Enclosure 1

Cooper Nuclear Station Pressure and Temperature Limits Report (PTLR) for 32 Effective Full-Power Years (EFPY) (Non-Proprietary version of ER 15-015)

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

Engineering Report No. 15-019 Rev 1
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Engineering Report Cover Sheet

Engineering Report Title:
Cooper Nuclear Station Pressure and Temperature Limits Report (PTLR)
for 32 Effective Full-Power Years (EFPY)
(Non-Proprietary version of ER 15-015)

Engineering Report Type: (3)

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Rev 1: Added Attachment 1 for Cooper specific pages from BWRVIP-135, Revision 3. Added reference to Cooper Nuclear Station in Title and updated References to include GL 96-03.

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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].

It should also be noted that the P-T curves referenced in this PTLR have previously been approved by the NRC in Amendment 245 [24]. No changes are being made under this PTLR to the current P-T curves that were approved by the NRC [24] and currently in effect at CNS.

2.0 Applicability

This report is applicable to the CNS RPV for up to 32 Effective Full-Power Years (EFPY).

The following CNS Technical Specification (TS) is 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 consistent with Reference [1] and Reference [2], which have been approved by the NRC in References [25] and [26], respectively.

2. The neutron fluence is calculated in accordance with NRC Regulatory Guide 1.190 (RG 1.190) [3], using the RAMA computer code, 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 [5], as documented in Reference [6].

4. The pressure and temperature limits were calculated consistent with Reference [1], "Pressure – Temperature Limits Report Methodology for Boiling Water Reactors,", as documented in NPPD calculation NEDC 07-048, Reference [7].

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

    - Initial issue of PTLR revised to include BWRVIP-135 Cooper specific information.

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 [17], 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 32 EFPY for Cooper Nuclear Station, as documented in Reference [7] and approved by the NRC in CNS Amendment 245 [24]. The CNS P-T curves for 32 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 32 EFPY (Reference [6]). 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 at or near isothermal conditions (Figure 1: Curve A): **≤ 25°F/hour**[1] [7].
- 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] [7].
- 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** [7].

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[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.

To address the NRC condition regarding lowest service temperature in Reference [1] the minimum temperature is set to 70°F, which is equal to the $RT_{NDT,max}$ + 60°F, for all curves. [24]

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 CNS vessel plate, weld, and forging materials [6]; 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 1.41×10^{18} n/cm^2 at 32 EFPY used in the P-T curve evaluation were obtained from Reference [4] and are calculated in accordance with RG 1.190 [3]. 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 32 EFPY for the limiting lower intermediate shell plate is 1.02×10^{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 [7]. 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 2.94 x 10^{17} n/cm^2 at 32 EFPY used in the P-T curve evaluation of the WLI nozzle was obtained from Reference [4] and is calculated in accordance with RG 1.190 [3]. This fluence value applies to the limiting WLI nozzle (Heat No. EV-26067). The fluence value for the WLI nozzle is based upon an attenuation factor of 0.72 for a postulated 1/4T flaw. As a result, the 1/4T fluence for 32 EFPY for the limiting WLI nozzle is 2.13 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 32 EFPY [6].

Per Reference [6] 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) [19]. 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 [8] for the feedwater (FW) nozzle (non-beltline) pressure and thermal down shock stresses.
- Mechanical and PrepPost, Release 11.0 (Service Pack 1) [9] for the development of the generic WLI nozzle stress intensity factors in [2].
- Mechanical APDL and PrepPost, Release 12.1 [10] 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, 13, 14, 16]. 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 [11] Quality Assurance Program for nuclear quality-related work. Benchmarking consistent with NRC GL 83-11, Supplement 1 [12] 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 [13, 14]. Detailed information regarding the analysis can be found in References [13] and [14]. 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 [13], and a thermal ramp were analyzed [14]. 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 [13, 14]. 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 [13]. Material properties were taken at 350°F, which is approximately the average temperature for the shutdown transient, from the 1989 ASME Code [15]. 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 [16]. Detailed information regarding the analysis can be found in Reference [16]. 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 [16]. The use of temperature independent material properties is consistent with original design basis documents.
- Intial RT_{NDT} values were reported in the ART calculation in amendment 120 [22].

6.0 References

1. BWROG-TP-11-022-A, Revision 1, Pressure Temperature Limits Report Methodology for Boiling Water Reactors, August 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. Nuclear Regulatory Commission, Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence", March 2001.
4. Cooper Nuclear Station Calculation NEDC 07-032, Revision 3, "CNS Review of Trans Ware 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, SI File No. 1100445.201.
5. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel materials", May 1988.
6. Cooper Nuclear Station Calculation NEDC07-045, Revision 2, "Review of SIA Calculation COOP-27Q-301, ΔRT_{NDT} and ART Evaluation", April 2013 that incorporated Structural Integrity Associates Calculation No. 1100445.301, Revision 1, "ΔRT_{NDT} and ART Evaluation", July 2010.
7. Cooper Nuclear Station Calculation, NEDC07-048, Revision 6, "Revised Pressure Temperature Curves", April 2013 that incorporated Structural Integrity Associates Calculation No. 1100445.303, Revision 0, "Revised P-T Curve Calculation", August 2011.
8. ANSYS, Revision 5.3, ANSYS Inc., October 1996.
9. ANSYS Mechanical and PrepPost, Release 11.0 (w/ Service Pack 1), ANSYS, Inc., August 2007.

