Exelon Generation Company, LLC

Braidwood Station
35100 South Route 53, Suite 84
Braceville, IL 60407-9619

www.exeloncorp.com

May 9, 2014
BW140044

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

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and 50-457

Subject: Braidwood Unit 1 - Pressure and Temperature Limits Reports (PTLRs), Revision 7
Braidwood Unit 2 - Pressure and Temperature Limits Reports (PTLRs), Revision 6

The purpose of this letter is to transmit revisions to the Pressure and Temperature Limits Reports (PTLRs) for Braidwood Station, Units 1 and 2 in accordance with Technical Specification (TS) 5.6.6, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)."

The lower limit for Reactor Coolant System (RCS) pressure associated with RCS heatup and RCS cooldown was extended from 0 psig to 0 psia to bound RCS pressure experienced during RCS vacuum fill operations and to maintain compliance with Technical Specification 3.4.3, "RCS Pressure and Temperature (P/T) Limits." The RCS heatup limitations (i.e., Figure 2.1 and Table 2.1a) and RCS cooldown limitations (i.e., Figure 2.2 and Table 2.1b) were revised accordingly.

Please direct any questions you may have regarding this matter to Mr. Phillip Raush, Regulatory Assurance Manager, at (815) 417-2800.

Sincerely,

Mark E. Kanavos
Site Vice President
Braidwood Station

Attachments: 1. Braidwood Unit 1 Pressure Temperature Limits Report (PTLR), Revision 7
2 Braidwood Unit 2 Pressure Temperature Limits Report (PTLR), Revision 6

cc: NRC Regional Administrator, Region III
NRS Senior Resident Inspector, Braidwood Station
NRC Project Manager, NRR – Braidwood and Byron Stations
Illinois Emergency Management Agency – Division of Nuclear Safety

ATTACHMENT 1

Braidwood Unit 1
Pressure and Temperature Limits Report (PTLR), Revision 7

BRAIDWOOD UNIT 1

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

Revision 7

BRAIDWOOD - UNIT 1
PRESSURE AND TEMPERATURE LIMITS REPORT

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BRAIDWOOD - UNIT 1
PRESSURE AND TEMPERATURE LIMITS REPORT

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BRAIDWOOD - UNIT 1
PRESSURE AND TEMPERATURE LIMITS REPORT

1.0 Introduction

This Pressure and Temperature Limits Report (PTLR) for Braidwood Unit 1 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 (TS) 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 and Temperature Limits

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

a) Optional use of ASME Code Section XI, Appendix G, Article G-2000, 1996 Addenda,
b) Use of ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves, Section XI, Division 1",
c) Use of ASME Code Case N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessel, Section XI, Division 1", and
d) Elimination of the flange requirements documented in WCAP-16143-P.

These exceptions to the methodology in WCAP 14040-NP-A, Revision 2 have been reviewed and accepted by the NRC in References 2, 8, 9 and 10.

WCAP 15364, Revision 2 (Reference 11), provides the basis for the Braidwood Unit 1 P/T curves, along with the best estimate chemical compositions, fluence projections and adjusted reference temperatures used to determine these limits. WCAP-16143-P, Reference 12, documents the technical basis for the elimination of the flange requirements.

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-15364, Revision 2 (Reference 11) 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 defined in WCAP-15364, Revision 2 (Reference 11). Consistent with the methodology described in Reference 1 and exceptions 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, Article G2000, 1996 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: NOZZLE SHELL FORGING 5P-7016
LIMITING ART VALUES AT 32 EFPY: 1/4T, 48°F
3/4T, 35°F

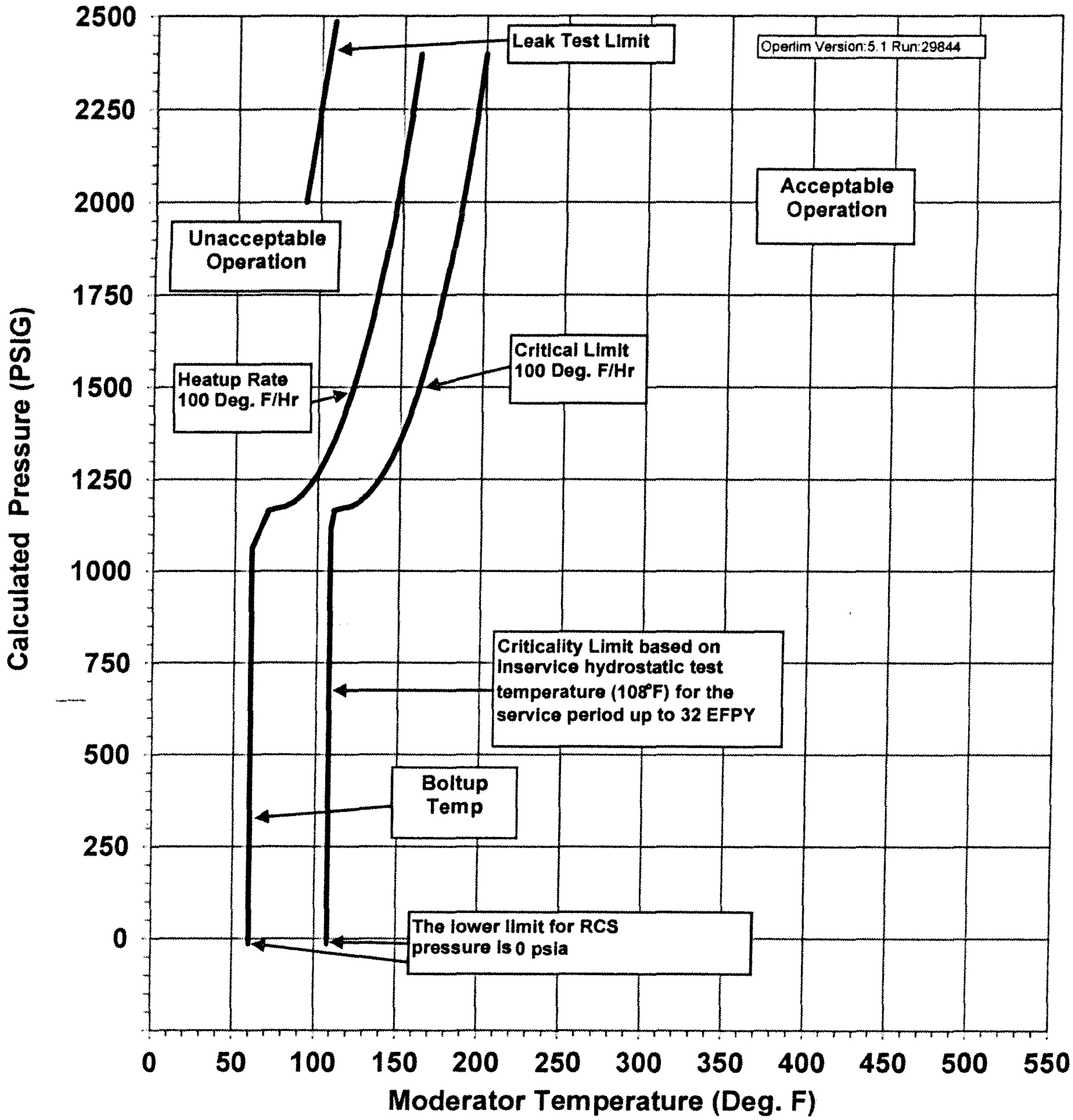

Figure 2.1
Braidwood Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable for 32 EFPY (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: NOZZLE FORGING 5P-7016
LIMITING ART VALUES AT 32 EFPY: 1/4T, 48°F
3/4T, 35°F

