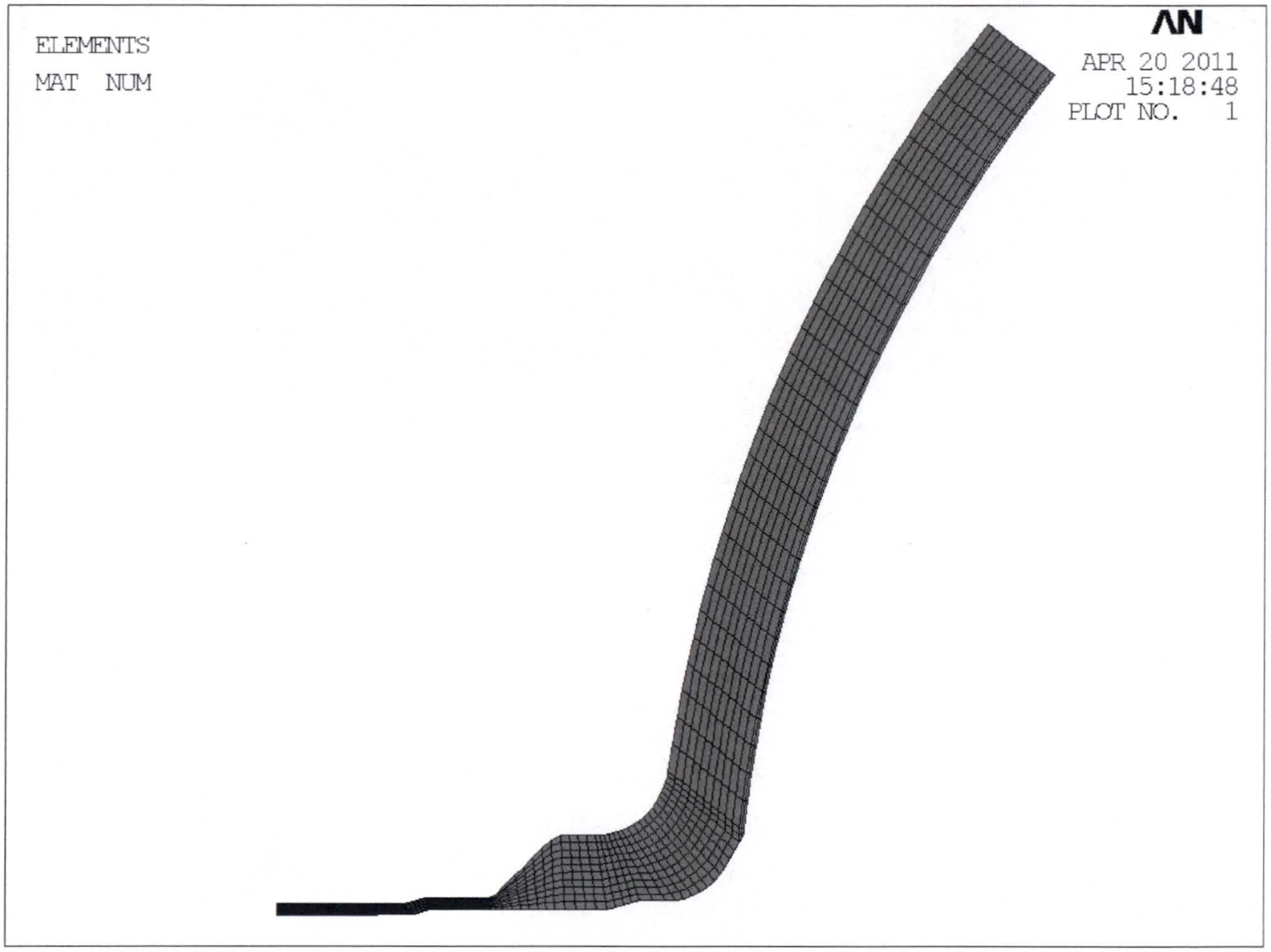

Figure 5: Cooper Core Differential Pressure Nozzle Finite Element Model [20]