10. ANSYS Mechanical APDL and PrepPost, Release 12.1 x64, ANSYS, Inc., November 2009.

11. U. S. Code of Federal Regulations, Title 10, Energy, Part 50, Appendix B, "Quality Assurance for Nuclear Power Plants and Fuel Reprocessing Plants".

12. U. S. Nuclear Regulatory Commission, Generic Letter 83-11, Supplement 1, "License Qualification for Performing Safety Analyses", June 24, 1999.

13. 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.

14. 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.

15. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Appendices, 1989 Edition.

16. Cooper Nuclear Station Calculation, NEDC07-048, Revision 6, "Revised Pressure Temperature Curves", April 2013 that incorporated Structural Integrity Associates Calculation No. 1100445.304, Revision 0, "Core Differential Pressure Nozzle Finite Element Model and Stress Analysis," August 2011.

17. U. S. Code of Federal Regulations, Title 10, Part 50, Section 59, "Changes, tests and experiments," Aug. 28. 2007.

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

19. 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.** (See Supplement 1)

20. 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.

21. 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.**

22. Cooper Nuclear Station Amendment 120 as approved by the NRC on April 26, 1988. (ML021360424)

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

24. Cooper Nuclear Station Amendment 245 as approved by the NRC on February 22, 2013. (ML13032A526).

25. U.S. NRC Letter to BWROG, "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).

26. U.S. NRC Letter to BWROG, "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)

27. 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 32 EFPY [7]

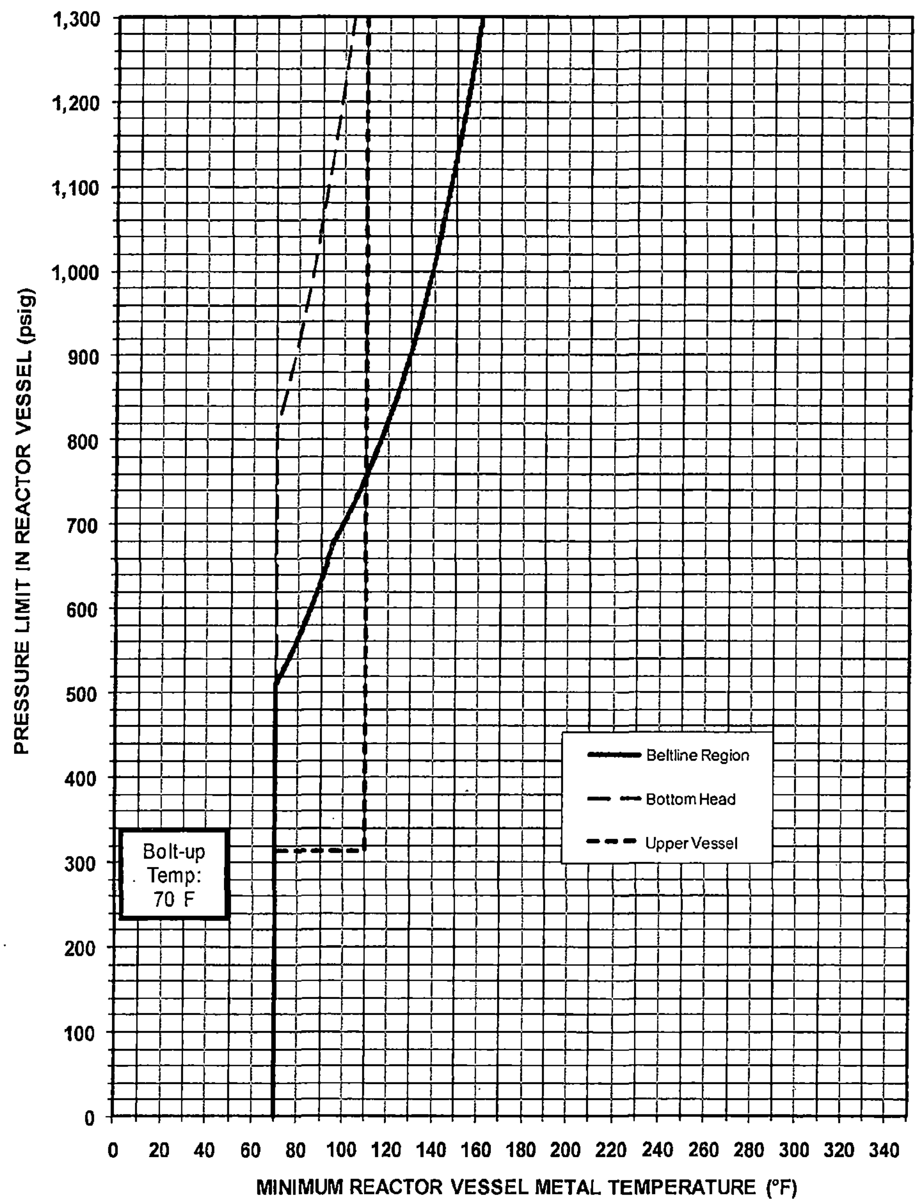

Figure 2: CNS P-T Curve B (Normal Operation – Core Not Critical) for 32 EFPY [7]