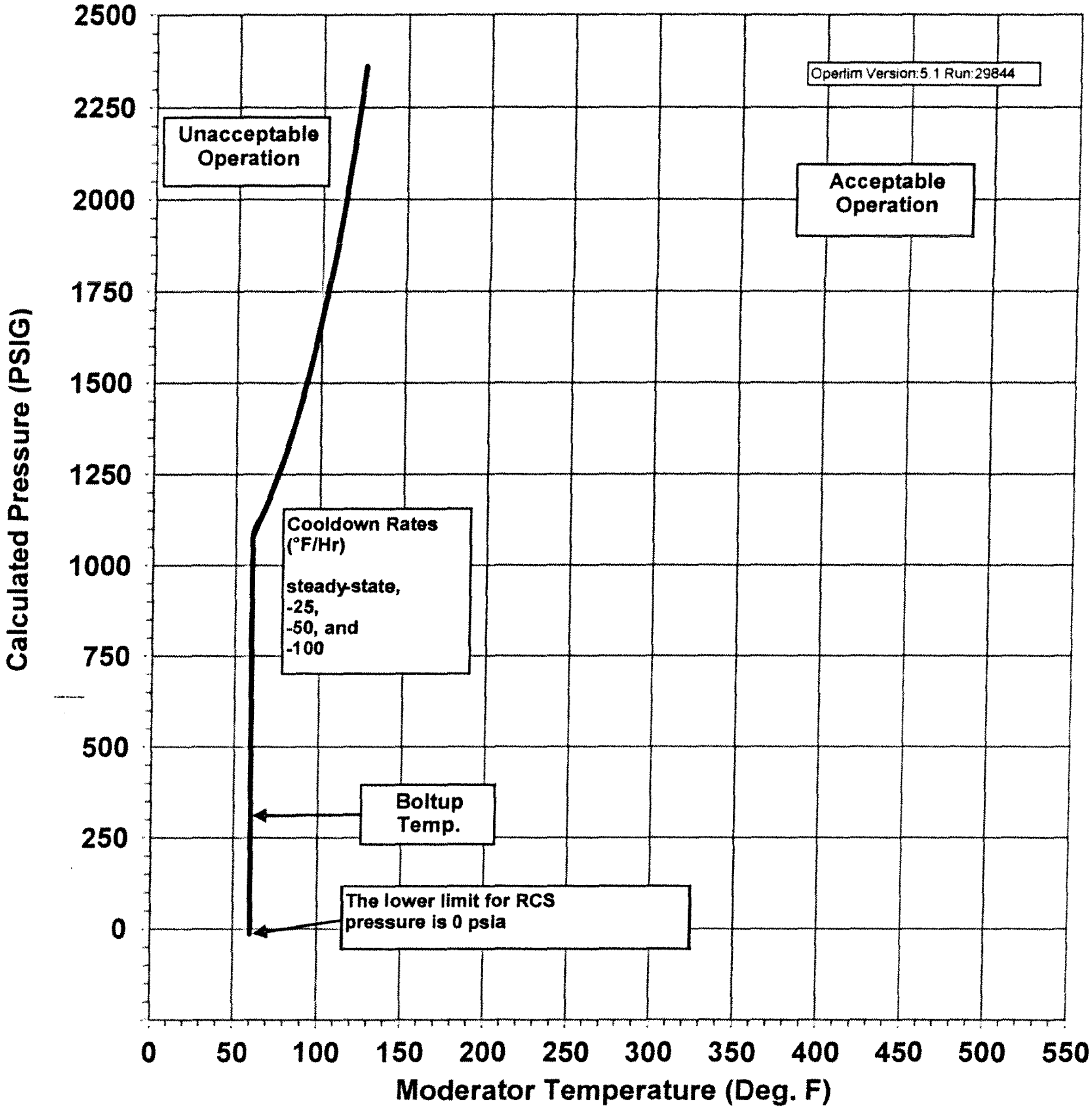

Figure 2.2
Braidwood Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50 and 100°F/hr) Applicable for 32 EFPY (Without Margins for Instrumentation Errors)

Table 2.1a
Braidwood Unit 1 Heatup Data Points at 32 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	108	Note 1	91	2000
60	1064	108	1114	108	2485
65	1114	110	1166		
70	1166	115	1172		
75	1172	120	1176		
80	1176	125	1188		
85	1188	130	1207		
90	1207	135	1234		
95	1234	140	1267		
100	1267	145	1308		
105	1308	150	1357		
110	1357	155	1414		
115	1414	160	1479		
120	1479	165	1554		
125	1554	170	1638		
130	1638	175	1732		
135	1732	180	1838		
140	1838	185	1956		
145	1956	190	2088		
150	2088	195	2235		
155	2235	200	2397		
160	2397				

Note 1:The Minimum acceptable pressure is 0 psia

Table 2.1b
Braidwood Unit 1 Cooldown Data Points at 32 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	1082	60	1078	60	1078*	60	1078*
65	1133	65	1133	65	1133	65	1133
70	1188	70	1188	70	1188	70	1188
75	1250	75	1250	75	1250	75	1250
80	1318	80	1318	80	1318	80	1318
85	1393	85	1393	85	1393	85	1393
90	1476	90	1476	90	1476	90	1476
95	1568	95	1568	95	1568	95	1568
100	1669	100	1669	100	1669	100	1669
105	1781	105	1781	105	1781	105	1781
110	1905	110	1905	110	1905	110	1905
115	2042	115	2042	115	2042	115	2042
120	2194	120	2194	120	2194	120	2194
125	2361	125	2361	125	2361	125	2361

* Refer to Reference 13

Note 1:The Minimum acceptable pressure is 0 psia

3.0 Low Temperature Overpressure Protection and Boltup

This section provides the Braidwood Unit 1 power operated relief valve lift settings, low temperature overpressure protection (LTOP) system arming temperature, and minimum reactor vessel boltup temperature.

