Exelon Generation.

Byron Generating Station
4450 North German Church Road
Byron, IL 61010-9794
www.exeloncorp.com

March 27, 2014

LTR: BYRON 2014-0040
File: 2.01.0300
1.10.0101

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Byron Station, Units 1 and 2
Facility Operating License Nos, NPF-37 and NPF-66
NRC Docket No, STN 50-454 and 50-455

Subject: Pressure and Temperature Limits Report (PTLR) Revised for Negative Pressure Application during Reactor Coolant System Vacuum Fill
Byron Station, Units 1 and 2

In accordance with Technical Specification 5.6.6, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," we are submitting the March 2014 revisions to the Byron Station Units 1 and 2 PTLR documents. The PTLRs are revised to change the lowest pressure value for Figure 2.1, "Reactor Coolant System Heatup Limitations" and Figure 2.2, "Reactor Coolant System Cooldown Limitations" from 0 psig to minus(-)14 psig, which is applicable during vacuum fill of the Reactor Coolant System.

In addition, minor editorial corrections were made to the Unit 1 PTLR text to document the correct supporting references.

Should you have any questions concerning these reports, please contact Steven Gackstetter, Regulatory Assurance Manager, at (815) 406-2800.

Respectfully,

Faber A. Kearney
Site Vice President
Byron Nuclear Generating Station

FAK/GC/sg

Attachments: 1. Byron Unit 1 Pressure and Temperature Limits Report, March 2014
2. Byron Unit 2 Pressure and Temperature Limits Report, March 2014

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Byron Station

BYRON UNIT 1

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

(March 2014)

BYRON - UNIT 1
PRESSURE AND TEMPERATURE LIMITS REPORT

Table of Contents

List of Figures

List of Tables

BYRON - UNIT 1
PRESSURE AND TEMPERATURE LIMITS REPORT

1.0 Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)

This Pressure and Temperature Limits Report (PTLR) for Byron Unit 1 has been prepared in accordance with the requirements of Byron TS 5.6.6 (RCS Pressure and Temperature Limits Report). Revisions to the PTLR shall be provided to the NRC after issuance.

The Technical Specifications addressed in this report are listed below:

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

2.0 RCS Pressure and Temperature Limits

This section provides the Byron Unit 1 Heatup and Cooldown Limitations.

The PTLR limits for Byron 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 6, 10, 11 and 12.

WCAP-15391, Revision 1, Reference 7, provides the basis for the Byron Unit 1 P/T curves, along with the best estimate chemical compositions, fluence projections, and adjusted reference temperatures used to determine these limits. The weld metal data integration for Byron and Braidwood Units 1 and 2 is documented in Reference 2. WCAP-16143-P, Reference 11, 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 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 defined in WCAP-15391, Rev. 1 (Reference 7). Consistent with the methodology described in Reference 1, 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 G-2000, 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: INTERMEDIATE SHELL FORGING
LIMITING ART VALUES AT 32 EFPY: 1/4T, 106°F
3/4T, 97°F

