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

Byron Station, Unit 1 and Unit 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Subject: Issuance of Reactor Coolant System Pressure and Temperature Limits Reports for Byron Station Units 1 and 2

In accordance with Technical Specification (TS) 5.6.6, " Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," section c., we are submitting the revised RCS PTLRs for Byron Station Unit 1 and Unit 2.

The RCS PTLRs for each unit were revised in accordance with (TS) 5.6.6.b. to extend the pressure-temperature limit time exposure by 2 effective full power years.

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

Respectfully,

Stephen Kuczynski

Stephen E. Kuczynski
Site Vice President
Byron Nuclear Generating Station

SEK/jel/rah

Attachment: Byron Station, Unit 1and 2 PTLR

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Byron Station
NRC Project Manager – NRR – Byron Station
Office of Nuclear Facility Safety – Illinois Department of Nuclear Safety

A001

Attachment

Byron Station, Unit 1 and Unit 2 Reactor Coolant System Pressure and Temperature Limits Reports

BYRON UNIT 1

PRESSURE TEMPERATURE LIMITS REPORT (PTLR)

(October 2004)

BYRON - UNIT 1
PRESSURE AND TEMPERATURE LIMITS REPORT
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1.0 Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)

This PTLR for Unit 1 has been prepared in accordance with the requirements of 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 Operating Limits

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 (Reference 1) was used with the following exceptions:

a) Optional use of ASME Code Section XI, Appendix G, Article G-2000, 1996 Addenda, and
b) Use of RELAP computer code for calculation of LTOP setpoints for Byron Unit 1 replacement steam generators.

These exceptions to the methodology in WCAP-14040-NP-A have been reviewed and accepted by the NRC in Reference 16.

WCAP-15124, Reference 17, 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. Reference 23 evaluated the effect of higher fluence from 5% uprate on the existing PT curves.

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

2.1.1 The RCS temperature rate-of-change limits defined in Reference 17 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.

2.0 Operating Limits (continued)

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 2.1. The RCS P/T limits for cooldown are shown in Figure 2.2 and Table 2.1. These limits are defined in WCAP-15124, Rev. 0 (Reference 17). 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 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.

2.2 Low Temperature Overpressure Protection (LTOP) System Setpoints TS-LCO 3.4.12.

The power operated relief valves (PORVs) shall each have maximum lift settings in accordance with Figure 2.3 and Table 2.2. These limits are based on References 5, 13, and 14. The Residual Heat Removal (RH) Suction Relief Valves are also analyzed to individually provide low temperature overpressure protection. This analysis for the RH Suction Relief Valves remains valid with the current Appendix G limits contained in this PTLR document and will be reevaluated in the future as the Appendix G limits are revised.

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 maximum lift settings shown in Figure 2.3 and Table 2.2 account for appropriate instrument error.

2.3 LTOP Enable Temperature

The as analyzed LTOP enable temperature is 200°F (Reference 15 and17).

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 2.2 extends to 450°F where the pressure setpoint is 2350 psig. This is intended to prohibit PORV lift for an inadvertent LTOP system arming at power.

2.4 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 17).

2.5 Reactor Vessel Minimum Pressurization Temperature (Non-Technical Specification)

Based on the steady-state limits specified in Table 2.1, the minimum temperature at which the Reactor Vessel may be pressurized (i.e., in an unvented condition) shall be ≥ 65°F, plus an allowance for the uncertainty of the temperature instrument, determined using a technique consistent with ISA-S67.04-1994.