3.1 LTOP System Setpoints (LCO 3.4.12)

The power operated relief valves (PORVs) shall each have maximum lift settings in accordance with Figure 3.1 and Table 3.1. These limits are based on References 3 and 4.

The LTOP setpoints are based on P/T limits which were established in accordance with 10 CFR 50, Appendix G without allowance for instrumentation error and in accordance with 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 1 procedures governing the heatup and cooldown of the RCS require the arming of the LTOP System for RCS temperature of 350°F and below and disarming of LTOP for RCS temperature above 350°F.

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.

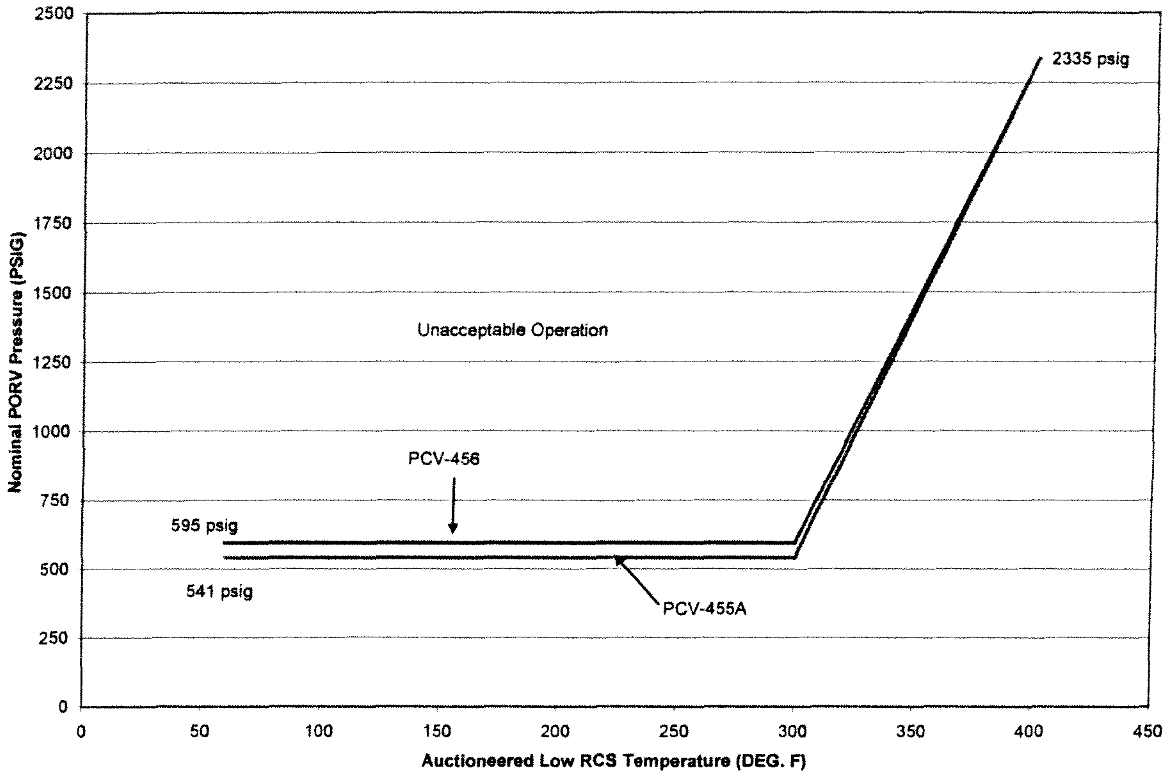

Figure 3.1
Braidwood Unit 1 Nominal PORV Setpoints for the Low Temperature Overpressure Protection (LTOP) System Applicable for 32 EFPY (Includes Instrumentation Uncertainty)

Table 3.1
Data Points for Braidwood Unit 1 Nominal PORV Setpoints for the LTOP System Applicable for 32 EFPY (Includes Instrumentation Uncertainty)

PCV-455A

(1TY-0413M)	
AUCTIONEERED LOW RCS TEMP. (DEG. F)	RCS PRESSURE (PSIG)
60	541
300	541
400	2335

PCV-456

(1TY-0413P)	
AUCTIONEERED LOW RCS TEMP. (DEG. F)	RCS PRESSURE (PSIG)
60	595
300	595
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 5) is in compliance with Appendix H to 10 CFR 50, "Reactor Vessel Radiation Surveillance Program." The material test requirements and the acceptance standard 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 third and final reactor vessel material irradiation surveillance specimens (Capsule W) have been removed and analyzed to determine changes in material properties. The surveillance capsule testing has been completed for the original operating period. The remaining three capsules, V, 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 1 Capsule Withdrawal Summary[a]

Capsule	Capsule Location	Lead Factor	Withdrawal EFPY[b]	Fluence (n/cm^2, E>1.0 MeV)
U	58.5°	4.02	1.16	0.388×10^{19}
X	238.5°	4.06	4.30	1.17×10^{19}
W	121.5°	4.05	7.79	1.98×10^{19}
Z[c]	301.5°	4.09	12.01 (EOC 10)	2.79×10^{19}
V[c]	61.0°	3.92	17.69 (EOC 14)	3.71×10^{19}
Y[c]	241.0°	3.81	12.01 (EOC 10)	2.60×10^{19}

Notes:

(a) Source document is CN-AMLRS-10-7 (Reference 14), Table 5.7-3.
(b) Effective Full Power Years (EFPY) from plant startup.
(c) Standby Capsules Z, V, and Y were removed and placed in the spent fuel pool. No testing or analysis has been performed on these capsules. If license renewal is sought, one of these standby capsules may need to be tested to determine the effect of neutron irradiation on the reactor vessel surveillance materials during the period of extended operation.

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 1 adjusted reference temperature (ART) values at the 1/4T and 3/4T locations for 32 EFPY.

Table 5.4 shows the calculation of ARTs at 32 EFPY for the limiting Braidwood Unit 1 reactor vessel material, i.e. weld WF-562 (HT # 442011, Based on Surveillance Capsules U and X Data).

Table 5.5 provides the RT_{PTS} calculation for Braidwood Unit 1 Beltline Region Materials at EOL (32 EFPY), (Reference 7).

