BRAIDWOOD UNIT 2

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

Revision 8

BRAIDWOOD - UNIT 2
PRESSURE AND TEMPERATURE LIMITS REPORT

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1.0 Introduction

This Pressure and Temperature Limits Report (PTLR) for Braidwood Unit 2 has been prepared in accordance with the requirements of Braidwood Technical Specification (TS) 5.6.6, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)". Revisions to the PTLR shall be provided to the NRC after issuance.

The Technical Specifications addressed in this report are listed below:

LCO 3.4.3 RCS Pressure and Temperature (P/T) Limits; and
LCO 3.4.12 Low Temperature Overpressure Protection (LTOP) System.

2.0 RCS Pressure Temperature Limits

The PTLR limits for Braidwood Unit 2 were developed using a methodology specified in the Technical Specifications. The methodology listed in WCAP-14040-A, Revision 4 (Reference 1) was used with the following exception:

a) Elimination of the flange requirements documented in WCAP-16143-P.
b) The initial reference temperatures of the inlet/outlet nozzle forging to shell welds are determined using BAW-2308 in lieu of the ASME NB-2300 requirements.

WCAP-18370-NP, Revision 0 (Reference 7), provides the basis for the Braidwood Unit 2 P/T curves, along with the best estimate chemical compositions, fluence projections and adjusted reference temperatures used to determine these limits. The "Master Curve" fracture toughness properties from BAW-2308 Revision 1-A Safety Evaluation (SE) and Revision 2-A SE (Reference 2) are used for the inlet/outlet nozzle to upper shell forgings welds. WCAP-16143-P, (Reference 8), documents the technical basis for the elimination of the flange requirements. These exceptions to the methods in WCAP-14040-A, Revision 4 have been reviewed and accepted by the NRC in References 9, 10, 11 and 12.

2.1 RCS Pressure and Temperature (P/T) Limits (LCO 3.4.3)

2.1.1 The RCS temperature rate-of-change limits defined in WCAP-18370-NP, Revision 0 (Reference 7) are:

a. A maximum heatup of 100°F in any 1-hour period.

b. A maximum cooldown of 100°F in any 1-hour period, and

c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

2.1.2 The RCS P/T limits for heatup, inservice hydrostatic and leak testing, and criticality are specified by Figure 2.1 and Table 2.1a. The RCS P/T limits for cooldown are shown in Figure 2.2 and Table 2.1b. These limits are in WCAP-18370-NP, Revision 0 (Reference 7) using the limiting material between Braidwood Units 1 and 2. This approach is conservative. Consistent with the methodology described in Reference 1, with the exception noted in Section 2.0, the RCS P/T limits for heatup and cooldown shown in Figures 2.1 and 2.2 are provided without margins for instrument error. These limits were developed using ASME Boiler and Pressure Vessel Code Section XI, Appendix G, 1998 Edition through the 2000 Addenda. The criticality limit curve specifies pressure-temperature limits for core operation to provide additional margin during actual power production as specified in 10 CFR 50, Appendix G.

The P/T limits for core operation (except for low power physics testing) are that the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum permissible temperature in the corresponding P/T curve for heatup and cooldown.

Material Property Basis Limiting Material: BRAIDWOOD UNIT 2 Nozzle Shell Forging 5P-7056 Limiting ART Values at 57 EFPY 1/4T 75°F 3/4T 61°F