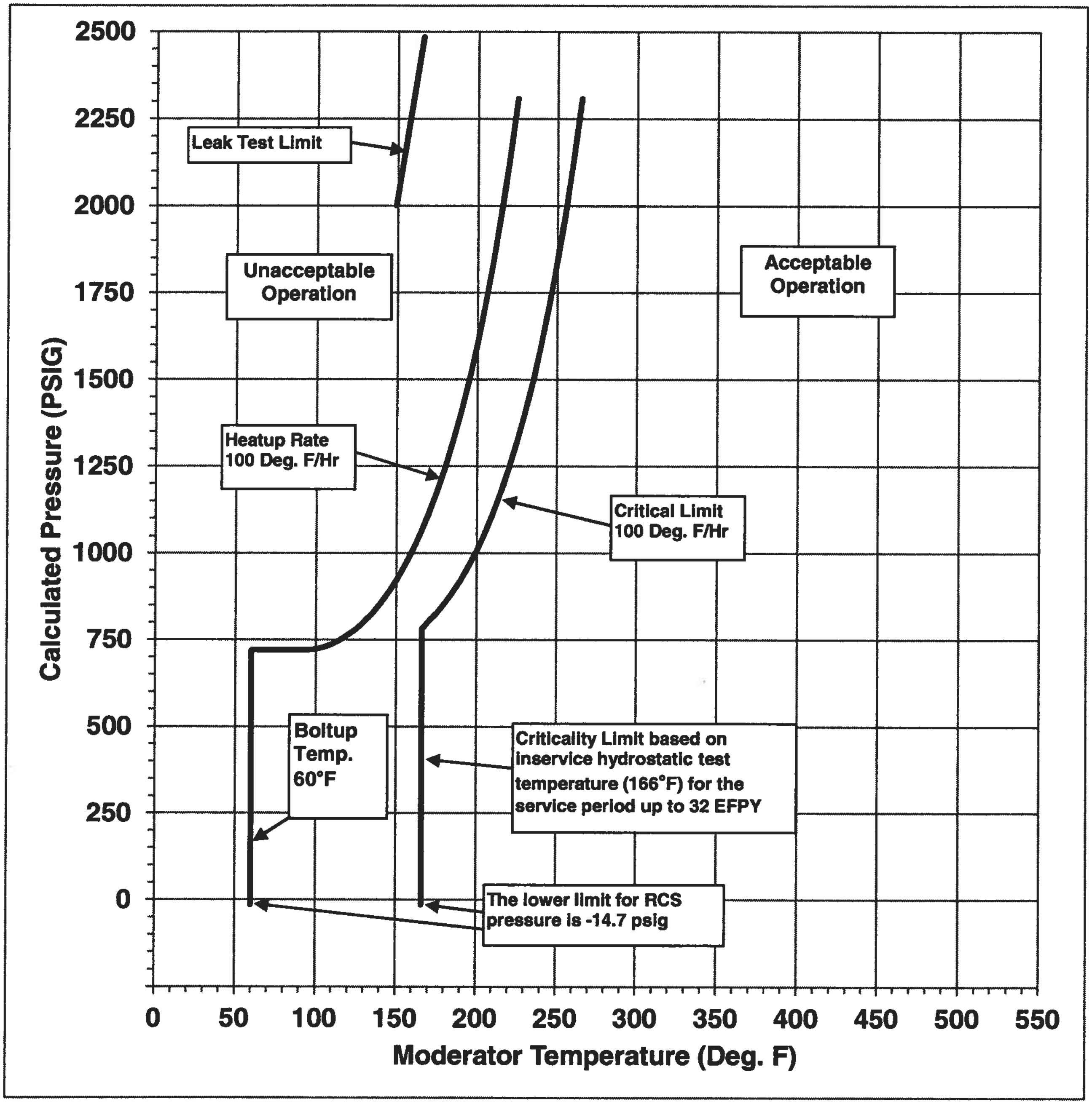

Figure 2.1
Byron Unit 1 Reactor Coolant System Heatup Limitations (Heatup rates of 100°F/hr) Applicable for 32 EFPY (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE SHELL FORGING
LIMITING ART VALUES AT 32 EFPY: 1/4T, 106°F
3/4T, 97°F

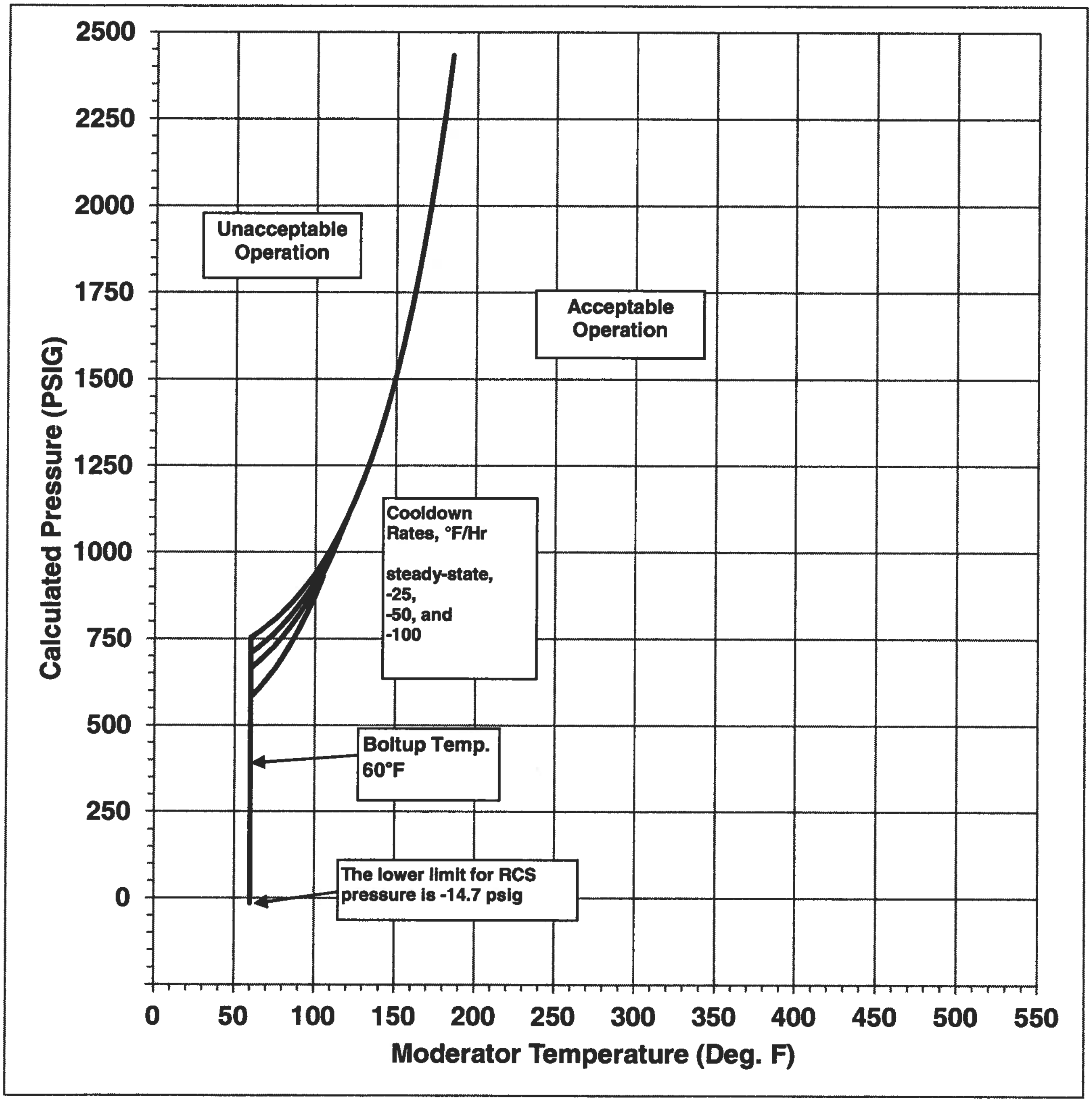

Figure 2.2
Byron Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown rates of 0, 25, 50 and 100°F/hr) Applicable for 32 EFPY (Without Margins for Instrumentation Errors)

Table 2.1a
Byron 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	-14.7	166	-14.7	149	2000
60	720	166	720	166	2485
65	720	166	720		
70	720	166	720		
75	720	166	720		
80	720	166	720		
85	720	166	720		
90	720	166	720		
95	720	166	723		
100	723	166	729		
105	729	166	737		
110	737	166	749		
115	749	166	764		
120	764	166	781		
125	781	170	802		
130	802	175	826		
135	826	180	854		
140	854	185	886		
145	886	190	921		
150	921	195	962		
155	962	200	1007		
160	1007	205	1057		
165	1057	210	1113		
170	1113	215	1175		
175	1175	220	1244		
180	1244	225	1321		
185	1321	230	1406		
190	1406	235	1499		
195	1499	240	1603		
200	1603	245	1718		
205	1718	250	1844		
210	1844	255	1984		
215	1984	260	2138		
220	2138	265	2308		
225	2308				

Table 2.1b
Byron 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	-14.7	60	-14.7	60	-14.7	60	-14.7
60	753	60	709	60	665	60	581
65	769	65	726	65	685	65	606
70	787	70	746	70	706	70	633
75	806	75	767	75	730	75	663
80	827	80	791	80	757	80	697
85	851	85	817	85	786	85	735
90	877	90	846	90	819	90	777
95	906	95	879	95	855	95	823
100	937	100	914	100	895	100	874
105	973	105	954	105	940	105	931
110	1011	110	997	110	989		
115	1054	115	1045	115	1043		
120	1102	120	1099				
125	1154						
130	1212						
135	1276						
140	1347						
145	1425						
150	1512						
155	1607						
160	1713						
165	1829						
170	1958						
175	2101						
180	2258						
185	2433						

Note: For each cooldown rate, the steady-state pressure values shall govern the temperature where no allowable pressure values are provided.