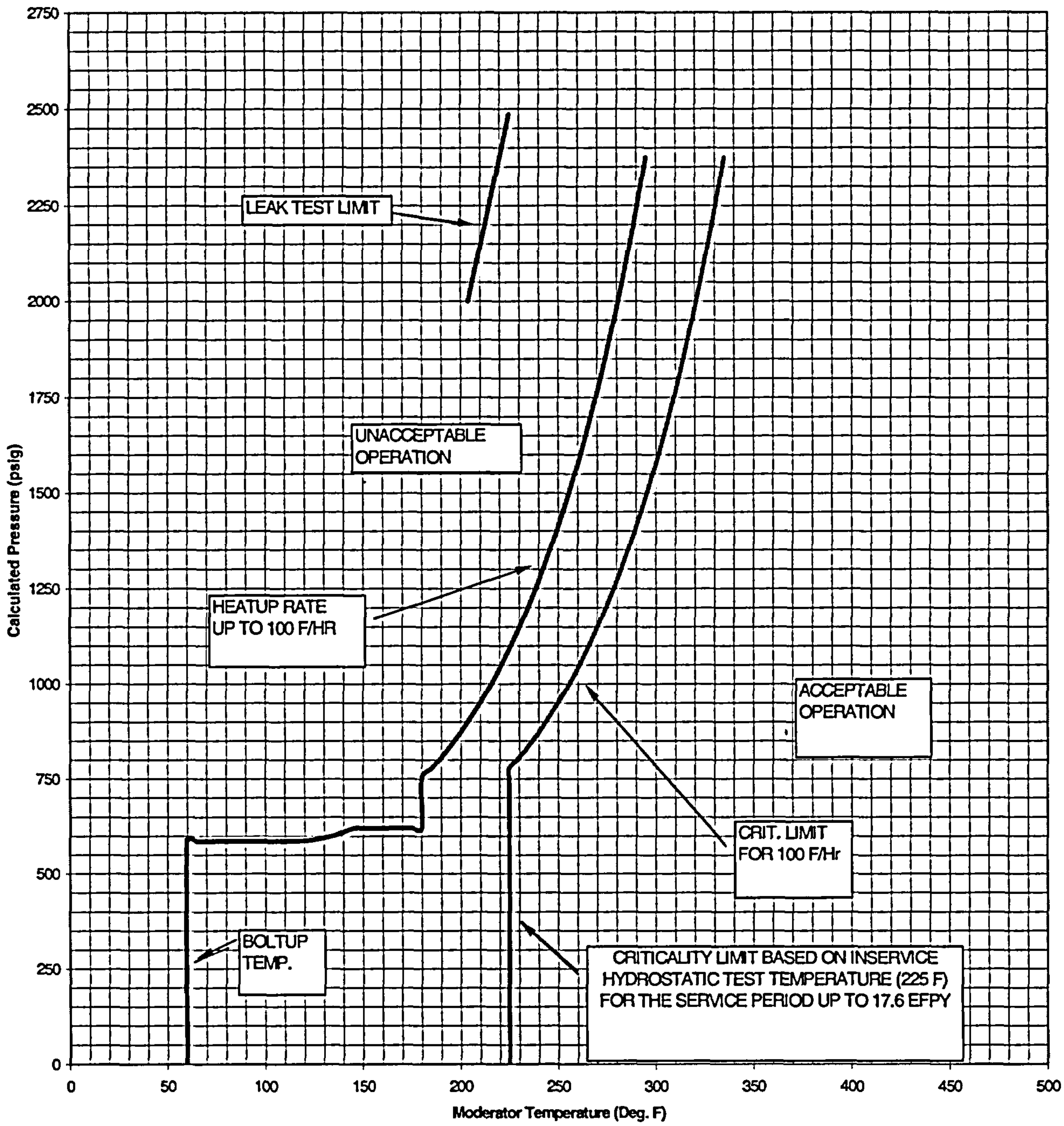

Figure 2.1:

Byron Unit 1 Reactor Coolant System Heatup Limitations (Heatup rates up to 100°F/hr) Applicable for the First 17.6 EFPY (Without margins for instrumentation errors and using 1996 Appendix G Methodology)