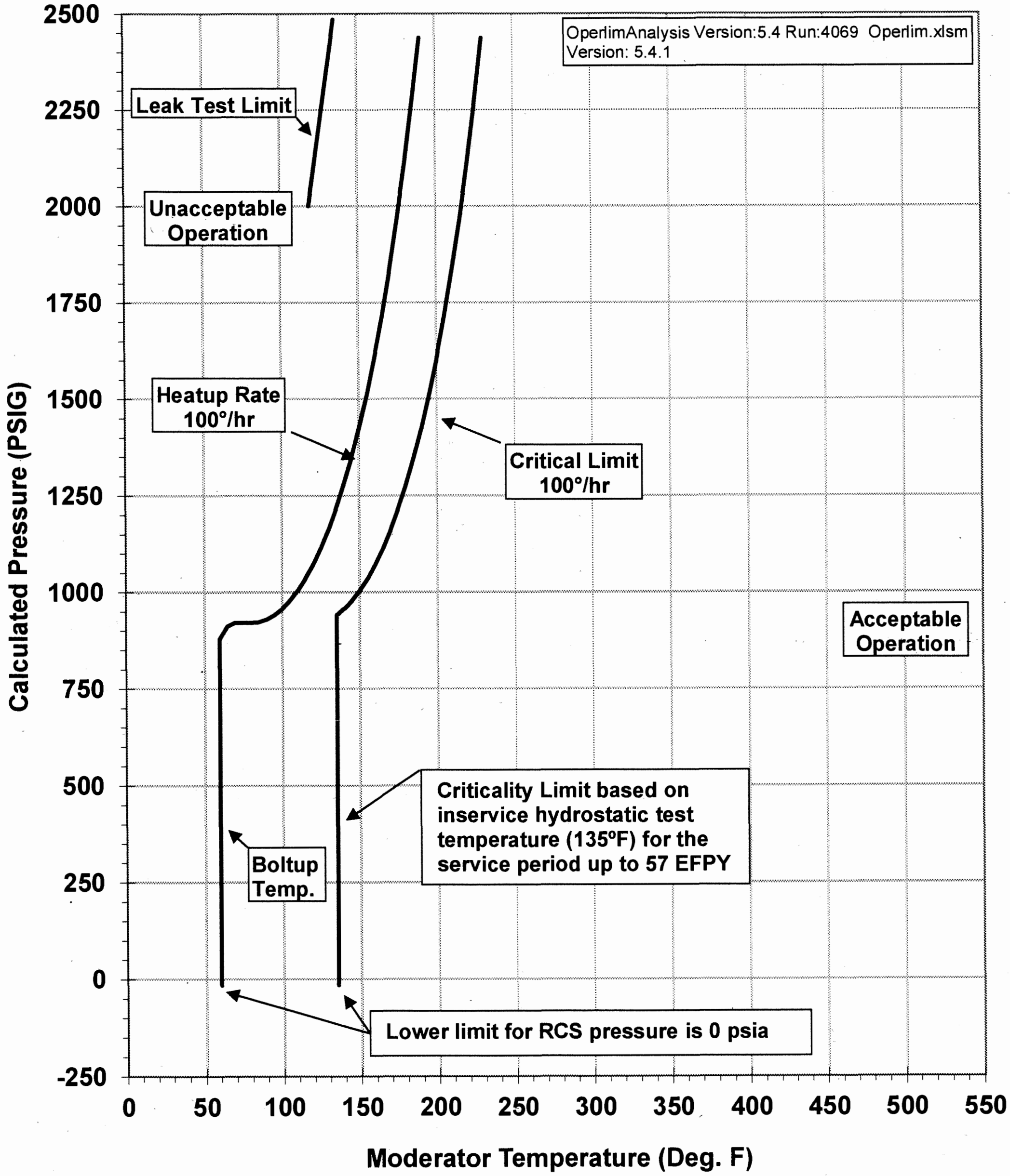

Figure 2.1

Braidwood Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable to 57 EFPY (Without Margins for Instrumentation Errors)

Material Property Basis Limiting Material: BRAIDWOOD UNIT 2 Nozzle Shell Forging 5P-7056 Limiting ART Values at 57 EFPY 1/4T 75°F 3/4T 61°F