3.0 Low Temperature Overpressure Protection and Boltup

This section provides the Byron 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 5.

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

The required enable temperature for the PORVs shall be ≤ 350°F RCS temperature. (Byron 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 (Reference 7).

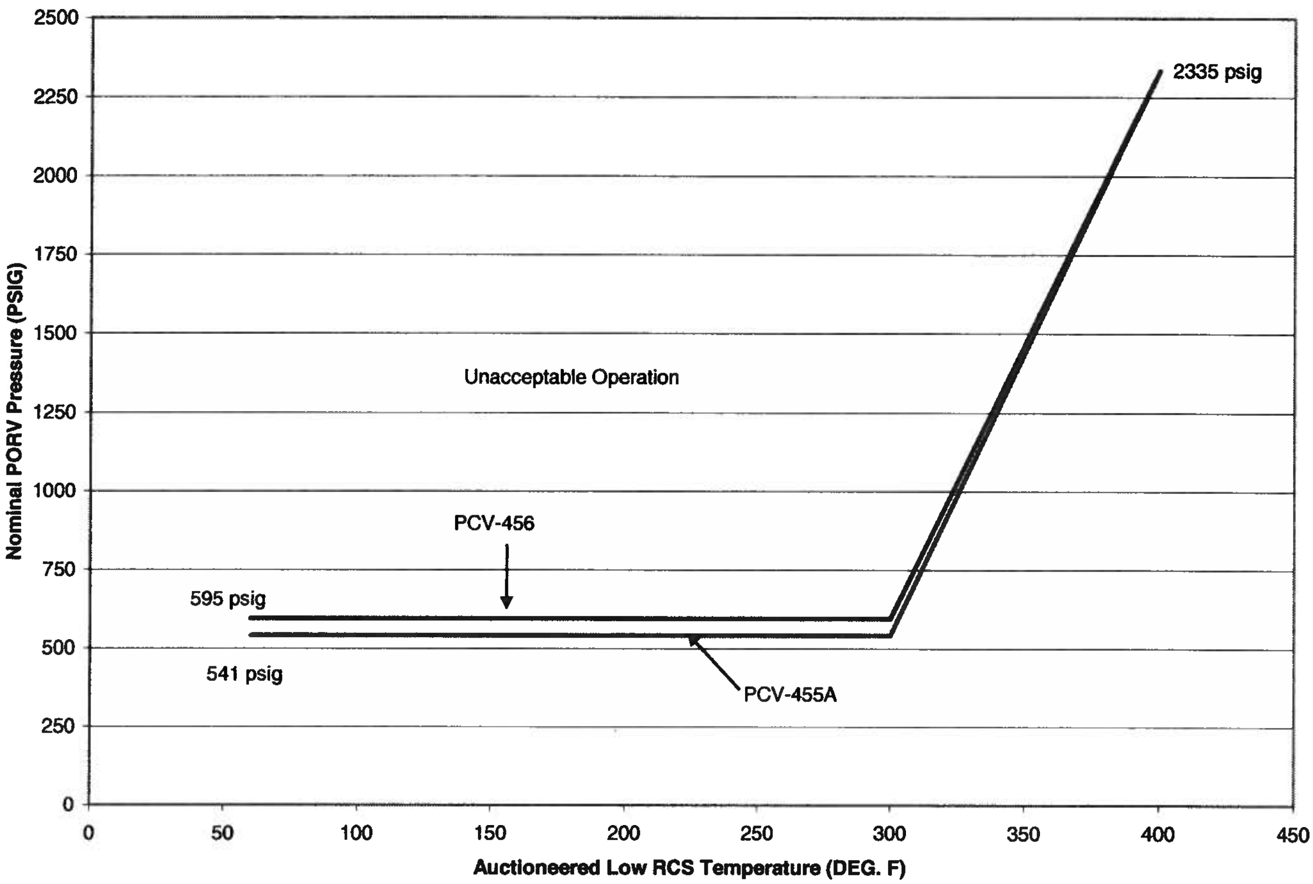

Figure 3.1
Byron Unit 1 Nominal PORV Setpoints for the Low Temperature Overpressure Protection (LTOP) System Applicable for 32 EFPY (Includes Instrumentation Uncertainty)

Table 3.1
Data Points for Byron 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 12) 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 the reactor vessel 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

Byron Unit 1 Surveillance Capsule Withdrawal Summary[a]

Capsule	Capsule Location	Lead Factor	Withdrawal EFPY[b]	Fluence (n/cm^2, E > 1.0 MeV)
U	58.5°	4.05	1.18	0.409×10^{19}
X	238.5°	4.09	5.67	1.49×10^{19}
W	121.5°	4.08	9.27	2.26×10^{19}
Z[c]	301.5°	4.11	14.59 (EOC 12)	3.34×10^{19}
V[c]	61.0°	3.89	14.59 (EOC 12)	3.16×10^{19}
Y[c]	241.0°	3.85	18.81 (EOC 15)	3.97×10^{19}

Notes:

(a) Source document is CN-AMLRS-10-8 (Reference 4), 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 Byron Unit 1 adjusted reference temperature (ART) values at the 1/4T and 3/4T locations for 32 EFPY.

Table 5.4 provides the RT_{PTS} values for Byron Unit 1 for 32 EFPY obtained from Reference 4.