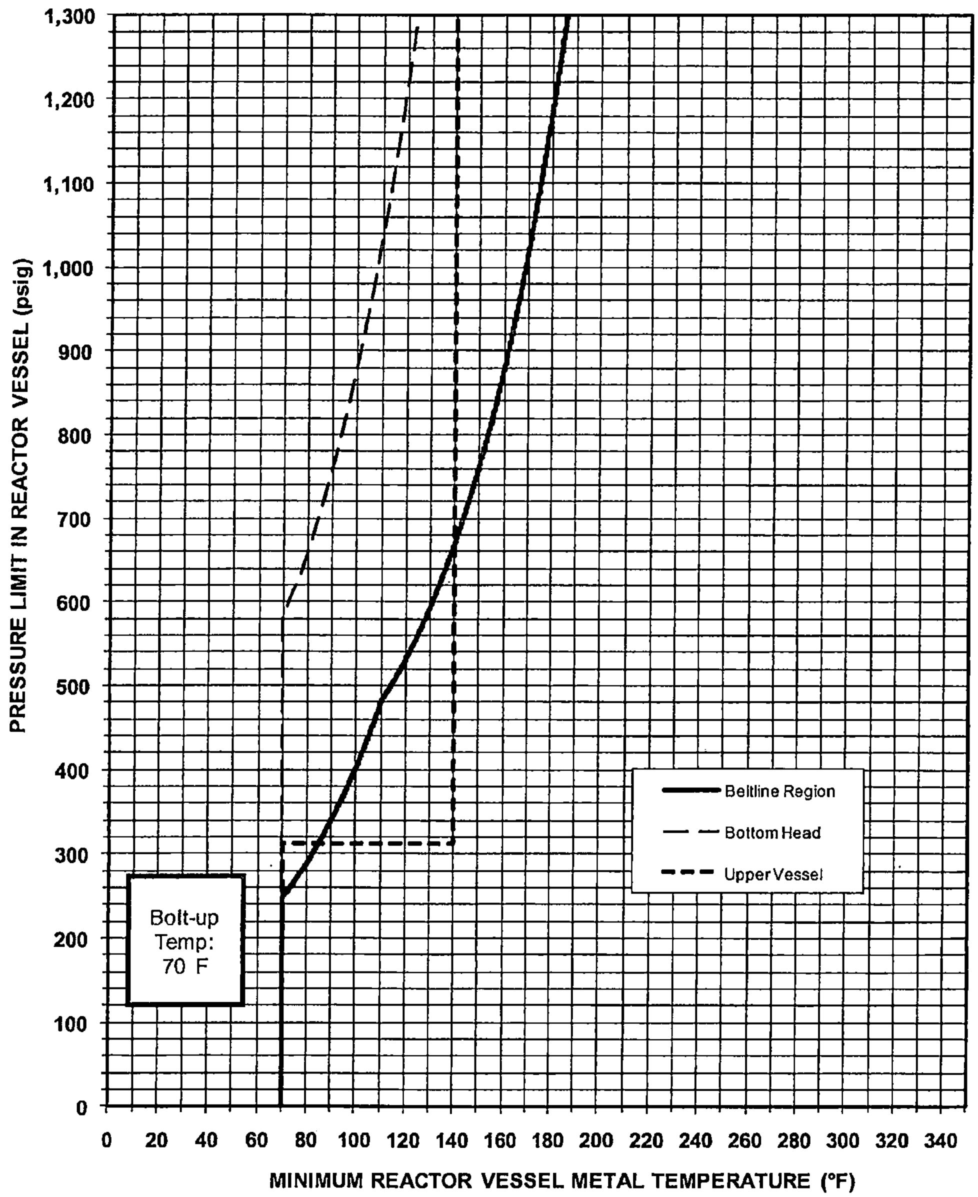

Figure 3: CNS P-T Curve C (Normal Operation – Core Critical) for 32 EFPY [7]