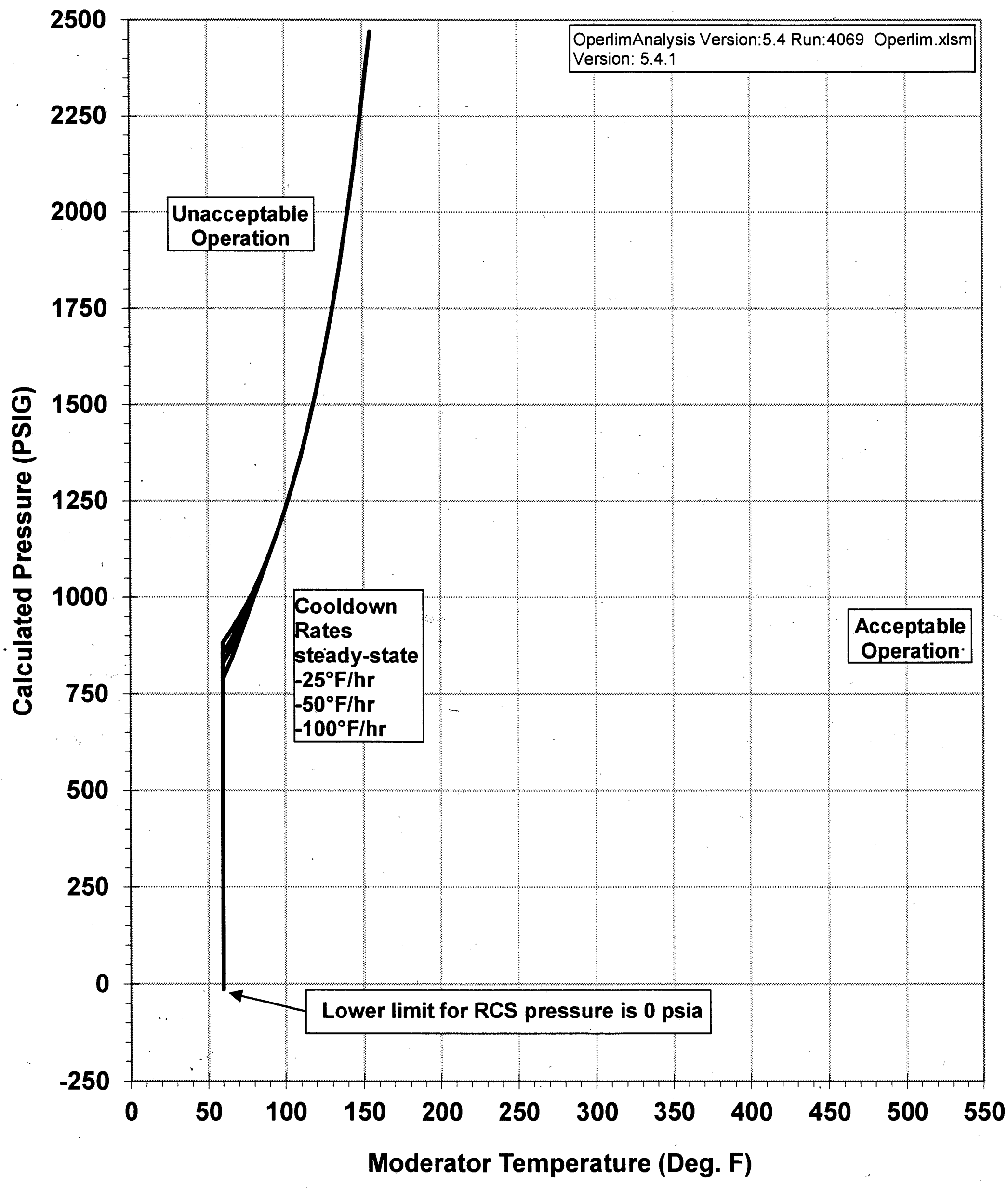

Figure 2.2
Braidwood Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50, and 100°F/hr) Applicable to 57 EFPY (Without Margins for Instrumentation Errors)

Table 2.1a
Braidwood Unit 2 Heatup Data Points at 57 EFPY
(Without Margins for Instrumentation Errors)

Heatup Curve					
100 F Heatup		Criticality Limit		Leak Test Limit	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
60	Note 1	135	Note 1	118	2000
60	879	135	940	135	2485
65	912	140	957		
70	921	145	978		
75	921	150	1004		
80	921	155	1035		
85	923	160	1071		
90	929	165	1113		
95	940	170	1161		
100	957	175	1215		
105	978	180	1277		
110	1004	185	1345		
115	1035	190	1422		
120	1071	195	1508		
125	1113	200	1604		
130	1161	205	1710		
135	1215	210	1827		
140	1277	215	1957		
145	1345	220	2102		
150	1422	225	2261		
155	1508	230	2437		
160	1604				
165	1710				
170	1827				
175	1957				
180	2102				
185	2261				
190	2437				

Note 1: The Minimum acceptable pressure is 0 psia

Table 2.1b
Braidwood Unit 2 Cooldown Data Points at 57 EFPY
(Without Margins for Instrumentation Errors)

Cooldown Curves							
Steady State		25 °F Cooldown		50 °F Cooldown		100 °F Cooldown	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
60	Note 1	60	Note 1	60	Note 1	60	Note 1
60	882	60	854	60	828	60	788
65	912	65	886	65	864	65	835
70	944	70	923	70	905	70	887
75	980	75	963	75	950	75	944
80	1020	80	1007	80	1000	80	1000
85	1063	85	1056	85	1055	85	1055
90	1112	90	1110	90	1110	90	1110
95	1165	95	1165	95	1165	95	1165
100	1224	100	1224	100	1224	100	1224
105	1290	105	1290	105	1290	105	1290
110	1362	110	1362	110	1362	110	1362
115	1442	115	1442	115	1442	115	1442
120	1530	120	1530	120	1530	120	1530
125	1627	125	1627	125	1627	125	1627
130	1735	130	1735	130	1735	130	1735
135	1854	135	1854	135	1854	135	1854
140	1986	140	1986	140	1986	140	1986
145	2131	145	2131	145	2131	145	2131
150	2292	150	2292	150	2292	150	2292
155	2469	155	2469	155	2469	155	2469

Note 1: The Minimum acceptable pressure is 0 psia

3.0 Low Temperature Overpressure Protection and Boltup

This section provides the Braidwood Unit 2 low temperature overpressure protection (LTOP) system pressurizer power operated relief valve (PORV) lift settings, LTOP system arming temperature, and minimum reactor vessel boltup temperature.

3.1 LTOP System Setpoints (LCO 3.4.12).

Two PORVs shall each have nominal lift settings in accordance with Figure 3.1 and Table 3.1. These limits are based on Reference 3.