Table 5.1

Byron 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
Intermediate Shell Forging (Tangential)	U	4.09 x 10^{19}	0.752	28.55	21.47	0.57
	X	1.49 x 10^{19}	1.110	9.82	10.90	1.23
	W	2.26 x 10^{19}	1.221	49.20	60.06	1.49
Intermediate Shell Forging (Axial)	U	0.409 x 10^{19}	0.752	18.52	13.93	0.57
	X	1.49 x 10^{19}	1.110	53.03	58.89	1.23
	W	2.26 x 10^{19}	1.221	29.34	35.82	1.49
				Sum:	201.06	6.58
	$CF_{IS\ Forging} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (201.06) \div (6.58) =$ **30.6°F**					
Byron Unit 1 Surveillance Weld Material (Heat #442002)	U	4.09 x 10^{19}	0.752	11.22 (5.61)	8.44	0.57
	X	1.49 x 10^{19}	1.110	80.22 (40.11)	89.08	1.23
	W	2.26 x 10^{19}	1.221	102.68 (51.34)	125.34	1.49
Byron Unit 2 Surveillance Weld Material (Heat #442002)	U	0.406 x 10^{19}	0.750	16.88 (8.44)	12.66	0.56
	W	1.20 x 10^{19}	1.051	57.76 (28.88)	60.70	1.10
	X	2.18 x 10^{19}	1.211	108.02 (54.01)	130.86	1.47
				SUM:	427.08	6.42
	$CF_{Weld\ Metal} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (427.08) \div (6.42) =$ **66.5°F**					

Notes:

a) Source document is CN-AMLRS-10-8 (Reference 4), Table 5.2-1.

b) f = fluence; ΔRT_{NDT} values are the measured 30 ft-lb shift values taken from Reference 13. ΔRT_{NDT} values for the surveillance weld data are adjusted by a ratio of 2.0 (pre-adjusted values are listed in parentheses).

c) FF = fluence factor = $f^{(0.28 - 0.10*\log f)}$.

Table 5.2

Byron Unit 1 Reactor Vessel Material Properties [a]

Material Description	Cu (%)	Ni (%)	Initial RT_{NDT} (°F)[b]
Closure Head Flange 124K358VA1	--	0.74	60
Vessel Flange 123J219VA1	--	0.73	10
Nozzle Shell Forging 123J218	0.05	0.72	30
Intermediate Shell Forging 5P-5933	0.04	0.74	40
Lower Shell Forging 5P-5951	0.04	0.64	10
Intermediate to Lower Shell Forging Circ. Weld Seam WF-336 (Heat # 442002)	0.04	0.63	-30
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-501 (Heat # 442011)	0.03	0.67	10
Byron Unit 1 Surveillance Program Weld Metal (Heat # 442002)	0.02	0.69	--
Byron Unit 2 Surveillance Program Weld Metal (Heat # 442002)	0.02	0.71	--
Braidwood Units 1 & 2 Surveillance Program Weld Metals (Heat # 442011)	0.03	0.67, 0.71	--

a) Reference 7.

b) The initial RT_{NDT} values for the plates and welds are based on measured data.

Table 5.3

Summary of Byron Unit 1 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	0.598×10^{19}	74	59
Intermediate Shell Forging	1.77×10^{19}	93	78
→ Using non-credible surveillance data	1.77×10^{19}	**102**	**85**
Lower Shell Forging	1.77×10^{19}	63	48
Nozzle to Intermediate Shell Forging Circ. Weld Seam (Heat # 442011)	0.598×10^{19}	69	49
→ Using credible Braidwood Units 1 and 2 surveillance data	0.598×10^{19}	47	35
Intermediate to Lower Shell Forging Circ. Weld Seam (Heat # 442002)	1.72×10^{19}	79	49
→ Using credible surveillance data	1.72×10^{19}	**65**	**46**

Note:

(a) The source document containing detailed calculations is CN-AMLRS-10-8 (Reference 4), 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

RT_{PTS} Calculation for Byron 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/cm^2, E > 1.0 MeV)	FF	IRT_{NTD} [C] (°F)	ΔRT_{NTD} (°F)	σ_u [c] (°F)	σ_Δ [d] (°F)	Margin	RT_{PTS} (°F)
Nozzle Shell Forging	1.1	31	0.598×10^{19}	0.8560	30	26.5	0	13.3	26.5	83
Intermediate Shell Forging	1.1	26	1.77×10^{19}	1.1569	40	30.1	0	15.0	30.1	100
→ Using non-credible surveillance data	2.1	30.6	1.77×10^{19}	1.1569	40	35.4	0	17	34	**109**
Lower Shell Forging	1.1	26	1.77×10^{19}	1.1569	10	30.1	0	15.0	30.1	70
Nozzle to Intermediate Shell Forging Circ. Weld Seam (Heat # 442011)	1.1	41	0.598×10^{19}	0.8560	10	35.1	0	17.5	35.1	80
→ Using credible Braidwood Units 1 and 2 surveillance data	2.1	26.1	0.598×10^{19}	0.8560	10	22.3	0	11.2	22.3	55
Intermediate to Lower shell Forging Circ Weld Seam (Heat # 442002)	1.1	54	1.72×10^{19}	1.1492	-30	62.1	0	28	56	88
→ Using credible surveillance data	2.1	66.5	1.72×10^{19}	1.1492	-30	76.4	0	14	28	74

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-10-8 (Reference 4), 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 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 to exceed $0.5 * \Delta RT_{NTD}$

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. WCAP-14824, Revision 2, "Byron Unit 1 Heatup and Cooldown Limit Curves for Normal Operation and Surveillance Weld Metal Integration for Byron & Braidwood", November 1997 with Westinghouse errata letters CAE-97-220, dated November 26, 1997 and CAE-97-231/CCE-97-314 and CAE-97-233/CCE-97-316, dated January 6, 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. Westinghouse Calculation Note CN-AMLRS-10-8, Revision 0, "Byron Units 1 and 2 Measurement Uncertainty Recapture (MUR) Uprate: Reactor Vessel Integrity Evaluations," A. E. Leicht, September 2010.

5. Byron Station Design Information Transmittal DIT-BYR-06-046, "Transmittal of Byron Unit 1 and Unit 2 Temperature and Pressure Uncertainties for Low Temperature Overpressure System (LTOPS) Power Operated Relief Valves (PORVS)," David Neidich, August 15, 2006.

6. NRC Letter from R. A. Capra, NRR, to O. D. Kingsley, Commonwealth Edison Co., "Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, Acceptance for Referencing of Pressure Temperature Limits Report (TAC Numbers M98799, M98800, M98801, and M98802)," January 21, 1998.