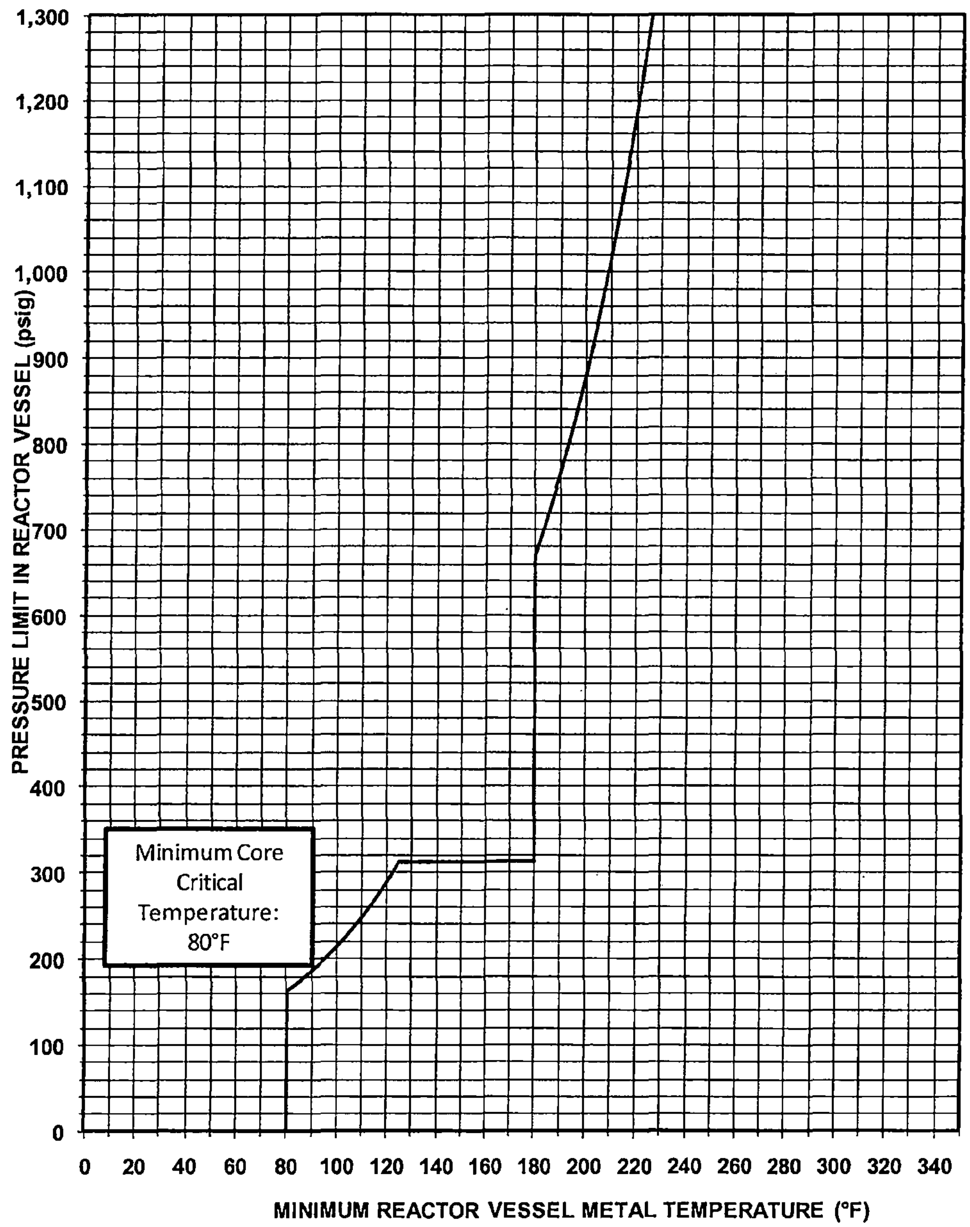

Figure 4: Cooper Feedwater Nozzle Finite Element Model [14]