The LTOP setpoints are based on P/T limits that were established in accordance with 10 CFR 50, Appendix G without allowance for instrumentation error. The LTOP setpoints were developed using the methodology described in Reference 1. The LTOP PORV nominal lift settings shown in Figure 3.1 and Table 3.1 account for appropriate instrument error.

3.2 LTOP Enable Temperature

Braidwood Unit 2 procedures governing the heatup and cooldown of the RCS require the arming of the LTOP system for RCS temperature less than 350°F and disarming of LTOP for RCS temperature of 350°F and above.

Note that the last LTOP PORV segment in Table 3.1 extends to 400°F where the pressure setpoint is 2335 psig. This is intended to prohibit PORV lift for an inadvertent LTOP system arming at power.

3.3 Reactor Vessel Boltup Temperature (Non-Technical Specification)

The minimum boltup temperature for the Reactor Vessel Flange shall be ≥ 60°F. Boltup is a condition in which the Reactor Vessel head is installed with tension applied to any stud, and with the RCS vented to atmosphere.

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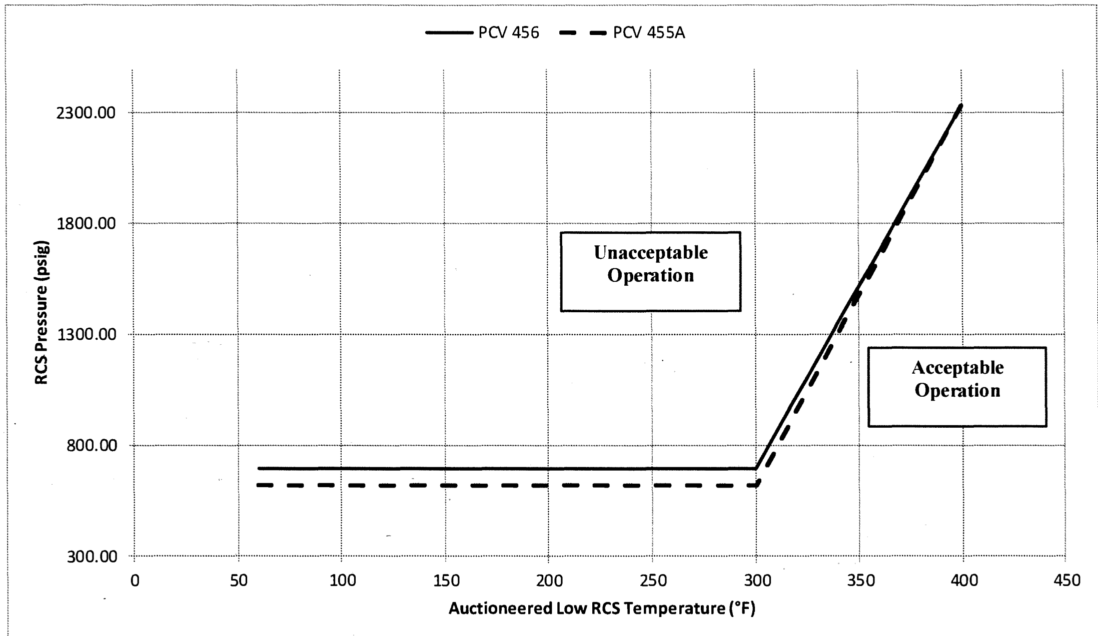

Figure 3.1
Braidwood Unit 2 Nominal PORV Setpoints for the Low Temperature Overpressure Protection (LTOP) System Applicable for 57 EFPY (Includes Instrumentation Uncertainty)

Table 3.1
Data Points for Braidwood Unit 2 Nominal PORV Setpoints for the LTOP System Applicable for 57 EFPY (Includes Instrumentation Uncertainty)

PCV-455A

RCS TEMP. (DEG. F)	RCS Pressure (PSIG)
60	620
300	620
400	2335

PCV-456

RCS TEMP. (DEG. F)	RCS Pressure (PSIG)
60	695
300	695
400	2335

Note: To determine nominal lift setpoints for RCS Pressure and RCS Temperatures greater than 300°F, linearly interpolate between the 300°F and 400°F data points shown above. (Setpoints extend to 400°F to prevent PORV liftoff from an inadvertent LTOP system arming while at power).

4.0 Reactor Vessel Material Surveillance Program

The pressure vessel material surveillance program (Reference 4) is in compliance with Appendix H to 10 CFR 50, "Reactor Vessel Radiation Surveillance Program." The material test requirements and the acceptance standards utilize the reference nil-ductility temperature, RT_{NDT}, which is determined in accordance with ASME Boiler and Pressure Vessel Code, Section III, NB-2331. The empirical relationship between RT_{NDT} and the fracture toughness of the reactor vessel steel is developed in accordance with Appendix G, "Protection Against Non-Ductile Failure," to Section XI of the ASME Boiler and Pressure Vessel Code. The surveillance capsule removal schedule meets the requirements of ASTM E185-82.