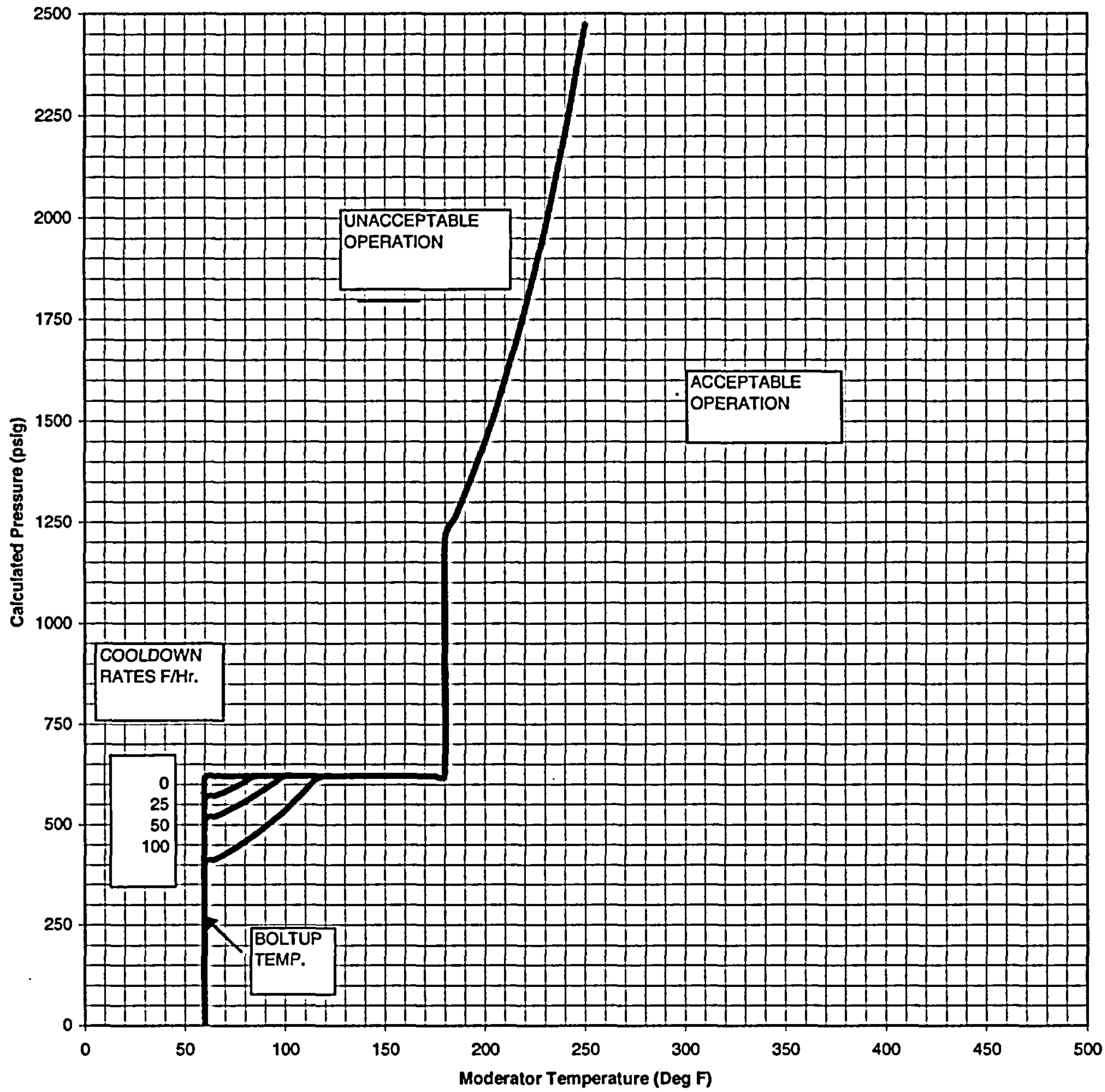

Figure 2.2:
Byron Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown rates up to 100°F/hr) Applicable for the First 17.6 EFPY (Without margins for instrumentation errors and using 1996 Appendix G Methodology)

Table 2.1

Byron Unit 1 Heatup and Cooldown Data Points at 17.6 EFPY (Without Margins for Instrumentation Errors and Using the 1996 Appendix G Methodology)