Table 5.1

Braidwood Unit 1 Calculation of Chemistry Factors Using Surveillance Capsule Data[a]

Material	Capsule	Capsule f[b] (n/cm^2, E > 1.0 MeV)	FF[c]	ΔRT_{NDT}[b] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Lower Shell Forging (Tangential)	U	0.388×10^{19}	0.738	5.78	4.26	0.54
	X	1.17×10^{19}	1.044	38.23	39.91	1.09
	W	1.98×10^{19}	1.186	24.14	28.64	1.41
Lower Shell Forging (Axial)	U	0.388×10^{19}	0.738	0.0[d]	0.00	0.54
	X	1.17×10^{19}	1.044	28.75	30.01	1.09
	W	1.98×10^{19}	1.186	37.11	44.03	1.41
				SUM:	146.85	6.08
	$CF_{LS\ Forging} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (146.85) \div (6.08) = \mathbf{24.1°F}$					
Braidwood Unit 1 Surveillance Weld Material	U	0.388×10^{19}	0.738	17.06	12.59	0.54
	X	1.17×10^{19}	1.044	30.15	31.47	1.09
	W	1.98×10^{19}	1.186	49.68	58.94	1.41
Braidwood Unit 2 Surveillance Weld Material	U	0.388×10^{19}	0.738	0.0[d]	0.00	0.54
	X	1.15×10^{19}	1.039	26.3	27.33	1.08
	W	2.07×10^{19}	1.198	23.9	28.63	1.44
				SUM:	158.96	6.10
	$CF_{Weld\ Metal} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (158.96) \div (6.10) = \mathbf{26.1°F}$					

<u>Notes</u>:

(a) Source document is CN-AMLRS-10-7 (Reference 14), Table 5.2-1.

(b) f = fluence; ΔRT_{NDT} values are the measured 30 ft-lb shift values taken from Reference 6.

(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 1 Reactor Vessel Material Properties

Material Description	Cu (%)	Ni (%)	Chemistry Factor	Initial RT_{NDT} (°F)[a]
Closure Head Flange Heat # 5P7381/3P6406	0.11	0.67	--	-20
Vessel Flange Heat # 122N357V	--	0.77	--	-10
Nozzle Shell Forging * Heat # 5P-7016	0.04	0.73	26.0°F[b]	10
Intermediate Shell Forging * Heat # [49D383/49C344]-1-1	0.05	0.73	31.0°F[b]	-30
Lower Shell Forging * Heat # [49D867/49C813]-1-1	0.05	0.74	31.0°F[b] 24.1°F[c]	-20
Circumferential Weld * (Intermediate Shell to Lower Shell) WF-562 (HT# 442011)	0.03	0.67	41.0°F[b] 26.1°F[c]	40
Upper Circumferential Weld * (Nozzle Shell to Intermediate Shell) WF-645 (HT# H4498)	0.04	0.46	54.0°F[b]	-25

* 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.

Table 5.3

Summary of Braidwood Unit 1 Adjusted Reference Temperature (ART) Values at 1/4T and 3/4T Locations for 32 EFPY[(a)]

Reactor Vessel Material	Surface Fluence (n/cm2, E>1.0 MeV)	32 EFPY	
		1/4T ART (°F)	3/4T ART (°F)
Nozzle Shell Forging	0.586×10^{19}	47	34
Intermediate Shell Forging	1.76×10^{19}	33	15
Lower Shell Forging	1.76×10^{19}	43	25
→Using credible surveillance data	1.76×10^{19}	21	15
Nozzle to Intermediate Shell Forging Circ. Weld Seam (Heat # H4498)	0.586×10^{19}	52	25
Intermediate to Lower Shell Forging Circ. Weld Seam (Heat # 442011)	1.70×10^{19}	122	99
→Using credible surveillance data	1.70×10^{19}	93	78

Notes:

(a) The source document containing detailed calculations is CN-AMLRS-10-7 (Reference 14), Tables 5.3.1-1 and 5.3.1-2. The ART values summarized in this table utilize the most recent fluence projections and materials data, but were not used in development of the P/T limit curves. See Figures 2.1 and 2.2 of this PTLR for the ART values used in development of the P/T limit curves.

Table 5.4

Braidwood Unit 1 Calculation of Adjusted Reference Temperatures (ARTs) at 32 EFPY at the Limiting Reactor Vessel Material, Nozzle Shell Forging 5P-7016

Parameter	Values	
Operating Time	32 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]	3.65×10^{18}	1.32×10^{18}
Fluence Factor, FF	0.772	0.475
ΔRT_{NDT}= CFxFF(°F)	18.8	12.4
Initial RT_{NDT}, I(°F)	10	10
Margin, M (°F)	18.8	12.4
ART= I+(CF*FF)+M,°F per RG 1.99, Revision 2	48	35

(a) The Braidwood Unit 1 reactor vessel wall thickness is 8.5 inches at the beltline region.
(b) Fluence f, is based upon f_{surf} (E > 1.0 Mev) = 6.08×10^{18} at 32 EFPY (Reference 11).

Table 5.5

RT_{PTS} Calculation for Braidwood Unit 1 Beltline Region Materials at EOL (32 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)
Nozzle Shell Forging	1.1	26	0.586×10^{19}	0.8504	10	22.1	0	11.1	22.1	54
Intermediate Shell Forging	1.1	31	1.76×10^{19}	1.1554	-30	35.8	0	17	34	40
Lower Shell Forging	1.1	31	1.76×10^{19}	1.1554	-20	35.8	0	17	34	50
→Using credible surveillance data	2.1	24.1	1.76×10^{19}	1.1554	-20	27.8	0	8.5	17	25
Nozzle to Intermediate Shell Forging Circ. Weld Seam (Heat # H4498)	1.1	54	0.586×10^{19}	0.8504	-25	45.9	0	23.0	45.9	67
Intermediate to Lower Shell Forging Circ. Weld Seam (Heat # 442011)	1.1	41	1.70×10^{19}	1.1461	40	47.0	0	23.5	47.0	134
→Using credible surveillance data	2.1	26.1	1.70×10^{19}	1.1461	40	29.9	0	14	28	98

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 CN-AMLRS-10-7 (Reference 14), Table 5.5-1.
(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 with credible surveillance data $\sigma_\Delta = 8.5°F$ for Position 2.1; 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 $0.5*\Delta RT_{NDT}$.

6.0 References

1. WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," J.D. Andrachek, et al., January 1996.

2. NRC Letter from R. A. Capra to O.D. Kingsley, Commonwealth Edison Company, "Byron Station Units 1 and 2 and Braidwood Station Units 1 and 2, Acceptance for referring of pressure temperature limits report, (M98799, M98800, M98801, and M98802)," January 21 1998.

3. Westinghouse Letter to Exelon Nuclear, CAE-10-MUR-197, Revision 0, "Low Temperature Overpressure Protection (LTOP) System Evaluation Final Letter Report," M.P. Rudakewiz, September 8, 2010.