The fourth reactor vessel material irradiation surveillance specimens (Capsule V) have been analyzed to determine changes in material properties (Reference 5). The surveillance capsule testing has been completed for the original operating period. The remaining two capsules, Y and Z, were removed and placed in the spent fuel pool to avoid excessive fluence accumulation should they be needed to support life extension. The removal summary is provided in Table 4.1.

Table 4.1				
Braidwood Unit 2 Capsule Withdrawal Summary[a]				
Capsule	**Capsule Location**	**Lead Factor**	**Withdrawal EFPY[b]**	**Fluence (n/cm^2, E>1.0 MeV)**
U	58.5°	4.08	1.18 (EOC 1)	0.387×10^{19}
X	238.5°	4.03	4.24 (EOC 4A)	1.15×10^{19}
W	121.5°	4.06	8.56 (EOC 7)	2.07×10^{19}
Z[c]	301.5°	4.15	12.78 (EOC 10)	2.83×10^{19}
Y[c]	241.0°	3.90	12.78 (EOC 10)	2.66×10^{19}
V	61.0°	3.92	18.41 (EOC 14)	3.73×10^{19}

Notes:

(a) Source document is WCAP-18107-NP (Reference 5), Table 7-1.
(b) Effective Full Power Years (EFPY) from plant startup.
(c) Standby Capsules Z and Y were removed and placed in the spent fuel pool. No testing or analysis has been performed on these capsules.

5.0 Supplemental Data Tables

The following tables provide supplemental information on reactor vessel material properties and are provided to be consistent with Generic Letter 96-03. Some of the material property values shown were used as inputs to the P/T limits.

Table 5.1 shows the calculation of the surveillance material chemistry factors using surveillance capsule data.

Table 5.2 provides the reactor vessel material properties table.

Table 5.3 provides a summary of the Braidwood Unit 2 adjusted reference temperature (ART) values at the 1/4T and 3/4T locations for 57 EFPY.

Table 5.4 shows the calculation of ARTs at 57 EFPY for the limiting Braidwood Unit 2 reactor vessel material, i.e. Nozzle Shell Forging 5P-7056.

Table 5.5 provides the RT_{PTS} Calculation for Braidwood Unit 2 Beltline and Extended Beltline Regions Materials at EOLE (57 EFPY), (Reference 7).

Table 5.1

Braidwood Unit 2 Calculation of Chemistry Factors Using Surveillance Capsule Data(a)

Material	Capsule	Capsule f(b) (n/cm², E > 1.0 MeV)	FF(c)	ΔRT_{NDT}(b) (°F)	FF*ΔRT_{NDT} (°F)	FF²
Lower Shell Forging (Tangential)	U	0.387×10^{19}	0.737	0.0(d)	0.00	0.54
	X	1.15×10^{19}	1.039	0.0(d)	0.00	1.08
	W	2.07×10^{19}	1.198	4.6	5.51	1.44
	V	3.73×10^{19}	1.341	28.4	38.08	1.80
Lower Shell Forging (Axial)	U	0.387×10^{19}	0.737	0.0(d)	0.00	0.54
	X	1.15×10^{19}	1.039	33.8	35.12	1.08
	W	2.07×10^{19}	1.198	33.1	39.66	1.44
	V	3.73×10^{19}	1.341	63.3	84.88	1.80
				SUM:	203.25	9.71
	$CF_{LS\ Forging} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (203.25) \div (9.71) = \mathbf{20.9°F}$					
Braidwood Unit 1 Surveillance Weld Material	U	0.388×10^{19}	0.738	17.4	12.84	0.54
	X	1.17×10^{19}	1.044	29.8	31.11	1.09
	W	1.98×10^{19}	1.186	49.0	58.14	1.41
	V	3.71×10^{19}	1.340	62.8	84.13	1.79
Braidwood Unit 2 Surveillance Weld Material	U	0.387×10^{19}	0.737	0.0(d)	0.00	0.54
	X	1.15×10^{19}	1.039	26.1	27.12	1.08
	W	2.07×10^{19}	1.198	23.7	28.39	1.44
	V	3.73×10^{19}	1.341	45.6	61.14	1.80
				SUM:	302.87	9.69
	$CF_{Weld\ Metal} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (302.87) \div (9.69) = \mathbf{31.2°F}$					

<u>Notes</u>:

(a) Source document is WCAP-18370-NP (Reference 7), Table 5-2 and Table 5-3.
(b) f = fluence; ΔRT_{NDT} values are the measured 30 ft-lb shift values taken from Reference 5.
(c) FF = fluence factor = $f^{(0.28 - 0.10*\log f)}$
(d) Measured ΔRT_{NDT} values were determined to be negative, but physically a reduction should not occur; therefore, conservative values of zero are used.