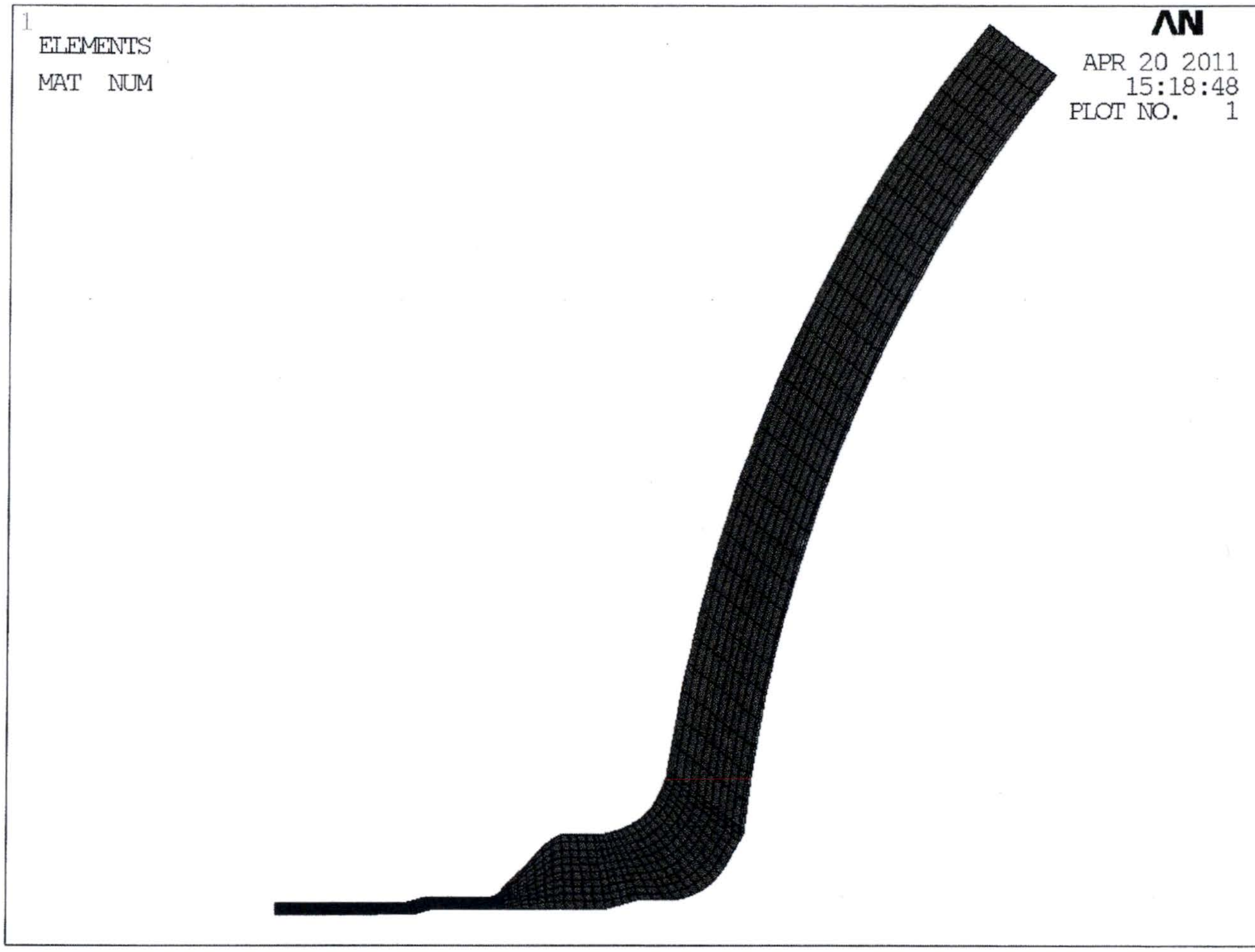

Figure 5: Cooper Core Differential Pressure Nozzle Finite Element Model [16]

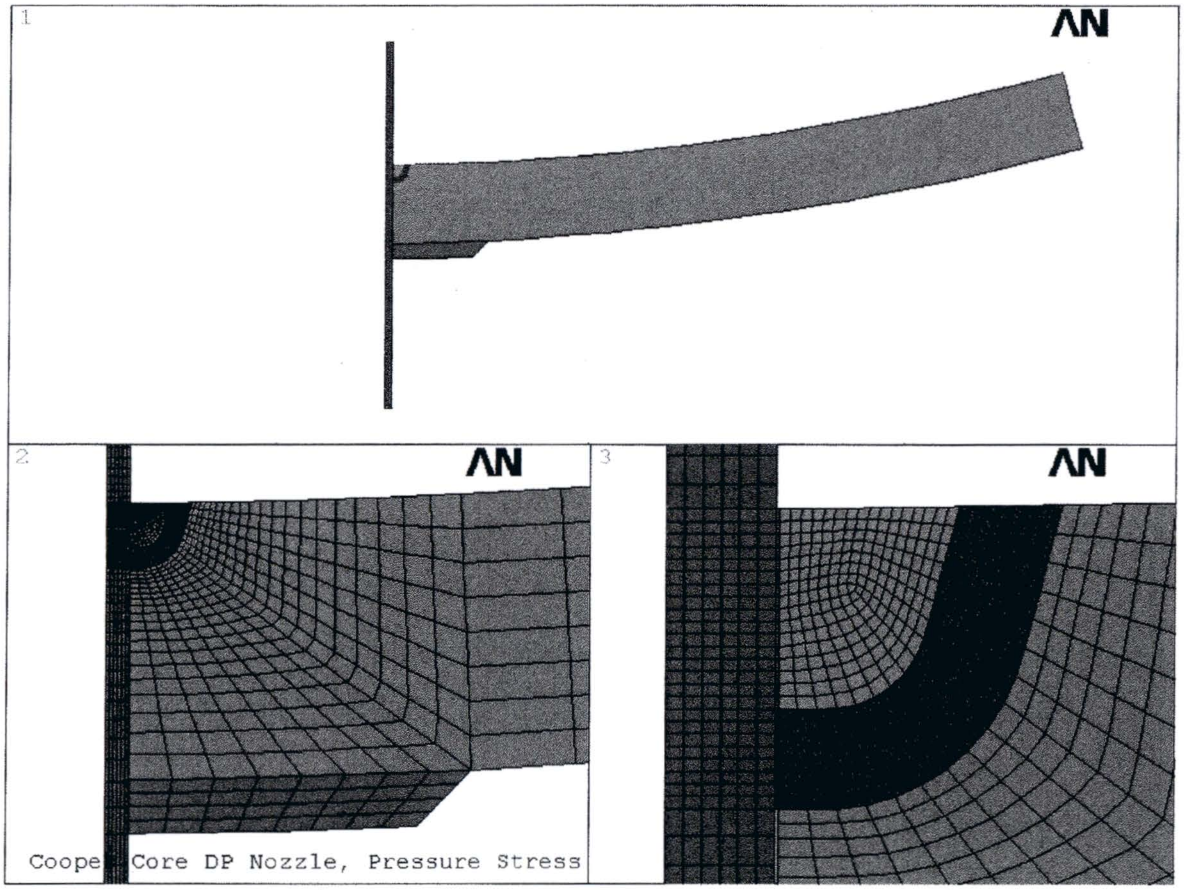

Table 1: CNS Pressure Test (Curve A) P-T Curves for 32 EFPY [7]

Beltline Region

P-T Curve Temperature	P-T Curve Pressure
70.00	0
70.00	50
70.00	100
70.00	150
70.00	200
70.00	250
70.00	300
70.00	312
70.00	313
70.00	350
70.00	400
70.00	450
70.00	500
77.37	550
85.28	600
92.11	650
100.07	700
109.08	750
116.71	800
123.34	850
129.19	900
134.42	950
139.16	1000
143.49	1050
147.47	1100
151.16	1150
154.60	1200
157.82	1250
160.83	1300