4. Byron & Braidwood Design Information Transmittal DIT-BRW-2006-0051, "Transmittal of Braidwood Unit 1 and Unit 2 Temperature and Pressure Uncertainties for Low Temperature Overpressure System (LTOPS) Power Operated Relief Valves (PORVS)," Nathan (Joe) Wolff Jr., July 18, 2006.

5. WCAP-9807, "Commonwealth Edison Company, Braidwood Station Unit 1 Reactor Vessel Radiation Surveillance Program," S.E. Yanichko, et al., February 1981.

6. WCAP-15316, Revision 1, "Analysis of Capsule W from Commonwealth Edison Company Braidwood Unit 1 Reactor Vessel Radiation Surveillance Program," E. Terek, et al., December 1999.

7. WCAP-15365, Revision 1, "Evaluation of Pressurized Thermal Shock for Braidwood Unit 1," J.H. Ledger, January 2002.

8. NRC Letter from G. F. Dick, Jr., NRR, to C. Crane, Exelon Generation Company, LLC, "Issuance of Amendments: Revised Pressure-Temperature Limits Methodology; Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2," dated October 4, 2004.

9. NRC Letter from M. Chawla to O.D. Kingsley, Exelon Generation Company, LLC, "Issuance of exemption from the Requirements of 10 CFR 50 Part 60 and Appendix G for Byron Station, Units 1 and 2, and Braidwood Stations, Units 1 and 2," dated August 8, 2001.

10. 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 - Issuance of Amendments Re: Reactor Coolant System Pressure and Temperature Limits Report (TAC Nos. MC8693, MC8694, MC8695, and MC8696)," November 27, 2006.

11. WCAP-15364, Revision 2, "Braidwood Unit 1 Heatup and Cooldown Limit Curves for Normal Operation," T.J. Laubham, November 2003.

12. WCAP-16143-P, Revision 0, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2," W. Bamford, et al., November 2003.

13. Westinghouse Letter to Exelon Nuclear, CCE-07-24, "Braidwood Unit 1 and 2 RCS HU/CD Limit Curve Table Values," dated February 15, 2007.

14. Westinghouse Calculation Note CN-AMLRS-10-7, Revision 0, "Braidwood Units 1 and 2 Measurement Uncertainty Recapture (MUR) Uprate: Reactor Vessel Integrity Evaluations," A.E. Leicht, September 2010.

ATTACHMENT 2

Braidwood Unit 2
Pressure and Temperature Limits Report (PTLR), Revision 6

BRAIDWOOD UNIT 2

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

Revision 6

BRAIDWOOD - UNIT 2
PRESSURE AND TEMPERATURE LIMITS REPORT

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BRAIDWOOD - UNIT 2
PRESSURE AND TEMPERATURE LIMITS REPORT

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-NP-A, Revision 2 (Reference 1) was used with the following exception:

a) Optional use of ASME Code Section XI, Appendix G, Article G-2000, 1996 Addenda,
b) Use of ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves, Section XI, Division 1", and
c) Use of ASME Code Case N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessel, Section XI, Division 1", and
d) Elimination of the flange requirements documented in WCAP-16143-P.

This exception to the methodology in WCAP 14040-NP-A, Revision 2 has been reviewed and accepted by the NRC in References 2, 7, 9, and 10.

WCAP 15373, Revision 2 (Reference 11), 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. WCAP-16143-P, Reference 12, documents the technical basis for the elimination of the flange requirements.

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 Reference 11 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 defined in WCAP-15373, Revision 2 (Reference 11). 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, Article G2000, 1996 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: Circumferential Weld WF-562 & Nozzle Shell Forging	
Limiting ART Values at 32 EFPY	1/4T 93°F (N-588) & 67°F ('96 App. G)
	3/4T 79°F (N-588) & 54°F ('96 App. G)

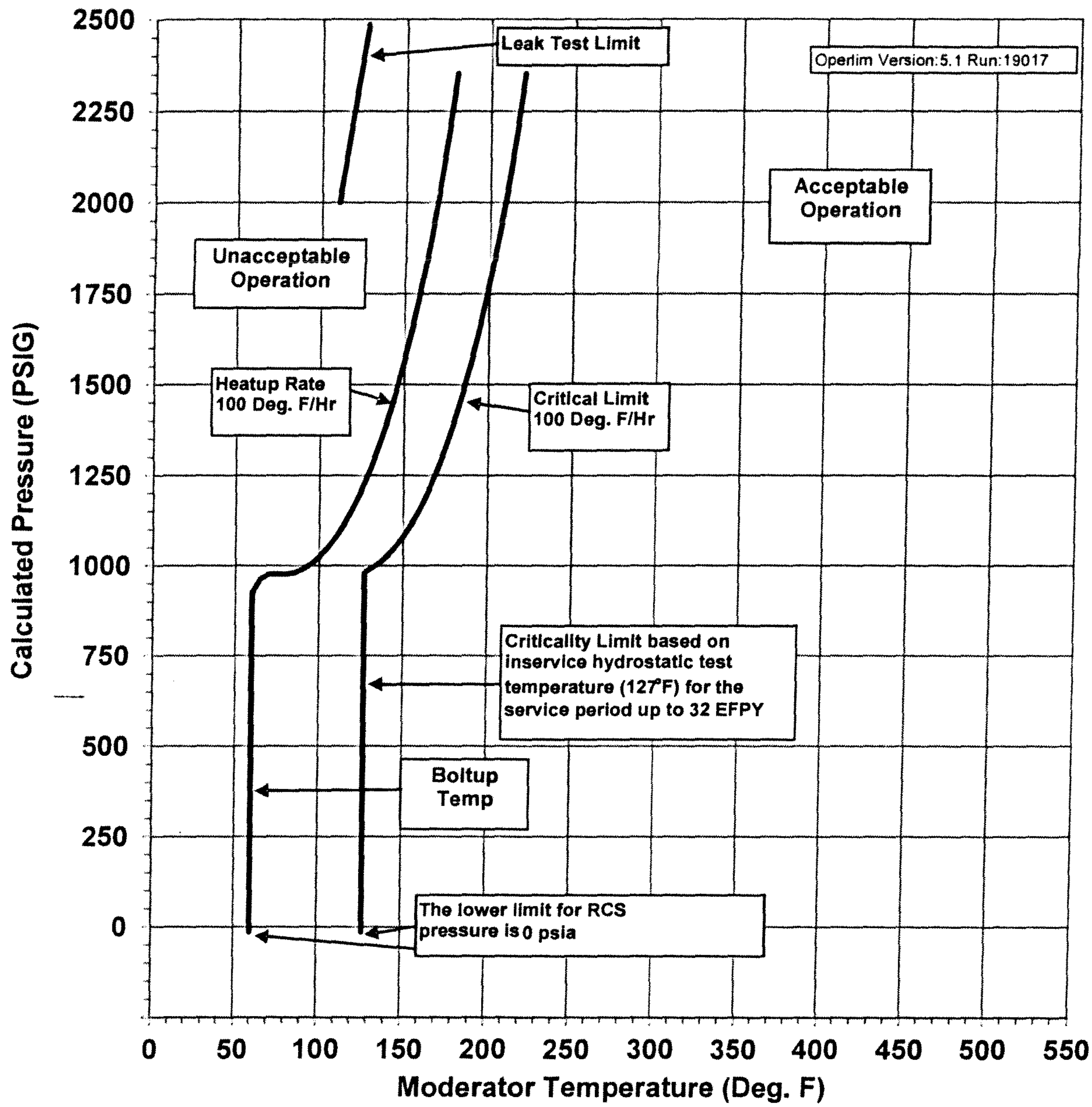

Figure 2.1

Braidwood Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable to 32 EFPY (Without Margins for Instrumentation Errors)