Table 5.2				
Braidwood Unit 2 Reactor Vessel Material Properties				
Material Description	Cu (%)	Ni (%)	Chemistry Factor	Initial RT_{NDT} (°F)[a]
Closure Head Flange Heat # 3P6566/5P7547/4P6986 Serial # 2031-V-1	--	0.75	--	20
Vessel Flange Heat # 124P455	0.07	0.70	--	20
Inlet Nozzle 01-001 Heat # 41-5414	0.07	0.83	44[b]	-10
Inlet Nozzle 01-002 Heat # 41-5414	0.07	0.85	44[b]	-10
Inlet Nozzle 02-001 Heat # 42-5417	0.09	0.88	58[b]	-10
Inlet Nozzle 02-002 Heat # 42-5417	0.09	0.89	58[b]	-10
Outlet Nozzle 01-002 Heat # 11-5266	0.09	0.86	58[b]	10
Outlet Nozzle 01-003 Heat # 11-5226	0.09	0.88	58[b]	-10
Outlet Nozzle 02-001 Heat # 4-3481	0.07	0.84	44[b]	-10
Outlet Nozzle 02-002 Heat # 4-3502	0.09	0.78	58[b]	-10
Nozzle Shell Forging * Heat # 5P-7056	0.04	0.90	26.0°F[b]	30
Intermediate Shell Forging * Heat # [49D963/49C904]-1-1	0.03	0.71	20.0°F[b]	-30
Lower Shell Forging * Heat # [50D102/50C97]-1-1	0.06	0.76	37.0°F[b] 20.9°F[c]	-30
Circumferential Weld * (Intermediate Shell to Lower Shell) Weld Seam WF-562 Heat # 442011	0.03	0.67	41.0F[b] 31.2F[c]	40
Circumferential Weld * (Nozzle Shell to Intermediate Shell) Weld Seam WF-645 Heat # H4498	0.04	0.46	54.0°F[b]	-25
Inlet Nozzle to Nozzle Shell Forging** Circumferential Weld Seams WF-654 (HT# 41404)	0.18	0.52	167[d]	-48.6[e]
Outlet Nozzle to Nozzle Shell Forging** Circumferential Weld Seams WF-654 (HT# 41404)	0.18	0.52	167[d]	-48.6[e]

Notes contained on the following page.

Notes: Source document is WCAP-18370-NP (Reference 7), Table 3-2 and Table 5-5
* Beltline Region Materials
** Extended Beltline Region Materials

a) The Initial RT_{NDT} values for the plates and welds are based on measured data.
b) Chemistry Factor calculated for Cu and Ni values per Regulatory Guide 1.99, Rev. 2, Position 1.1.
c) Chemistry Factor calculated for Cu and Ni values per Regulatory Guide 1.99, Rev. 2, Position 2.1
d) Value is the required minimum per condition from BAW-2308 (Reference 2).
e) Generic value taken from BAW-2308 (Reference 2)

Table 5.3

Summary of Braidwood Unit 2 Adjusted Reference Temperature (ART) Values at 1/4T and 3/4T Locations for 57 EFPY[(a)]

Reactor Vessel Material	Surface Fluence (n/cm², E>1.0 MeV)	57 EFPY	
		1/4T ART (°F)	3/4T ART (°F)
Inlet Nozzle 01-001	1.06×10^{17}	0.0[(b)]	
Inlet Nozzle 01-002	1.06×10^{17}	0.0[(b)]	
Inlet Nozzle 02-001	1.06×10^{17}	3.2[(b)]	
Inlet Nozzle 02-002	1.06×10^{17}	3.2[(b)]	
Outlet Nozzle 01-002	7.97×10^{16}	20.9[(b)]	
Outlet Nozzle 01-003	7.97×10^{16}	0.9[(b)]	
Outlet Nozzle 02-001	7.97×10^{16}	-1.7[(b)]	
Outlet Nozzle 02-002	7.97×10^{16}	0.9[(b)]	
Nozzle Shell Forging	0.994×10^{19}	74.5	60.5
Intermediate Shell Forging	2.95×10^{19}	16.3	5.0
Lower Shell Forging	3.03×10^{19}	47.1	35.3
→Using non-credible surveillance data	3.03×10^{19}	18.7	6.9
Nozzle to Intermediate Shell Forging Circ. Weld Seam (Heat # H4498)	0.994×10^{19}	67.4	38.4
Intermediate to Lower Shell Forging Circ. Weld Seam (Heat # 442011)	2.90×10^{19}	134.5	111.3
→Using credible surveillance data	2.90×10^{19}	104.0	94.3
Inlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-654 (HT# 41404)	1.06×10^{17}	37.0[(b)]	
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-654 (HT# 41404)	7.97×10^{16}	33.6[(b)(c)]	

Notes:

(a) The source document containing detailed calculations is WCAP-18370-NP (Reference 7), Table 7-2, Table 7-6, Table 7-7, and Table 7-8.