Table 1: CNS Pressure Test (Curve A) P-T Curves for 32 EFPY (continued)

Plant =	**CNS**	
Component =	**Bottom Head**	(penetrations portion)
Bottom Head thickness, t =	**6.813**	inches
Bottom Head Radius, R =	**110.5**	inches
ART =	**28.0**	°F ======> **All EFPY**
K_{It} =	**0.00**	(no thermal effects)
Safety Factor =	**1.50**	
Stress Concentration Factor =	**3.00**	(bottom head penetrations)
M_m =	**2.417**	
Temperature Adjustment =	**5.0**	°F (applied after bolt-up, instrument uncertainty)
Height of Water for a Full Vessel =	**831.75**	inches
Pressure Adjustment =	**30.0**	psig (hydrostatic pressure head for a full vessel at 70°F)
Pressure Adjustment =	**25.0**	psig (instrument uncertainty)

Gauge Fluid Temperature (°F)	K_{Ic} (ksi*inch$^{1/2}$)	K_{Im} (ksi*inch$^{1/2}$)	Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
65.0	76.66	51.10	70	0
65.0	76.66	51.10	70	814
67.0	78.43	52.29	72	834
69.0	80.28	53.52	74	855
71.0	82.20	54.80	76	877
73.0	84.20	56.13	78	900
75.0	86.28	57.52	80	923
77.0	88.44	58.96	82	948
79.0	90.70	60.47	84	973
81.0	93.05	62.03	86	1,000
83.0	95.49	63.66	88	1,028
85.0	98.03	65.35	90	1,056
87.0	100.68	67.12	92	1,086
89.0	103.43	68.95	94	1,118
91.0	106.30	70.86	96	1,150
93.0	109.28	72.85	98	1,184
95.0	112.38	74.92	100	1,219
97.0	115.62	77.08	102	1,256
99.0	118.98	79.32	104	1,294

Table 1: CNS Pressure Test (Curve A) P-T Curves for 32 EFPY (continued)

Plant =	**CNS**		
Component =	**Upper Vessel**		
ART =	**20.0**	°F ======>	**All EFPY**
Vessel Radius, R =	**110.375**	inches	
Nozzle comer thickness, t' =	**5.753**	inches, approximate	
K_{It} =	**0.00**	(no thermal effects)	
$K_{Ip\text{-}applied}$ =	**38.90**	ksi*inch$^{1/2}$	
Crack Depth, a =	**1.438**	inches	
Safety Factor =	**1.50**		
Temperature Adjustment =	**5.0**	°F (applied after bolt-up, instrument uncertainty)	
Height of Water for a Full Vessel =	**831.75**	inches	
Pressure Adjustment =	**30.0**	psig (hydrostatic pressure head for a full vessel at 70°F)	
Pressure Adjustment =	**25.0**	psig (instrument uncertainty)	
Reference Pressure =	**1,000**	psig (pressure at which the FEA stress coefficients are valid)	
Unit Pressure =	**1,563**	psig (hydrostatic pressure)	
Flange RT_{NDT} =	**20.0**	°F ======>	**All EFPY**

Gauge Fluid Temperature (°F)	K_{Ic} (ksi*inch$^{1/2}$)	K_{Ip} (ksi*inch$^{1/2}$)	P-T Curve Temperature (°F)	P-T Curve 10CFR50 Adjustments (psig)
65.0	84.20	56.13	70	0
65.0	84.20	56.13	70	313
67.0	86.28	57.52	110	313
69.0	88.44	58.96	110	1461
71.0	90.70	60.47	110	1499
73.0	93.05	62.03	110	1539
75.0	95.49	63.66	110	1581
77.0	98.03	65.35	110	1625

Table 2: CNS Core Not Critical (Curve B) P-T Curves for 32 EFPY [7]

Beltline Region

P-T Curve Temperature	P-T Curve Pressure
70.00	0
70.00	50
70.00	100
70.00	150
70.00	200
70.22	250
81.92	300
84.36	312
84.55	313
91.39	350
99.35	400
106.22	450
114.04	500
123.11	550
130.80	600
137.46	650
143.34	700
148.60	750
153.35	800
157.70	850
161.70	900
165.40	950
168.84	1000
172.07	1050
175.09	1100
177.96	1150
180.65	1200
183.21	1250
185.65	1300

Table 2: **CNS Core Not Critical (Curve B) P-T Curves for 32 EFPY (continued)**

Plant =	**CNS**	
Component =	**Bottom Head**	(penetrations portion)
Bottom Head thickness, t =	**6.813**	inches
Bottom Head Radius, R =	**110.5**	inches
ART =	**28.0**	°F ======> **All EFPY**
K_{It} =	**1.73**	ksi*inch$^{1/2}$
Safety Factor =	**2.00**	
Stress Concentration Factor =	**3.00**	(bottom head penetrations)
M_m =	**2.417**	
Temperature Adjustment =	**5.0**	°F (applied after bolt-up, instrument uncertainty)
Height of Water for a Full Vessel =	**831.75**	inches
Pressure Adjustment =	**30.0**	psig (hydrostatic pressure head for a full vessel at 70°F)
Pressure Adjustment =	**25.0**	psig (instrument uncertainty)
Heat Up and Cool Down Rate =	**100**	°F/Hr