Material Property Basis Limiting Material: Circumferential Weld WF-562 & Nozzle Shell Forging Limiting ART Values at 32 EFPY 1/4T 93°F (N-588) & 67°F ('96 App. G) 3/4T 79°F (N-588) & 54°F ('96 App. G)

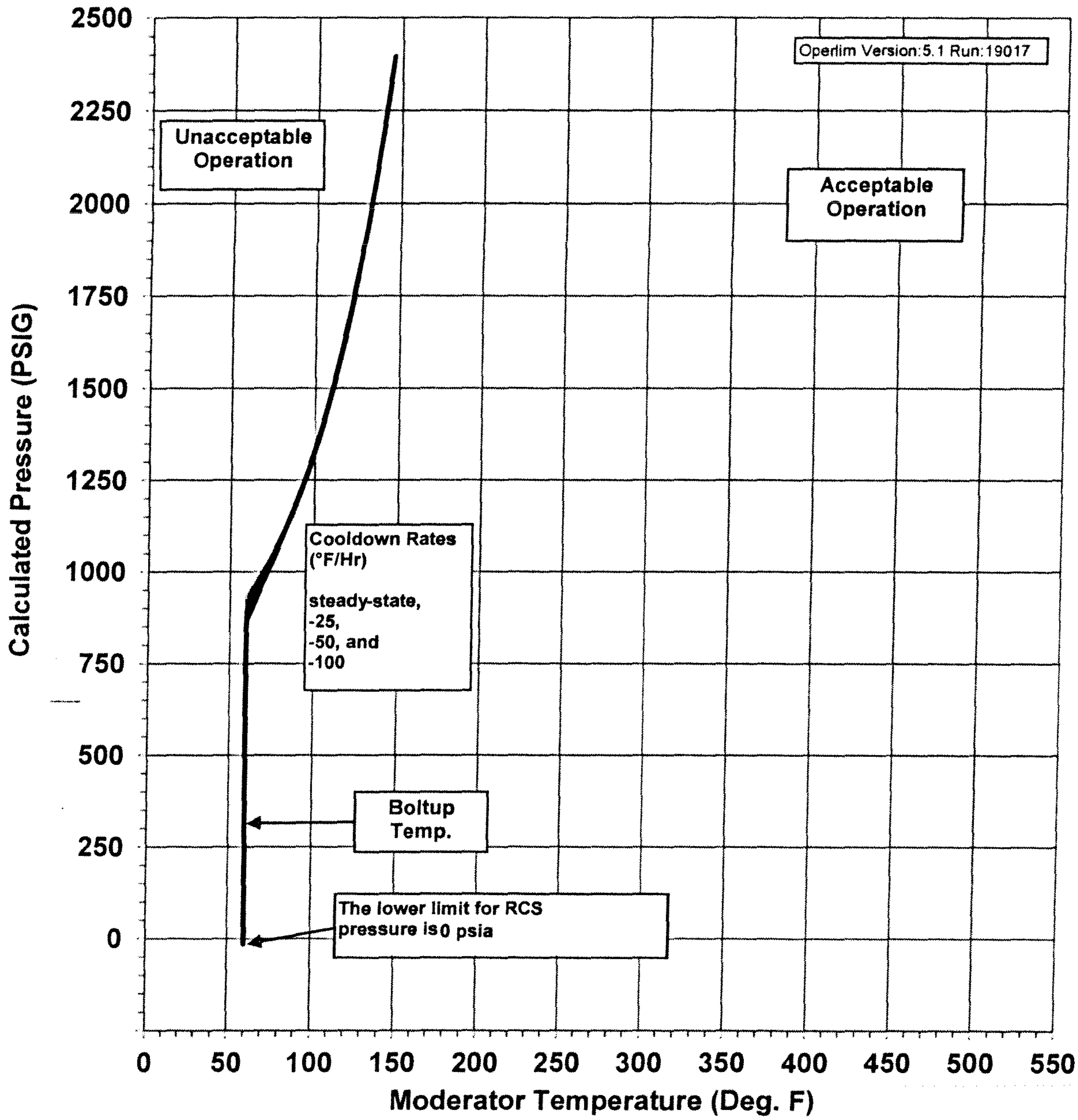

Figure 2.2

Braidwood Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50 and 100°F/hr) Applicable to 32 EFPY (Without Margins of Instrumentation Errors)

Table 2.1a
Braidwood Unit 2 Heatup Data Points at 32 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	127	Note 1	110	2000
60	924	127	965	127	2485
65	965	127	977*		
70	977	127	977		
75	977	127	981		
80	977	130	990		
85	981	135	1005		
90	990	140	1025		
95	1005	145	1051		
100	1025	150	1081		
105	1051	155	1118		
110	1081	160	1161		
115	1118	165	1210		
120	1161	170	1266		
125	1210	175	1329		
130	1266	180	1400		
135	1329	185	1480		
140	1400	190	1569		
145	1480	195	1668		
150	1569	200	1778		
155	1668	205	1901		
160	1778	210	2036		
165	1901	215	2186		
170	2036	220	2353		
175	2186				
180	2353				

* Refer to Reference 13

Note 1: The Minimum acceptable pressure is 0 psia

Table 2.1b
Braidwood Unit 2 Cooldown Data at 32 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	931	60	908	60	889	60	866
65	965	65	946	65	932	65	921
70	1003	70	989	70	980	70	980
75	1045	75	1036	75	1033	75	1033
80	1092	80	1088	80	1088	80	1088
85	1143	85	1143	85	1143	85	1143
90	1200	90	1200	90	1200	90	1200
95	1263	95	1263	95	1263	95	1263
100	1332	100	1332	100	1332	100	1332
105	1409	105	1409	105	1409	105	1409
110	1494	110	1494	110	1494	110	1494
115	1587	115	1587	115	1587	115	1587
120	1691	120	1691	120	1691	120	1691
125	1805	125	1805	125	1805	125	1805
130	1932	130	1932	130	1932	130	1932
135	2071	135	2071	135	2071	135	2071
140	2226	140	2226	140	2226	140	2226
145	2396	145	2396	145	2396	145	2396

Note 1: The Minimum acceptable pressure is 0 psia

3.0 Low Temperature Overpressure Protection and Boltup

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

3.1 LTOP System Setpoints (LCO 3.4.12).

The power operated relief valves (PORVs) shall each have nominal lift settings in accordance with Figure 3.1 and Table 3.1. These limits are based on References 3 and 8.