(b) The ART values for the extended beltline materials are conservatively calculated at the surface, i.e., without attenuation of the fluence.

(c) The outlet nozzle materials do not exceed the 1×10^{17} n/cm² fluence threshold at 57 EFPY; therefore, neutron irradiation embrittlement need not be considered for the nozzle materials. However, the results are included for information.

Table 5.4

Braidwood Unit 2 Calculation of Adjusted Reference Temperatures (ARTs) at 57 EFPY at the Limiting Reactor Vessel Material, Nozzle Shell Forging 5P-7056

Parameter	Values	
Operating Time	57 EFPY	
Location[a]	1/4T ART (°F)	3/4T ART(°F)
Chemistry Factor, CF (°F)	26.0	26.0
Fluence(f), n/cm^2 (E>1.0 Mev)[b]	$5.97x10^{18}$	$2.15x10^{18}$
Fluence Factor, FF	0.856	0.587
ΔRT_{NDT}= CFxFF(°F)	22.2	15.3
Initial RT $_{NDT,}$, I(°F)	30	30
Margin, M(°F)	22.2	15.3
ART= I+(CF*FF)+M, °F per RG 1.99, Revision 2	74.5	60.5

<u>Notes</u>: Source document is WCAP-18370-NP (Reference 7), Table 7-6, Table 7-7 and Table 7-9

a) The Braidwood Unit 2 reactor vessel wall thickness is 8.5 inches at the beltline region.
b) Fluence, f, is the calculated peak clad/base metal interface fluence (E>1.0 Mev) = $9.94x10^{18}$ n/cm^2 at 57 EFPY (Reference 7).

Table 5.5

RT_{PTS} Calculation for Braidwood Unit 2 Beltline and Extended Beltline Regions Materials at EOLE (57 EFPY)[a,b]

Reactor Vessel Material	R.G. 1.99, Rev. 2 Position	CF (°F)	Fluence (n/cm², E>1.0 MeV)	FF	IRT_{NDT}[c] (°F)	ΔRT_{NDT} (°F)	σ_u[c] (°F)	σ_Δ[d] (°F)	Margin (°F)	RT_{PTS} (°F)
Reactor Vessel Extended Beltline Materials										
Inlet Nozzle 01-001	1.1	44.0	1.06×10^{17}	0.114	-10	5.0	0	2.5	5.0	0.0
Inlet Nozzle 01-002	1.1	44.0	1.06×10^{17}	0.114	-10	5.0	0	2.5	5.0	0.0
Inlet Nozzle 02-001	1.1	58.0	1.06×10^{17}	0.114	-10	6.6	0	3.3	6.6	3.2
Inlet Nozzle 02-002	1.1	58.0	1.06×10^{17}	0.114	-10	6.6	0	3.3	6.6	3.2
Outlet Nozzle 01-002	1.1	58.0	7.97×10^{16}	0.094	10	5.4[e]	0	2.7	5.4	20.9
Outlet Nozzle 01-003	1.1	58.0	7.97×10^{16}	0.094	-10	5.4[e]	0	2.7	5.4	0.9
Outlet Nozzle 02-001	1.1	44.0	7.97×10^{16}	0.094	-10	4.1[e]	0	2.1	4.1	-1.7
Outlet Nozzle 02-002	1.1	58.0	7.97×10^{16}	0.094	-10	5.4[e]	0	2.7	5.4	0.9
Inlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams	1.1	167.0[f]	1.06×10^{17}	0.114	-48.6[f]	19.0	18[f]	28.0[f]	66.6	37.0
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams	1.1	167.0[f]	7.97×10^{16}	0.094	-48.6[f]	15.7[e]	18[f]	28.0[f]	66.6	33.6

Table 5.5 (continued)

Reactor Vessel Beltline Materials										
Nozzle Shell Forging	1.1	26	0.994×10^{19}	0.998	30	26.0	0	13.0	26.0	81.9
Intermediate Shell Forging	1.1	20	2.95×10^{19}	1.287	-30	25.7	0	12.9	25.7	21.5
Lower Shell Forging	1.1	37	3.03×10^{19}	1.293	-30	47.8	0	17	34	51.8
→Using non-credible surveillance data	2.1	20.9	3.03×10^{19}	1.293	-30	27.0	0	13.5	27.0	24.1
Nozzle to Intermediate Shell Forging Circ. Weld Seam (Heat # H4498)	1.1	54	0.994×10^{19}	0.998	-25	53.9	0	27.0	53.9	82.8
Intermediate to Lower Shell Forging Circ. Weld Seam (Heat # 442011)	1.1	41	2.90×10^{19}	1.283	40	52.6	0	26.3	52.6	145.2
→Using credible surveillance data	2.1	31.2	2.90×10^{19}	1.283	40	40.0	0	14	28	108