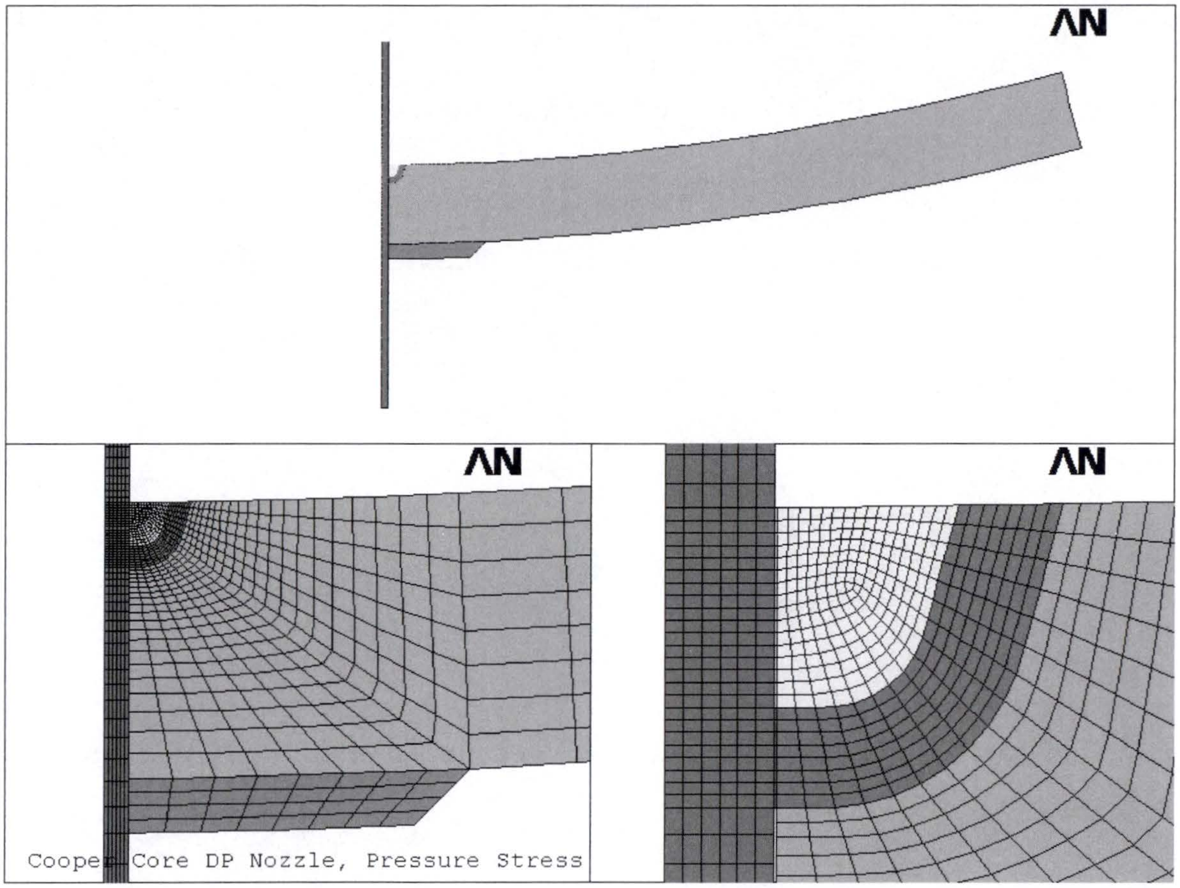

Table 1: CNS Pressure Test (Curve A) P-T Curves for 54 EFPY

Beltline Region

Curve A - Pressure Test	
P-T Curve Temperature	**P-T Curve Pressure**
°F	*psi*
70.0	0.0
70.0	426.0
80.6	466.6
89.3	507.2
96.8	547.7
103.2	588.3
109.0	628.9
120.9	678.1
130.6	727.2
138.7	776.4
145.6	825.5
151.7	874.7
157.2	923.9
162.1	973.0
166.5	1022.2
170.6	1071.4
174.4	1120.5
178.0	1169.7
181.2	1218.9
184.3	1268.0
187.2	1317.2

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

Non-Beltline Region

Curve A - Pressure Test	
P-T Curve Temperature	**P-T Curve Pressure**
°F	*psi*
70.0	0.0
70.0	312.6
110.0	312.6
110.0	1563.0

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

Bottom Head Region

Curve A - Pressure Test	
P-T Curve Temperature	**P-T Curve Pressure**
°F	*psi*
70.0	0.0
70.0	814.0
74.8	864.0
79.2	913.9
83.3	963.8
87.0	1013.8
90.5	1063.7
93.8	1113.6
96.8	1163.5
99.7	1213.5
102.4	1263.4
105.0	1313.3

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

Beltline Region

Curve B - Core Not Critical	
P-T Curve Temperature	**P-T Curve Pressure**
°F	*psi*
70.0	0.0
70.0	184.3
86.2	233.8
98.5	283.3
108.3	332.8
116.5	382.3
123.6	431.7
135.5	480.9
145.1	530.1
153.2	579.3
160.1	628.5
166.2	677.7
171.6	726.8
176.5	776.0
181.0	825.2
185.1	874.4
188.9	923.6
192.4	972.8
195.7	1022.0
198.8	1071.1
201.7	1120.3
204.4	1169.5
207.0	1218.7
209.5	1267.9
211.9	1317.1