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 of 350°F and below and disarming of LTOP for RCS temperature above 350°F.

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.

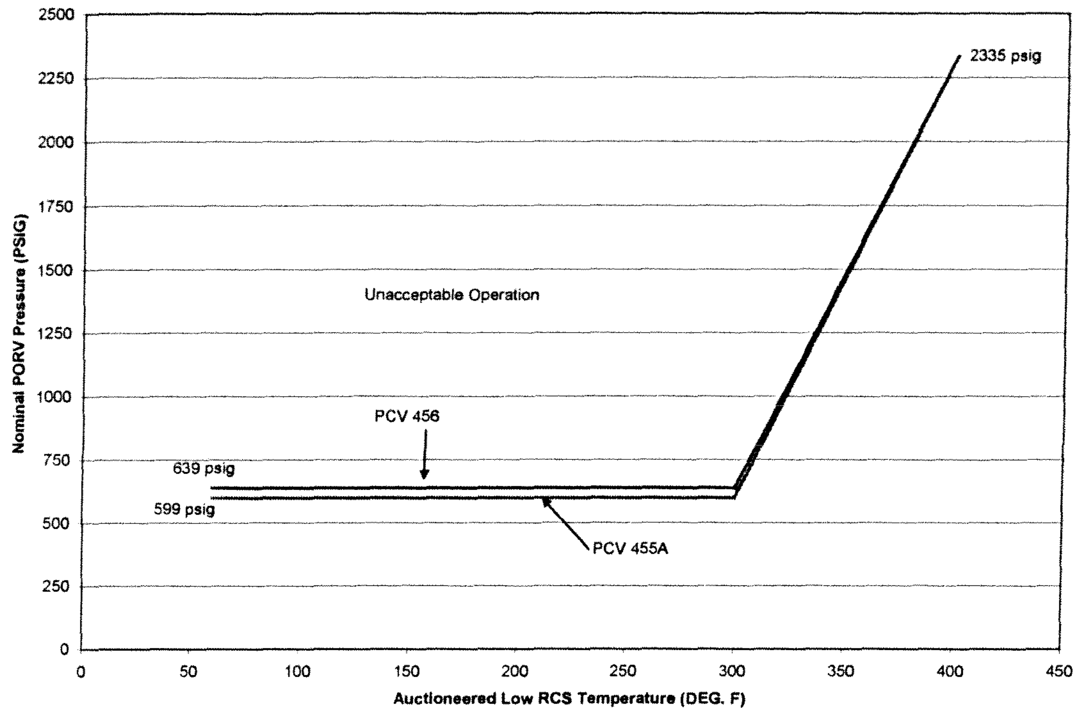

Figure 3.1
Braidwood Unit 2 Nominal PORV Setpoints for the Low Temperature Overpressure Protection (LTOP) System Applicable for 32 EFPY (Includes Instrumentation Uncertainty)

Table 3.1
Data Points for Braidwood Unit 2 Nominal PORV Setpoints for the LTOP System Applicable for 32 EFPY (Includes Instrumentation Uncertainty)

PCV-455A

RCS TEMP. (DEG. F)	RCS Pressure (PSIG)
60	599
300	599
400	2335

PCV-456

RCS TEMP. (DEG. F)	RCS Pressure (PSIG)
60	639
300	639
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 third and final reactor vessel material irradiation surveillance specimens (Capsule W) have been removed and analyzed to determine changes in material properties. The surveillance capsule testing has been completed for the original operating period. The remaining three capsules, V, 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	0.388×10^{19}
X	238.5°	4.03	4.24	1.15×10^{19}
W	121.5°	4.06	8.56	2.07×10^{19}
Z[c]	301.5°	4.14	12.78 (EOC 10)	2.83×10^{19}
V[c]	61.0°	3.92	18.42 (EOC 14)	3.73×10^{19}
Y[c]	241.0°	3.89	12.78 (EOC 10)	2.66×10^{19}

<u>Notes</u>:

(a) Source document is CN-AMLRS-10-7 (Reference 14), Table 5.7-4.
(b) Effective Full Power Years (EFPY) from plant startup.
(c) Standby Capsules Z, V, and Y were removed and placed in the spent fuel pool. No testing or analysis has been performed on these capsules. If license renewal is sought, one of these standby capsules may need to be tested to determine the effect of neutron irradiation on the reactor vessel surveillance materials during the period of extended operation.

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 32 EFPY.

Table 5.4 shows the calculation of ARTs at 32 EFPY for the limiting Braidwood Unit 2 reactor vessel material.

Table 5.5 provides the RT_{PTS} Calculation for Braidwood Unit 2 Beltline Region Materials at EOL (32 EFPY), (Reference 6).

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)	$FF*\Delta RT_{NDT}$ (°F)	FF^2
Lower Shell Forging (Tangential)	U	0.388×10^{19}	0.738	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	4.53	5.43	1.44
Lower Shell Forging (Axial)	U	0.388×10^{19}	0.738	0.0[d]	0.00	0.54
	X	1.15×10^{19}	1.039	33.94	35.26	1.08
	W	2.07×10^{19}	1.198	33.2	39.78	1.44
				SUM:	80.47	6.12
	$CF_{LS\ Forging} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (80.47) \div (6.12) = \mathbf{13.2°F}$					
Braidwood Unit 1 Surveillance Weld Material	U	0.388×10^{19}	0.738	17.06	12.59	0.54
	X	1.17×10^{19}	1.044	30.15	31.47	1.09
	W	1.98×10^{19}	1.186	49.68	58.94	1.41
Braidwood Unit 2 Surveillance Weld Material	U	0.388×10^{19}	0.738	0.0[d]	0.00	0.54
	X	1.15×10^{19}	1.039	26.3	27.33	1.08
	W	2.07×10^{19}	1.198	23.9	28.63	1.44
				SUM:	158.96	6.10
	$CF_{Weld\ Metal} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (158.96) \div (6.10) = \mathbf{26.1°F}$					

Notes:

(a) Source document is CN-AMLRS-10-7 (Reference 14), Table 5.2-2.
(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
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] 13.2°F[c]	-30
Circumferential Weld * (Intermediate Shell to Lower Shell) Weld Seam WF-562 Heat # 442011	0.03	0.67	41.0F[b] 26.1F[c]	40
Circumferential Weld * (Nozzle Shell to Intermediate Shell) Weld Seam WF-645 Heat # H4498	0.04	0.46	54.0°F[b]	-25

* 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

Table 5.3

Summary of Braidwood Unit 2 Adjusted Reference Temperature (ART) Values at 1/4T and 3/4T Locations for 32 EFPY[(a)]

Reactor Vessel Material	Surface Fluence (n/cm^2, E>1.0 MeV)	32 EFPY	
		1/4T ART (°F)	3/4T ART (°F)
Nozzle Shell Forging	0.559×10^{19}	66	54
Intermediate Shell Forging	1.73×10^{19}	10	-1
Lower Shell Forging	1.73×10^{19}	41	24
→Using non-credible surveillance data	1.73×10^{19}	-3	-11
Nozzle to Intermediate Shell Forging Circ. Weld Seam (Heat # H4498)	0.559×10^{19}	51	24
Intermediate to Lower Shell Forging Circ. Weld Seam (Heat # 442011)	1.67×10^{19}	122	99
→Using credible surveillance data	1.67×10^{19}	92	78

Notes:

(a) The source document containing detailed calculations is CN-AMLRS-10-7 (Reference 14), Tables 5.3.1-3 and 5.3.1-4. The ART values summarized in this table utilize the most recent fluence projections and materials data, but were not used in development of the P/T limit curves. See Figures 2.1 and 2.2 of this PTLR for the ART values used in development of the P/T limit curves.

Table 5.4

Braidwood Unit 2 Calculation of Adjusted Reference Temperatures (ARTs) at 32 EFPY at the Limiting Reactor Vessel Material, Nozzle Shell Forging 5P-7056

Parameter	Values	
Operating Time	32 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]	3.40×10^{18}	1.23×10^{18}
Fluence Factor, FF	0.703	0.460
ΔRT_{NDT}= CFxFF(°F)	18.3	12.0
Initial RT $_{NDT,}$, I(°F)	30	30
Margin, M(°F)	18.3	12.0
ART= I+(CF*FF)+M, °F per RG 1.99, Revision 2	67	54

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) =5.67×10^{18} n/cm^2 at 32 EFPY (Reference 11).

Table 5.5

RT_{PTS} Calculation for Braidwood Unit 2 Beltline Region Materials at EOL (32 EFPY)[a,b]

Reactor Vessel Material	R.G. 1.99, Rev. 2 Position	CF (°F)	Fluence (n/cm^2, 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)
Nozzle Shell Forging	1.1	26	0.559×10^{19}	0.8373	30	21.8	0	10.9	21.8	74
Intermediate Shell Forging	1.1	20	1.73×10^{19}	1.1508	-30	23.0	0	11.5	23.0	16
Lower Shell Forging	1.1	37	1.73×10^{19}	1.1508	-30	42.6	0	17	34	47
→Using non-credible surveillance data	2.1	13.2	1.73×10^{19}	1.1508	-30	15.2	0	7.6	15.2	0
Nozzle to Intermediate Shell Forging Circ. Weld Seam (Heat # H4498)	1.1	54	0.559×10^{19}	0.8373	-25	45.2	0	22.6	45.2	65
Intermediate to Lower Shell Forging Circ. Weld Seam (Heat # 442011)	1.1	41	1.67×10^{19}	1.1413	40	46.8	0	23.4	46.8	134
→Using credible surveillance data	2.1	26.1	1.67×10^{19}	1.1413	40	29.8	0	14	28	98

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 CN-AMLRS-10-7 (Reference 14), Table 5.5-2.
(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 $0.5*\Delta RT_{NDT}$.

6.0 References

1. WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", J.D. Andrachek, et al., January 1996.

2. NRC Letter from R. A. Capra to O.D. Kingsley, Commonwealth Edison Company, "Byron Station Units 1 and 2 and Braidwood Station Units 1 and 2, Acceptance for referring of pressure temperature limits report, (M98799, M98800, M98801, and M98802)," January 21, 1998.

3. Westinghouse Letter to Exelon Nuclear, CAE-10-MUR-197, Revision 0, "Low Temperature Overpressure Protection (LTOP) System Evaluation Final Letter Report," M.P. Rudakewiz, September 8, 2010.

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

5. WCAP-15369, "Analysis of Capsule W from the Commonwealth Edison Company Braidwood Unit 2 Reactor Vessel Radiation Surveillance Program," March 2000.

6. WCAP-15381, "Evaluation of Pressurized Thermal Shock for Braidwood Unit 2", T.J. Laubham, September 2000.

7. NRC Letter from G. F. Dick, Jr., NRR, to C. Crane, Exelon Generation Company, LLC, "Issuance of Amendments: Revised Pressure-Temperature Limits Methodology; Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2," dated October 4, 2004.

8. Byron & Braidwood Design Information Transmittal DIT-BRW-2006-0051, "Transmittal of Braidwood Unit 1 and Unit 2 Temperature and Pressure Uncertainties for Low Temperature Overpressure System (LTOPS) Power Operated Relief Valves (PORVS)," Nathan (Joe) Wolff Jr., July 18, 2006.

9. NRC Letter from M. Chawla to O.D. Kingsley, Exelon Generation Company, LLC, "Issuance of exemption from the Requirements of 10 CFR 50 Part 60 and Appendix G for Byron Station, Units 1 and 2, and Braidwood Stations, Units 1 and 2," dated August 8, 2001.

10. 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 - Issuance of Amendments Re: Reactor Coolant System Pressure and Temperature Limits Report (TAC Nos. MC8693, MC8694, MC8695, and MC8696)," November 27, 2006.

11. WCAP-15373, Revision 2, "Braidwood Unit 2 Heatup and Cooldown Limits for Normal Operation," T.J. Laubham et al., November 2003.

12. WCAP-16143-P, Revision 0, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2," W. Bamford, et al., November 2003.

13. Westinghouse Letter to Exelon Nuclear, CCE-07-24, "Braidwood Unit 1 and 2 RCS HU/CD Limit Curve Table Values," dated February 15, 2007.

14. Westinghouse Calculation Note CN-AMLRS-10-7, Revision 0, "Braidwood Units 1 and 2 Measurement Uncertainty Recapture (MUR) Uprate: Reactor Vessel Integrity Evaluations," A.E. Leicht, September 2010, and Westinghouse evaluation MCOE-LTR-13-102 Rev. 0, "Byron and Braidwood Closure Head/Vessel Flange Region: MUR Uprate Assessment," November 2013.