7. WCAP- 15391, Revision 1, "Byron Unit 1 Heatup and Cooldown Limit Curves for Normal Operation," T. J. Laubham, et al., November 2003.

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-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.

12. WCAP-9517, "Commonwealth Edison Company, Byron Station Unit 1 Reactor Vessel Surveillance Program", J.A. Davidson, July 1979.

13. WCAP-15123, Revision 1, "Analysis of Capsule W from Commonwealth Edison Company Byron Unit 1 Reactor Vessel Radiation Surveillance Program," T.J. Laubham, et al, January 1999.

BYRON UNIT 2

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

(March 2014)

BYRON - UNIT 2
PRESSURE AND TEMPERATURE LIMITS REPORT

Table of Contents

List of Figures

List of Tables

BYRON - UNIT 2
PRESSURE AND TEMPERATURE LIMITS REPORT

1.0 Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)

This Pressure and Temperature Limits Report (PTLR) for Byron Unit 2 has been prepared in accordance with the requirements of Byron TS-5.6.6 (RCS Pressure and Temperature Limits Report). Revisions to the PTLR shall be provided to the NRC after issuance.

The Technical Specifications addressed in this report are listed below:

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

2.0 RCS Pressure and Temperature Limits

This section provides the Byron Unit 2 Heatup and Cooldown Limitations.

The PTLR limits for Byron 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) 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 Vessels, 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 8, 10, 11 and 12.

WCAP-15392, Revision 2 (Reference 7), provides the basis for the Byron Unit 2 P/T curves, along with the best estimate chemical compositions, fluence projections, and adjusted reference temperatures used to determine these limits. The weld metal data integration for Byron and Braidwood Units 1 and 2 is documented in Reference 2. WCAP-16143-P, Reference 13, 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 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 defined in WCAP-15392, Revision 2 (Reference 7). Consistent with the methodology described in Reference 1, 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 G-2000, 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-447 & NOZZLE SHELL FORGING
LIMITING ART VALUES AT 30.5 EFPY: 1/4T, 107°F (N-588) & 52°F ('96 App. G)
3/4T, 89°F (N-588) & 37°F ('96 App. G)

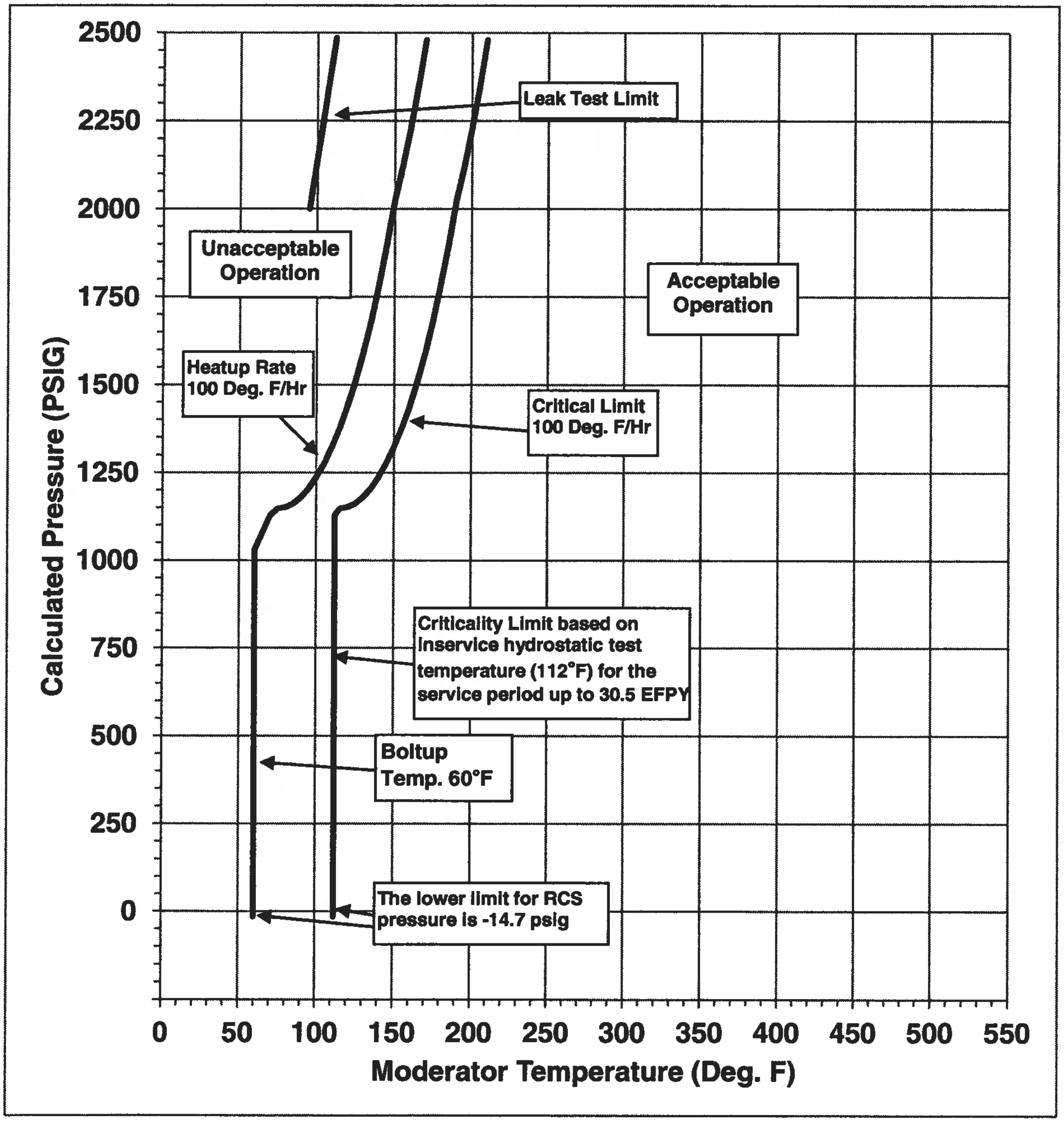

Figure 2.1
Byron Unit 2 Reactor Coolant System Heatup Limitations (Heatup rates of 100°F/hr) Applicable for 30.5 EFPY (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: CIRCUMFERENTIAL WELD WF-447 & NOZZLE SHELL FORGING
LIMITING ART VALUES AT 30.5 EFPY: 1/4T, 107°F (N-588) & 52°F ('96 App. G)
3/4T, 89°F (N-588) & 37°F ('96 App. G)