Gauge Fluid Temperature (°F)	K_{Ic} (ksi*inch$^{1/2}$)	K_{Im} (ksi*inch$^{1/2}$)	Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
65.0	76.66	37.46	70	0
65.0	76.66	37.46	70	582
67.0	78.43	38.35	72	597
69.0	80.28	39.27	74	613
71.0	82.20	40.23	76	629
73.0	84.20	41.23	78	646
75.0	86.28	42.27	80	664
77.0	88.44	43.36	82	682
79.0	90.70	44.49	84	701
81.0	93.05	45.66	86	721
83.0	95.49	46.88	88	742
85.0	98.03	48.15	90	764
87.0	100.68	49.47	92	786
89.0	103.43	50.85	94	810
91.0	106.30	52.28	96	834
93.0	109.28	53.78	98	859
95.0	112.38	55.33	100	886
97.0	115.62	56.94	102	913
99.0	118.98	58.63	104	942
101.0	122.48	60.38	106	972
103.0	126.12	62.20	108	1,003
105.0	129.92	64.09	110	1,035
107.0	133.86	66.07	112	1,068
109.0	137.97	68.12	114	1,103
111.0	142.25	70.26	116	1,140
113.0	146.70	72.48	118	1,178
115.0	151.33	74.80	120	1,217
117.0	156.15	77.21	122	1,258

Table 2: **CNS Core Not Critical (Curve B) P-T Curves for 32 EFPY (continued)**

Plant =	**CNS**	
Component =	**Upper Vessel**	
ART =	**20.0**	°F ======> **All EFPY**
Vessel Radius, R =	**110.375**	inches
Nozzle corner thickness, t =	**5.753**	inches, approximate
K_{It} =	**63.45**	ksi*inch$^{1/2}$
$K_{Ip\text{-}applied}$ =	**38.90**	ksi*inch$^{1/2}$
Crack Depth, a =	**1.438**	inches
Safety Factor =	**2.00**	
Temperature Adjustment =	**5.0**	°F (applied after bolt-up, instrument uncertainty)
Height of Water for a Full Vessel =	**831.75**	inches
Pressure Adjustment =	**30.0**	psig (hydrostatic pressure head for a full vessel at 70°F)
Pressure Adjustment =	**25.0**	psig (instrument uncertainty)
Reference Pressure =	**1,000**	psig (pressure at which the FEA stress coefficients are valid)
Unit Pressure =	**1,563**	psig (hydrostatic pressure)
Flange RT_{NDT} =	**20.0**	°F ======> **All EFPY**

Gauge Fluid Temperature (°F)	**K_{Ic} (ksi*inch$^{1/2}$)**	**K_{Ip} (ksi*inch$^{1/2}$)**	**P-T Curve Temperature (°F)**	**P-T Curve Pressure (psig)**
65.0	84.20	10.37	70	0
65.0	84.20	19.46	70	313
67.0	86.28	20.51	140	313
69.0	88.44	19.26	140	440
71.0	90.70	20.06	140	461
73.0	93.05	20.91	140	482
75.0	95.49	21.80	140	505
77.0	98.03	22.73	140	529
79.0	100.68	23.71	140	554
81.0	103.43	24.74	140	581
83.0	106.30	25.82	140	609
85.0	109.28	26.95	140	638
87.0	112.38	28.15	140	668
89.0	115.62	29.40	140	701
91.0	118.98	30.71	140	734
93.0	122.48	32.09	140	770
95.0	126.12	33.54	140	807
97.0	129.92	35.04	140	846
99.0	133.86	36.63	140	887
101.0	137.97	38.30	140	929
103.0	142.25	40.04	140	974
105.0	146.70	41.87	140	1021
107.0	151.33	43.79	140	1071
109.0	156.15	45.80	140	1122
111.0	161.17	47.90	140	1176
113.0	166.39	50.10	140	1233
115.0	171.83	52.39	140	1292

Table 3: CNS Core Critical (Curve C) P-T Curves for 32 EFPY [7]

Plant =	CNS	
Curve A Leak Test Temperature =	148.0	°F
Curve A Pressure =	1,100.0	psig
Unit Pressure =	1,563	psig (hydrostatic pressure)
Flange RT_{NDT} =	20.0	°F

P-T Curve Temperature	P-T Curve Pressure
80.00	0
80.00	50
80.00	100
80.00	150
94.91	200
110.21	250
121.92	300
124.36	312
180.00	313
180.00	350
180.00	400
180.00	450
180.00	500
180.00	550
180.00	600
180.00	650
183.34	700
188.60	750
193.35	800
197.70	850
201.70	900
205.40	950
208.84	1000
212.07	1050
215.09	1100
217.96	1150
220.65	1200
223.21	1250
225.65	1300

Table 4: CNS ART Calculations for 32 EFPY [6]