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

Non-Beltline Region

Curve B - Core Not Critical	
P-T Curve Temperature	**P-T Curve Pressure**
°F	*psi*
70.0	0.0
70.0	312.6
140.0	312.6
140.0	1563.0

Table 2: **CNS Core Not Critical (Curve B) P-T Curves for 54 EFPY (continued)**

Bottom Head Region

Curve B - Core Not Critical	
P-T Curve Temperature	**P-T Curve Pressure**
°F	*psi*
70.0	0.0
70.0	498.6
76.2	547.0
81.6	595.4
86.6	643.7
91.1	692.1
95.2	740.5
99.0	788.9
102.5	837.3
105.8	885.7
108.9	934.0
111.8	982.4
114.6	1030.8
117.2	1079.2
119.7	1127.6
122.1	1175.9
124.3	1224.3
126.5	1272.7
128.6	1321.1

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

Curve C - Core Critical	
P-T Curve Temperature	**P-T Curve Pressure**
°F	*psi*
80.0	0.0
80.0	126.2
104.0	169.8
120.2	213.5
132.4	257.1
142.2	300.8
150.4	344.4
157.4	388.1
163.6	431.7
175.5	480.9
185.1	530.1
193.2	579.3
200.1	628.5
206.2	677.7
211.6	726.8
216.5	776.0
221.0	825.2
225.1	874.4
228.9	923.6
232.4	972.8
235.7	1022.0
238.8	1071.1
241.7	1120.3
244.4	1169.5
247.0	1218.7
249.5	1267.9
251.9	1317.1

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

Non-Beltline Region

Curve C - Core Critical	
P-T Curve Temperature	**P-T Curve Pressure**
°F	*psi*
80.0	0.0
80.0	211.8
87.3	245.4
93.5	279.0
98.8	312.6
180.0	312.6
180.0	1563.0

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

Bottom Head Region

Curve C - Core Critical	
P-T Curve Temperature	**P-T Curve Pressure**
°F	*psi*
80.0	0.0
80.0	331.9
90.9	381.1
99.8	430.4
107.4	479.6
113.9	528.9
119.7	578.1
124.9	627.4
129.7	676.6
134.0	725.8
137.9	775.1
141.6	824.3
145.0	873.6
148.2	922.8
151.2	972.1
154.1	1021.3
156.8	1070.6
159.3	1119.8
161.7	1169.0
164.1	1218.3
166.3	1267.5
168.4	1316.8

Table 4: CNS ART Calculations for 54 EFPY

	Beltline ID	Code No.	Heat No.	Flux Type	Initial RT_{NDT} (°F)	Cu (wt%)	Ni (wt%)	CF (°F)	ΔRT_{NDT} (°F)	Margin Terms σ_Δ (°F)	Margin Terms σ_i (°F)	Total Margin (°F)	ART (°F)
Plates	Lower Shell Plate	G-2803-1	C2274-1	-	14.0	0.20	0.68	153.0	69.3	17.0	0.0	34.0	117.3
	Lower Shell Plate	G-2803-2	C2307-1	-	0.0	0.21	0.73	162.8	73.8	17.0	0.0	34.0	107.8
	Lower Shell Plate	G-2803-3	C2274-2	-	-8.0	0.20	0.68	153.0	69.3	17.0	0.0	34.0	95.3
	Lower Int. Shell Plate	G-2802-1	C2331-2	-	10.0	0.16	0.62	{{[redacted]}}	77.7	8.5	0.0	17.0	104.7
	Lower Int. Shell Plate	G-2802-2	C2307-2	-	-20.0	0.21	0.76	{{[redacted]}}	134.2	8.5	0.0	17.0	131.2
	Lower Int. Shell Plate	G-2801-7	C2407-1	-	-10.0	0.13	0.65	92.3	47.9	17.0	0.0	34.0	71.9
Welds	Lower Shell Axial Welds	2-233A	12420	LINDE 1092	-50.0	0.270	1.035	254.4	114.4	28.0	0.0	56.0	120.4
	Lower Shell Axial Welds	2-233B	12420	LINDE 1092	-50.0	0.270	1.035	254.4	114.4	28.0	0.0	56.0	120.4
	Lower Shell Axial Welds	2-233C	12420	LINDE 1092	-50.0	0.270	1.035	254.4	114.4	28.0	0.0	56.0	120.4
	Lower Int. Shell Axial Welds	1-233A	27204/12008	LINDE 1092	-50.0	0.219	0.996	231.1	92.2	28.0	0.0	56.0	98.2
	Lower Int. Shell Axial Welds	1-233B	27204/12008	LINDE 1092	-50.0	0.219	0.996	231.1	92.2	28.0	0.0	56.0	98.2
	Lower Int. Shell Axial Welds	1-233C	27204/12008	LINDE 1092	-50.0	0.219	0.996	231.1	92.2	28.0	0.0	56.0	98.2
	Lower/Lower Int. Shell Circ Weld	1-240	21935	LINDE 1092	-50.0	0.183	0.704	172.2	80.2	28.0	0.0	56.0	86.2
Nozzles	Nozzle N-16A	G-2822	EV-26067		-10.0	0.13	0.65	92.3	23.7	8.3	0.0	16.5	37.4
	Nozzle N-16B	G-2822	EV-26067		10.0	0.16	0.62	118.5	30.4	10.6	0.0	21.2	70.8