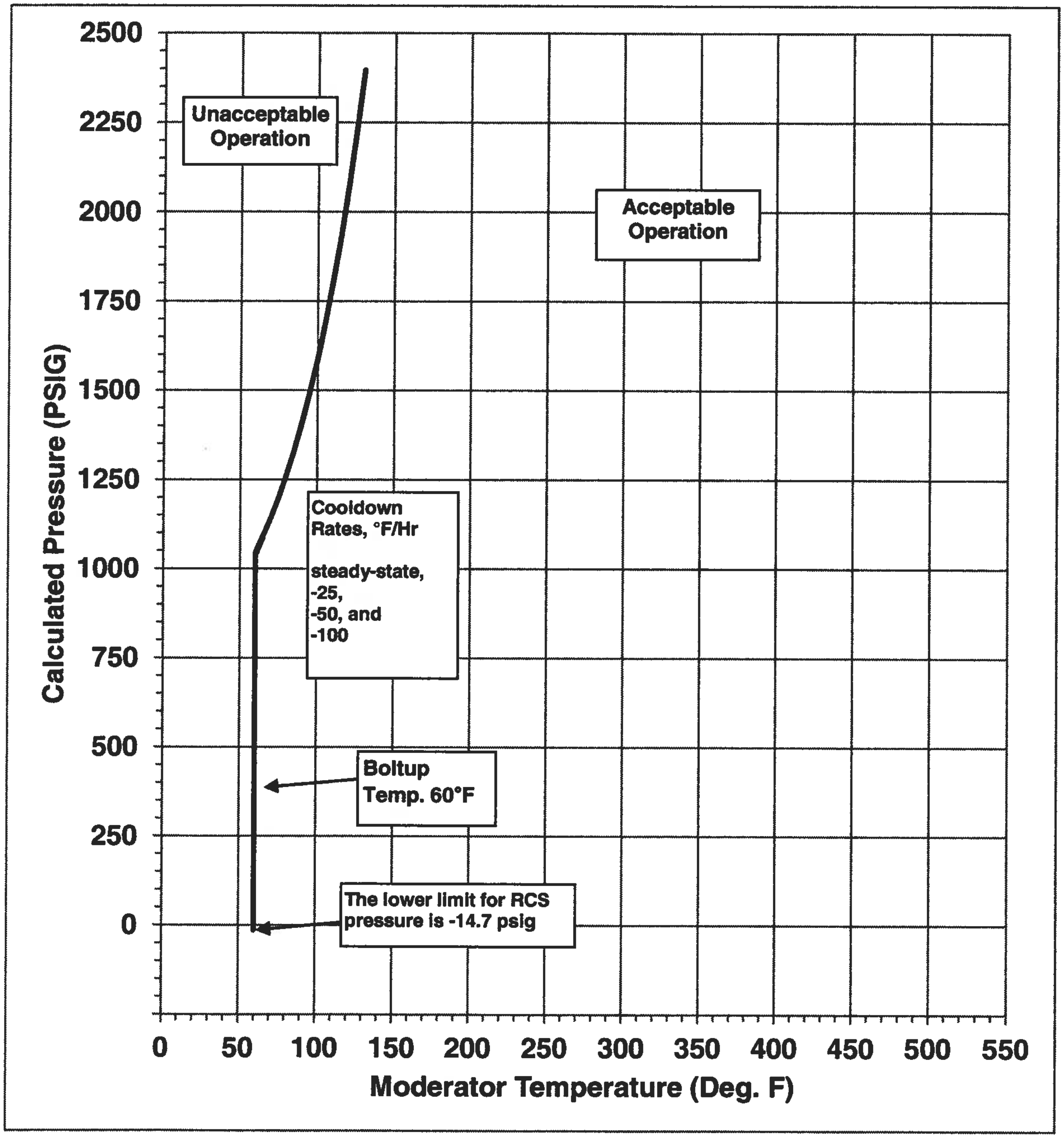

Figure 2.2

Byron Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50 and 100°F/hr) Applicable for 30.5 EFPY (Without Margins for Instrumentation Errors)

Table 2.1a
Byron Unit 2 Heatup Data Points at 30.5 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	-14.7	112	-14.7	95	2000
60	1030	112	1078	112	2485
65	1078	112	1128		
70	1128	115	1148		
75	1148	120	1152		
80	1152	125	1162		
85	1162	130	1180		
90	1180	135	1205		
95	1205	140	1237		
100	1237	145	1276		
105	1276	150	1323		
110	1323	155	1377		
115	1377	160	1440		
120	1440	165	1511		
125	1511	170	1592		
130	1592	175	1682		
135	1682	180	1784		
140	1784	185	1897		
145	1897	190	2023		
150	2023	195	2120		
155	2120	200	2227		
160	2227	205	2347		
165	2347	210	2480		
170	2480				

Table 2.1b
Byron Unit 2 Cooldown Data Points at 30.5 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	-14.7	60	-14.7	60	-14.7	60	-14.7
60	1045	60	1036	60	1033		
65	1092	65	1088				
70	1143						
75	1200						
80	1263						
85	1332						
90	1409						
95	1494						
100	1587						
105	1691						
110	1805						
115	1932						
120	2071						
125	2226						
130	2396						

Note: For each cooldown rate, the steady-state pressure values shall govern the temperature where no allowable pressure values are provided.

3.0 Low Temperature Overpressure Protection and Boltup

This section provides the Byron 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 maximum lift settings in accordance with Figure 3.1 and Table 3.1. These limits are based on References 3 and 6.

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

The required enable temperature for the PORVs shall be ≤ 350°F RCS temperature. (Byron 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 (Reference 7).