	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	55.1	17.0	0.0	34.0	103.1
	Lower Shell Plate	G-2803-2	C2307-1	-	0.0	0.21	0.73	162.8	58.6	17.0	0.0	34.0	92.6
	Lower Shell Plate	G-2803-3	C2274-2	-	-8.0	0.20	0.68	153.0	55.1	17.0	0.0	34.0	81.1
	Lower Int. Shell Plate	G-2802-1	C2331-2	-	10.0	[[[redacted]]]	[[[redacted]]]	[[[redacted]]]	63.0	8.5	0.0	17.0	90.0
	Lower Int. Shell Plate	G-2802-2	C2307-2	-	-20.0	[[[redacted]]]	[[[redacted]]]	[[[redacted]]]	108.8	8.5	0.0	17.0	105.8
	Lower Int. Shell Plate	G-2801-7	C2407-1	-	-10.0	0.13	0.65	92.3	38.8	17.0	0.0	34.0	62.8
Welds	Lower Shell Axial Welds	2-233A	12420	LINDE 1092	-50.0	0.270	1.035	254.4	90.8	28.0	0.0	56.0	96.8
	Lower Shell Axial Welds	2-233B	12420	LINDE 1092	-50.0	0.270	1.035	254.4	90.8	28.0	0.0	56.0	96.8
	Lower Shell Axial Welds	2-233C	12420	LINDE 1092	-50.0	0.270	1.035	254.4	90.8	28.0	0.0	56.0	96.8
	Lower Int. Shell Axial Welds	1-233A	27204/12008	LINDE 1092	-50.0	0.219	0.996	231.1	73.7	28.0	0.0	56.0	79.7
	Lower Int. Shell Axial Welds	1-233B	27204/12008	LINDE 1092	-50.0	0.219	0.996	231.1	73.7	28.0	0.0	56.0	79.7
	Lower Int. Shell Axial Welds	1-233C	27204/12008	LINDE 1092	-50.0	0.219	0.996	231.1	73.7	28.0	0.0	56.0	79.7
	Lower/Lower Int. Shell Circ Weld	1-240	21935	LINDE 1092	-50.0	0.183	0.704	172.2	63.9	28.0	0.0	56.0	69.9
Nozzles	Nozzle N-16A	G-2822	EV-26067		-10.0	0.13	0.65	92.3	16.5	8.3	0.0	16.5	23.0
	Nozzle N-16B	G-2822	EV-26067		10.0	0.16	0.62	118.5	21.2	10.6	0.0	21.2	52.4

Fluence Data

	Beltline ID	Code No.	Heat No.	Wall Thickness (in.) Full	Wall Thickness (in.) 1/4t	Fluence at ID (n/cm^2)	Attenuation $e^{-0.24x}$	Fluence at 1/4t (n/cm^2)	Fluence Factor, FF $f^{(0.28 - 0.10 \log f)}$
Plates	Lower Shell Plate	G-2803-1	C2274-1	6.375	1.59	1.09E+18	0.68	7.44E+17	0.360
	Lower Shell Plate	G-2803-2	C2307-1	6.375	1.59	1.09E+18	0.68	7.44E+17	0.360
	Lower Shell Plate	G-2803-3	C2274-2	6.375	1.59	1.09E+18	0.68	7.44E+17	0.360
	Lower Int. Shell Plate	G-2802-1	C2331-2	5.375	1.34	1.41E+18	0.72	1.02E+18	0.421
	Lower Int. Shell Plate	G-2802-2	C2307-2	5.375	1.34	1.41E+18	0.72	1.02E+18	0.421
	Lower Int. Shell Plate	G-2801-7	C2407-1	5.375	1.34	1.41E+18	0.72	1.02E+18	0.421
Welds	Lower Shell Axial Welds	2-233A	12420	6.375	1.59	1.07E+18	0.68	7.30E+17	0.357
	Lower Shell Axial Welds	2-233B	12420	6.375	1.59	1.07E+18	0.68	7.30E+17	0.357
	Lower Shell Axial Welds	2-233C	12420	6.375	1.59	1.07E+18	0.68	7.30E+17	0.357
	Lower Int. Shell Axial Welds	1-233A	27204/12008	5.375	1.34	8.11E+17	0.72	5.87E+17	0.319
	Lower Int. Shell Axial Welds	1-233B	27204/12008	5.375	1.34	8.11E+17	0.72	5.87E+17	0.319
	Lower Int. Shell Axial Welds	1-233C	27204/12008	5.375	1.34	8.11E+17	0.72	5.87E+17	0.319
	Lower/Lower Int. Shell Circ Weld	1-240	21935	5.375	1.34	1.09E+18	0.72	7.90E+17	0.371
Nozzles	Nozzle N-16A	G-2822	EV-26067	5.375	1.34	2.94E+17	0.72	2.13E+17	0.179
	Nozzle N-16B	G-2822	EV-26067	5.375	1.34	2.94E+17	0.72	2.13E+17	0.179

Appendix A

COOPER REACTOR VESSEL MATERIALS SURVEILLANCE PROGRAM

In accordance with 10 CFR 50, Appendix H, Reactor Vessel Material Surveillance Program Requirements [18], two surveillance capsules were removed from the CNS reactor vessel in 1985 at 6.8 EFPY and 1991 at 11.2 EFPY [20, 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 the 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 [23]. Under the ISP, a capsule was scheduled for removal in 2003 but removal has been deferred to approximately 2017 at 32 EFPY [21]. 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 [21]. 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 [21] 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 [21].