Fluence Data

	Beltline ID	Code No.	Heat No.	Wall Thickness (in.) Full	Wall Thickness (in.) 1/4t	Fluence at ID (n/cm²)	Attenuation $e^{-0.24x}$	Fluence at 1/4t (n/cm²)	Fluence Factor, FF $f^{(0.28 - 0.10 \log f)}$
Plates	Lower Shell Plate	G-2803-1	C2274-1	6.375	1.59	1.75E+18	0.68	1.19E+18	0.453
	Lower Shell Plate	G-2803-2	C2307-1	6.375	1.59	1.75E+18	0.68	1.19E+18	0.453
	Lower Shell Plate	G-2803-3	C2274-2	6.375	1.59	1.75E+18	0.68	1.19E+18	0.453
	Lower Int. Shell Plate	G-2802-1	C2331-2	5.375	1.34	2.23E+18	0.72	1.62E+18	0.520
	Lower Int. Shell Plate	G-2802-2	C2307-2	5.375	1.34	2.23E+18	0.72	1.62E+18	0.520
	Lower Int. Shell Plate	G-2801-7	C2407-1	5.375	1.34	2.23E+18	0.72	1.62E+18	0.520
Welds	Lower Shell Axial Welds	2-233A	12420	6.375	1.59	1.72E+18	0.68	1.17E+18	0.450
	Lower Shell Axial Welds	2-233B	12420	6.375	1.59	1.72E+18	0.68	1.17E+18	0.450
	Lower Shell Axial Welds	2-233C	12420	6.375	1.59	1.72E+18	0.68	1.17E+18	0.450
	Lower Int. Shell Axial Welds	1-233A	27204/12008	5.375	1.34	1.26E+18	0.72	9.13E+17	0.399
	Lower Int. Shell Axial Welds	1-233B	27204/12008	5.375	1.34	1.26E+18	0.72	9.13E+17	0.399
	Lower Int. Shell Axial Welds	1-233C	27204/12008	5.375	1.34	1.26E+18	0.72	9.13E+17	0.399
	Lower/Lower Int. Shell Circ Weld	1-240	21935	5.375	1.34	1.75E+18	0.72	1.27E+18	0.466
Nozzles	Nozzle N-16A	G-2822	EV-26067	5.375	1.34	5.44E+17	0.72	3.94E+17	0.257
	Nozzle N-16B	G-2822	EV-26067	5.375	1.34	5.44E+17	0.72	3.94E+17	0.257

EPRI Proprietary Information

(such information is marked with double braces "{{xxx}}" and a bar in the right-hand margin)

Appendix A

COOPER REACTOR VESSEL MATERIALS SURVEILLANCE PROGRAM

In accordance with 10 CFR 50, Appendix H, Reactor Vessel Material Surveillance Program Requirements [24], two surveillance capsules were removed from the CNS reactor vessel in 1985 at 6.8 EFPY and 1991 at 11.2 EFPY [25, Attachment 3]. 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.

CNS is currently committed to use the BWRVIP ISP, and has made a licensing commitment to use the ISP for CNS 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. Nebraska Public Power District committed to use the ISP in place of its existing surveillance programs in the amendments issued by the NRC regarding the implementation of the Boiling Water Reactor Vessel and Internals Project Reactor Pressure Vessel Integrated Surveillance Program, dated October 31, 2003 [26]. Under the ISP, a capsule was scheduled for removal in 2003 but removal has been deferred to approximately 2017 at 32 EFPY [27]. CNS recently transitioned to 24 month refueling cycles during "even" years so the next capsule removal will occur in 2018 to align with a plant refueling outage as allowed by the ISP [27]. Additionally, CNS served as a host plant for three of the nine surveillance capsules irradiated as part of the Supplemental Surveillance Program; the SSP-A, SSP-B, and SSP-C capsules were removed from CNS and tested in 2003 [27] 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. CNS continues to be a host plant under the ISP. One additional standby Cooper capsule is currently scheduled to be removed and tested under the ISP during the license renewal period in approximately 2029 at 40 EFPY [27].