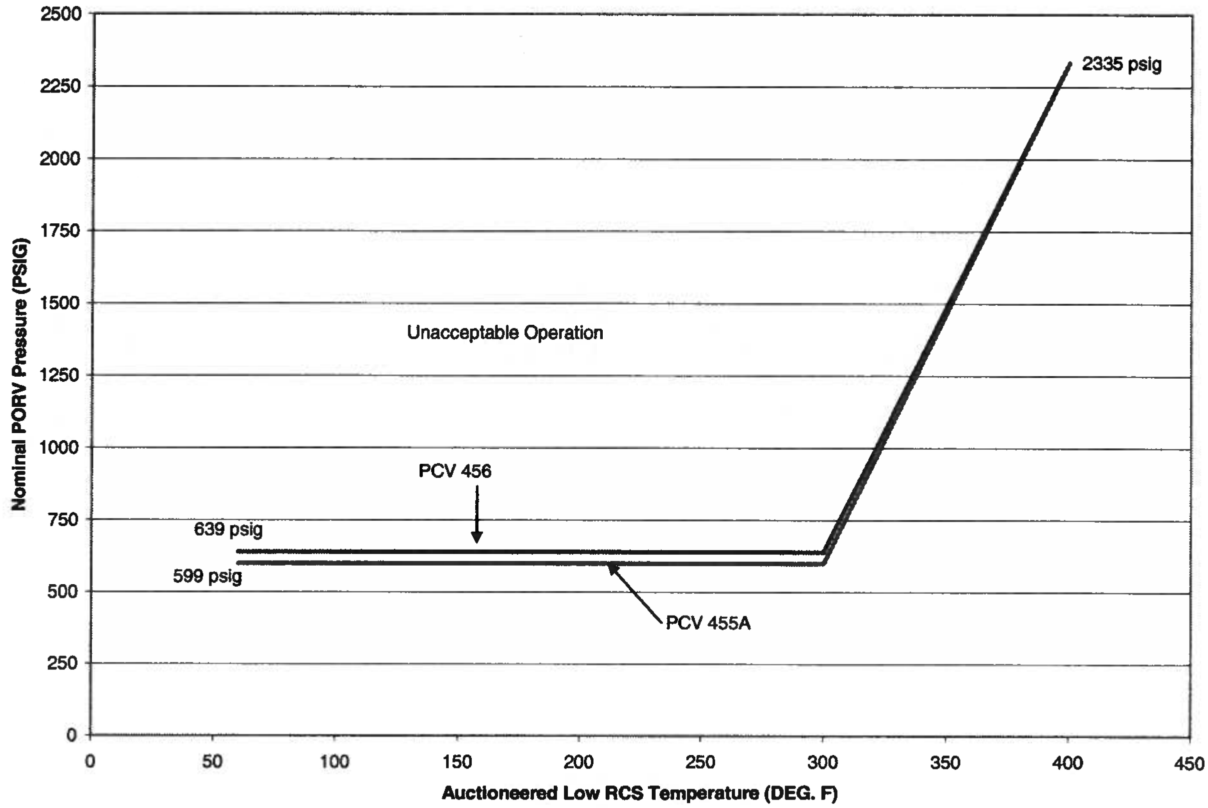

Figure 3.1
Byron Unit 2 Nominal PORV Setpoints for the Low Temperature Overpressure Protection (LTOP) System Applicable for the 30.5 EFPY (Includes Instrumentation Uncertainty)

Table 3.1
Data Points for Byron Unit 2 Nominal PORV Setpoints for the LTOP System Applicable for 30.5 EFPY (Includes Instrumentation Uncertainty)

PCV-455A

(2TY-0413M)	
AUCTIONEERED LOW RCS TEMP. (DEG. F)	RCS PRESSURE (PSIG)
50	599
300	599
400	2335

PCV-456

(2TY-0413P)	
AUCTIONEERED LOW RCS TEMP. (DEG. F)	RCS PRESSURE (PSIG)
50	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 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 the reactor vessel 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

Byron Unit 2 Surveillance 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	1.19	0.406×10^{19}
W	121.5°	4.07	4.67	1.20×10^{19}
X	238.5°	4.14	8.63	2.18×10^{19}
Z[c]	301.5°	4.11	14.28 (EOC 11)	3.25×10^{19}
V[c]	61.0°	3.88	14.28 (EOC 11)	3.07×10^{19}
Y[c]	241.0°	3.88	20.05 (EOC 15)	4.19×10^{19}

Notes:

(a) Source document is CN-AMLRS-10-8 (Reference 5), 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 Byron Unit 2 adjusted reference temperature (ART) values at the 1/4T and 3/4T locations for 32 EFPY.

Table 5.4 provides the RT_{PTS} values for Byron Unit 2 for 32 EFPY obtained from Reference 5.

Table 5.1

Byron Unit 2 Calculation of Chemistry Factors Using Surveillance Capsule Data [a]

Material	Capsule	Capsule $f^{(b)}$ (n/cm^2, E > 1.0 MeV)	$FF^{(c)}$	$\Delta RT_{NDT}^{(b)}$ (°F)	$FF*\Delta RT_{NDT}$ (°F)	FF^2
Lower Shell Forging (Tangential)	U	4.06×10^{19}	0.750	$0.0^{(d)}$	0.00	0.56
	W	1.20×10^{19}	1.051	3.65	3.84	1.10
	X	2.18×10^{19}	1.211	15.75	19.08	1.47
Lower Shell Forging (Axial)	U	04.06×10^{19}	0.750	19.76	14.82	0.56
	W	1.20×10^{19}	1.051	31.88	33.50	1.10
	X	2.18×10^{19}	1.211	38.91	47.14	1.47
				SUM:	118.38	6.27
	$CF_{LS\ Forging} = \Sigma(FF*\Delta RT_{NDT}) \div \Sigma(FF^2) = (118.38) \div (6.27) =$ **18.9°F**					
Byron Unit 1 Surveillance Weld Material (Heat #442002)	U	0.409×10^{19}	0.752	11.22 (5.61)	8.44	0.57
	X	1.49×10^{19}	1.110	80.22 (40.11)	89.08	1.23
	W	2.26×10^{19}	1.221	102.68 (51.34)	125.34	1.49
Byron Unit 2 Surveillance Weld Material (Heat #442002)	U	0.406×10^{19}	0.750	16.88 (8.44)	12.66	0.56
	W	1.20×10^{19}	1.051	57.76 (28.88)	60.70	1.10
	X	2.18×10^{19}	1.211	108.02 (54.01)	130.86	1.47
				SUM:	427.08	6.42
	$CF_{Weld\ Metal} = \Sigma(FF*\Delta RT_{NDT}) \div \Sigma(FF^2) = (427.08) \div (6.42) =$ **66.5°F**					

Notes:

a) Source document is CN-AMLRS-10-8 (Reference 5), Table 5.2-2.
b) f = fluence; ΔRT_{NDT} values are the measured 30 ft-lb shift values taken from Reference 9. ΔRT_{NDT} values for the surveillance weld data are adjusted by a ratio of 2.0 (pre-adjusted values are listed in parentheses).
c) FF = fluence factor = $f^{(0.28 - 0.10*\log f)}$.
d) Measured ΔRT_{NDT} value was determined to be negative, but physically a reduction should not occur; therefore a conservative value of zero is used.