Heatup Curve						Cooldown Curves							
100 F Heatup		Criticality Limit		Leak Test Limit		Steady State		25 F Cooldown		50 F Cooldown		100 F Cooldown	
T	P	T	P	T	P	T	P	T	P	T	P	T	P
60	0	225	0	204	2000	60	0	60	0	60	0	60	0
60	587	225	587	225	2485	60	613	60	561	60	509	60	402
65	587	225	587			65	621	65	572	65	520	65	414
70	587	225	587			70	621	70	582	70	531	70	427
75	587	225	587			75	621	75	594	75	544	75	442
80	587	225	587			80	621	80	607	80	557	80	458
85	587	225	587			85	621	85	620	85	572	85	475
90	587	225	587			90	621	90	621	90	588	90	494
95	587	225	587			95	621	95	621	95	605	95	514
100	587	225	587			100	621	100	621	100	621	100	535
105	587	225	587			105	621	105	621	105	621	105	559
110	587	225	587			110	621	110	621	110	621	110	584
115	587	225	587			115	621	115	621	115	621	115	611
120	588	225	587			120	621	120	621	120	621	120	621
125	591	225	587			125	621	125	621	125	621	125	621
130	596	225	587			130	621	130	621	130	621	130	621
135	602	225	587			135	621	135	621	135	621	135	621
140	611	225	587			140	621	140	621	140	621	140	621
145	621	225	588			145	621	145	621	145	621	145	621
150	621	225	591			150	621	150	621	150	621	150	621
155	621	225	596			155	621	155	621	155	621	155	621
160	621	225	602			160	621	160	621	160	621	160	621
165	621	225	611			165	621	165	621	165	621	165	621
170	621	225	622			170	621	170	621	170	621	170	621
175	621	225	634			175	621	175	621	175	621		
180	621	225	648			180	621	180	621				
180	750	225	665			180	1207	180	1205				
185	777	225	683			185	1261						
190	806	225	703			190	1319						
195	838	225	725			195	1382						
200	872	225	750			200	1449						
205	910	225	777			205	1521						
210	950	230	806			210	1599						
215	994	235	838			215	1683						
220	1041	240	872			220	1773						
225	1092	245	910			225	1869						
230	1147	250	950			230	1973						
235	1206	255	994			235	2085						

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Table 2.1 (Continued)

Heatup Curve						Cooldown Curves							
100 F Heatup		Criticality Limit		Leak Test Limit		Steady State		25 F Cooldown		50 F Cooldown		100 F Cooldown	
T	P	T	P	T	P	T	P	T	P	T	P	T	P
240	1269	260	1041			240	2205						
245	1338	265	1092			245	2334						
250	1411	270	1147			250	2473						
255	1490	275	1206										
260	1575	280	1269										
265	1666	285	1338										
270	1764	290	1411										
275	1869	295	1490										
280	1982	300	1575										
285	2104	305	1666										
290	2234	310	1764										
295	2374	315	1869										
		320	1982										
		325	2104										
		330	2234										
		335	2374										

Note 1: Heatup and Cooldown data includes the vessel flange requirements of 180 °F and 621 psig per 10CFR50, Appendix G.
Note 2: For each cooldown rate, the steady-state pressure values shall govern the temperature where no allowable pressure values are provided.
Note 3: Temperatures and pressures are given in ° F and psig, respectively.