Notes:

(a) The 10 CFR 50.61 methodology was utilized in the calculation of the RT_{PTS} values.
(b) The source document containing detailed calculations is WCAP-18370-NP (Reference 7), Table E-2.
(c) Initial RT_{NDT} values are based on measured data. Hence, $\sigma_u = 0°F$.
(d) Per the guidance of 10 CFR 50.61, the base metal $\sigma_\Delta = 17°F$ for Position 1.1 (without surveillance data) and for Position 2.1 with non-credible surveillance data; the weld metal $\sigma_\Delta = 28°F$ for Position 1.1 (without surveillance data) and with credible surveillance data $\sigma_\Delta = 14°F$ for Position 2.1. However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$.
(e) The outlet nozzle materials do not exceed the 1×10^{17} n/cm^2 fluence threshold at 57 EFPY; therefore, neutron irradiation embrittlement need not be considered for the nozzle materials. However, the results are included for information.
(f) The IRT_{NDT} values are based on BAW-2308 (Reference 2). Use of BAW-2308 as an exemption to the 10 CFR 50.61 methodology was approved in Reference 11. BAW-2308 requires the use of an $\sigma_I = 18°F$, $\sigma_\Delta = 28°F$, and a minimum CF of 167°F.

6.0 References

1. WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", J.D. Andrachek, et al., May 2004.

2. AREVA Document, BAW-2308, Revision 1-A and 2-A, "Initial RT_{NDT} of Linde 80 Weld Materials," August 2005 and March 2008.

3. LTR-SCS-19-14, Revision 1, "Braidwood Units 1 and 2 Low Temperature Overpressure Protection System (LTOPS) Analysis for 57 EFPY," September 19, 2019.

4. WCAP-11188, "Commonwealth Edison Company, Braidwood Station Unit 2 Reactor Vessel Surveillance Program," December 1986.

5. WCAP-18107-NP, Revision 0, "Analysis of Capsule V from the Exelon Generation Braidwood Unit 2 Reactor Vessel Radiation Surveillance Program," May 2016.

6. Not Used.

7. WCAP-18370-NP, Revision 0, "Braidwood Units 1 and 2 Heatup and Cooldown Limits for Normal Operation," June 2019.

8. WCAP-16143-P, Revision 1, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2," W. Bamford, et al., October 2014.

9. NRC Letter from R. F. Kuntz, NRR, to C. M. Crane, Exelon Generation Company, LLC, "Byron Station, Unit Nos. 1 and 2 and Braidwood Station Unit Nos. 1 and 2 – Exemption from the Requirements of 10 CFR Part 50, Appendix G (TAC Nos. MC8697, MC8698, MC8699, and MC8700)," November 22, 2006. [ADAMS Accession Number ML061890003]

10. NRC Letter from J. S. Wiebe, NRR, to B.C. Hanson, Exelon Generation Company, LLC, "Braidwood Station, Units 1 and 2, and Byron Station, Unit Nos. 1 and 2 – Issuance of Amendments to Utilize WCAP-16143-P, Revision 1 "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2," Dated October 16, 2014 (CAC Nos. MF5033, MF5034, MF5035 and MF5036)," October 28, 2015. [ADAMS Accession Number ML15232A441]

11. NRC Letter from J. S. Wiebe, NRR, to B.C. Hanson, Exelon Generation Company, LLC, "Braidwood Station, Units 1 and 2, and Byron Station, Unit Nos. 1 and 2- Exemption from the Requirements of 10CFR50.61 and 10FR50, Appendix G (EPID L-2019-LLE-0022)," August 31, 2020. [ADAMS Accession Number ML20022A336]

12. Issuance of Amendment Nos. 217, 217, 221, and 221 Regarding Reactor Coolant System Pressure and Temperature Limits Report Technical Specifications (EPID L-2019-LLA-0215)," September 18, 2020. [ADAMS Accession Number ML20163A046]. NRC Letter from J. S. Wiebe, NRR, to B.C. Hanson, Exelon Generation Company, LLC, "Braidwood Station, Units 1 and 2, and Byron Station, Unit Nos. 1 and 2- Issuance of Amendment Nos. 217, 217, 221, and 221 Regarding Reactor Coolant System Pressure and Temperature Limits Report Technical Specifications (EPID L-2019-LLA-0215)," September 18, 2020. [ADAMS Accession Number ML20163A046].