Table 5.2

Byron Unit 2 Reactor Vessel Material Properties [a]

Material Description	Cu (%)	Ni (%)	Initial RT_{NDT} (°F)[b]
Closure Head Flange 5P7382 / 3P6407	--	0.71	0
Vessel Flange 124L556VA1	--	0.70	30
Nozzle Shell Forging 4P-6107	0.05	0.74	10
Inter. Shell Forging [49D329/49C297]-1-1	0.01	0.70	-20
Lower Shell Forging [49D330/49C298]-1-1	0.06	0.73	-20
Circumferential Weld WF-447 (HT# 442002)	0.04	0.63	10
Upper Circumferential Weld WF-562 (HT# 442011)	0.03	0.67	40
Byron Unit 1 Surveillance Program Weld Metal (Heat # 442002)	0.02	0.69	--
Byron Unit 2 Surveillance Program Weld Metal (Heat # 442002)	0.02	0.71	--
Braidwood Units 1 & 2 Surveillance Program Weld Metal (Heat # 442011)	0.03	0.67, 0.71	--

a) Reference 7.
b) Initial RT_{NDT} values are based on measured data.

Table 5.3

Summary of Byron Unit 2 Adjusted Reference Temperatures (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	0.549×10^{19}	**53**	**38**
Intermediate Shell Forging	1.76×10^{19}	21	9
Lower Shell Forging	1.76×10^{19}	52	34
→Using credible surveillance data	1.76×10^{19}	16	8
Nozzle to Intermediate Shell Forging Circ. Weld Seam (Heat # 442011)	0.549×10^{19}	97	77
→Using credible Braidwood Units 1 and 2 surveillance data	0.549×10^{19}	76	64
Intermediate to Lower Shell Forging Circ. Weld Seam (Heat # 442002)	1.70×10^{19}	119	88
→Using credible surveillance data	1.70×10^{19}	**105**	**86**

Note:

(a) The source document containing detailed calculations is CN-AMLRS-10-8 (Reference 5), 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

RT_{PTS} Calculation for Byron 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_{NTD} [C] (°F)	ΔRT_{NTD} (°F)	σ_u [c] (°F)	σ_Δ [d] (°F)	Margin	RT_{PTS} (°F)
Nozzle Shell Forging	1.1	31	0.549×10^{19}	0.8323	10	25.8	0	12.9	25.8	62
Intermediate Shell Forging	1.1	20	1.76×10^{19}	1.1554	-20	23.1	0	11.6	23.1	26
Lower Shell Forging	1.1	37	1.76×10^{19}	1.1554	-20	42.7	0	17	34	57
→ Using credible surveillance data	2.1	18.9	1.76×10^{19}	1.1554	-20	21.8	0	8.5	17	19
Nozzle to Intermediate Shell Forging Circ. Weld Seam (Heat # 442011)	1.1	41	0.549×10^{19}	0.8323	40	34.1	0	17.1	34.1	108
→ Using credible Braidwood Units 1 and 2 surveillance data	2.1	26.1	0.549×10^{19}	0.8323	40	21.7	0	10.9	21.7	83
Intermediate to Lower shell Forging Circ Weld Seam (Heat # 442002)	1.1	54	1.70×10^{19}	1.1461	10	61.9	0	28	56	128
→ Using credible surveillance data	2.1	66.5	1.70×10^{19}	1.1461	10	76.2	0	14	28	114

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-10-8 (Reference 5), 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 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 to exceed $0.5*\Delta RT_{NTD}$.

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," Andrachek, J.D., et al., January 1996.

2. WCAP-14824, Revision 2, "Byron Unit 1 Heatup and Cooldown Limit Curves for Normal Operation and Surveillance Weld Metal Integration for Byron & Braidwood", November 1997 with Westinghouse errata letters CAE-97-220, dated November 26, 1997 and CAE-97-231/CCE-97-314 and CAE-97-233/CCE-97-316, dated January 6, 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-10398, "Commonwealth Edison Company, Byron Station Unit 2 Reactor Vessel Radiation Surveillance Program," Singer, L.R., December 1983.

5. Westinghouse Calculation Note CN-AMLRS-10-8, Revision 0, "Byron Units 1 and 2 Measurement Uncertainty Recapture (MUR) Uprate: Reactor Vessel Integrity Evaluations," A. E. Leicht, September 2010.

6. Byron Station Design Information Transmittal DIT-BYR-06-046, "Transmittal of Byron Unit 1 and Unit 2 Temperature and Pressure Uncertainties for Low Temperature Overpressure System (LTOPS) Power Operated Relief Valves (PORVS)," David Neidich, August 15, 2006.

7. WCAP- 15392, Revision 2, "Byron Unit 2 Heatup and Cooldown Limit Curves for Normal Operation," T. J. Laubham, et al., November 2003.

8. NRC Letter from R. A. Capra, NRR, to O. D. Kingsley, Commonwealth Edison Co., "Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, Acceptance for Referencing of Pressure Temperature Limits Report (TAC Numbers M98799, M98800, M98801, and M98802)," January 21, 1998.

9. WCAP-15176, Revision 0, "Analysis of Capsule X from Commonwealth Edison Company Byron Unit 2 Reactor Vessel Radiation Surveillance Program," T. J. Laubham, et al., March 1999.

10. 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.

11. 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.

12. 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.

13. 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.