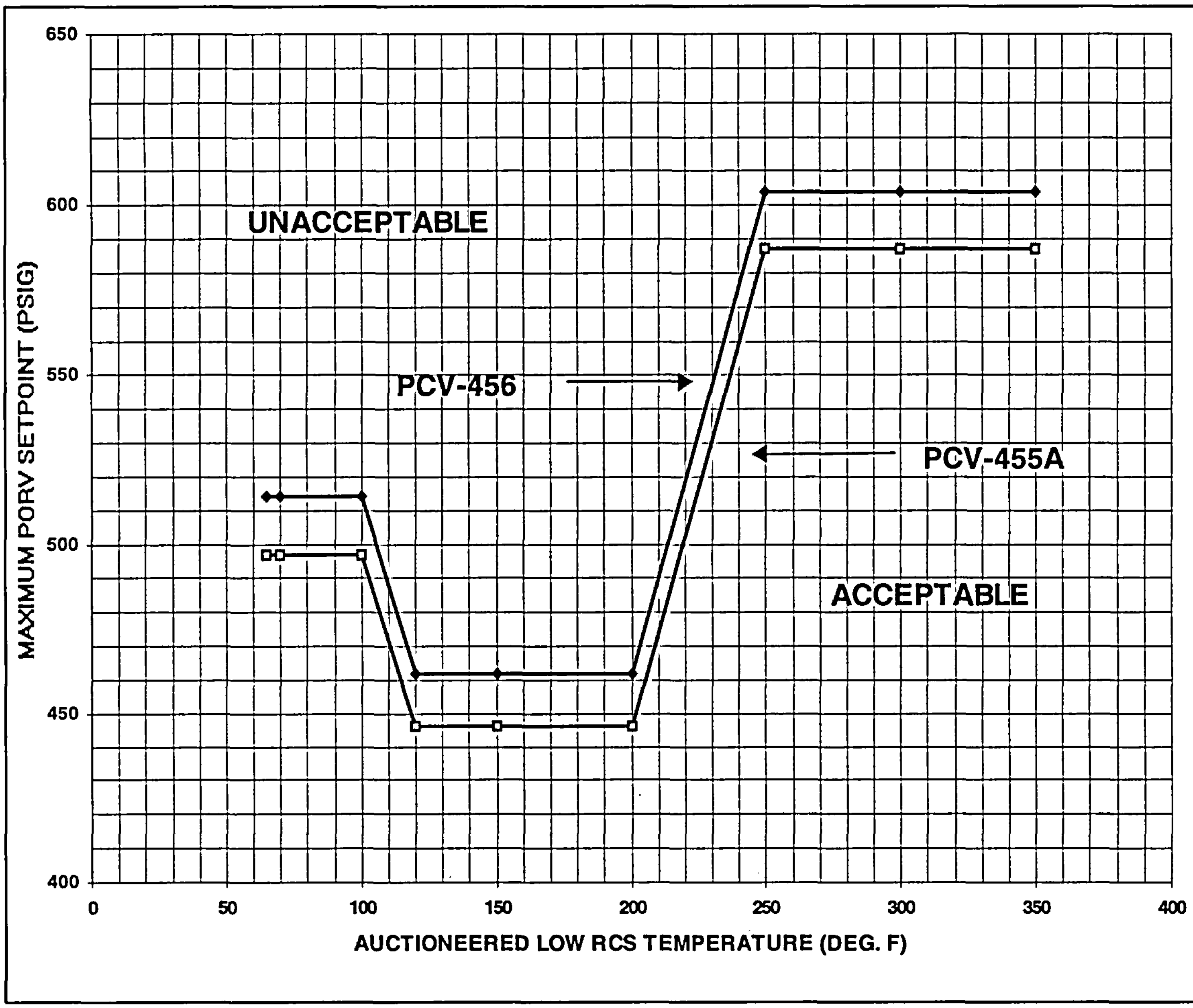

Figure 2.3
Byron Unit 1 Maximum Allowable Nominal PORV Setpoints for the Low Temperature Overpressure Protection (LTOP) System Applicable for the First 17.6 EFPY

Table 2.2
Data Points for Byron Unit 1 Maximum Allowable Setpointsfor the LTOP System Applicable for the First 17.6 EFPY

PCV-455A

(1TY-0413M)	
AUCTIONEERED LOW RCS TEMP. (DEG. F)	RCS PRESSURE (PSIG)
65	497
70	497
100	497
120	446
150	446
200	446
250	587
300	587
350	587
450	2350

PCV-456

(1TY-0413P)	
AUCTIONEERED LOW RCS TEMP. (DEG. F)	RCS PRESSURE (PSIG)
65	514
70	514
100	514
120	462
150	462
200	462
250	604
300	604
350	604
450	2350

Note: To determine maximum allowable lift setpoints for RCS Pressure and RCS Temperatures greater than 350°F, linearly interpolate between the 350°F and 450°F data points shown above. (Setpoints extend to 450°F to prevent PORV liftoff from an inadvertent LTOP system arming while at power.)

3.0 Reactor Vessel Material Surveillance Program

The pressure vessel material surveillance program (Ref. 6) 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, 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 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. Other capsules will be removed to avoid excessive fluence accumulation should they be needed to support life extension. The removal schedule is provided in Table 3.1. The time of specimen withdrawal may be modified to coincide with those refueling outages or reactor shutdowns most closely approaching the withdrawal schedule.

Table 3.1

Byron Unit 1 Capsule Withdrawal Schedule

Capsule	Vessel Location (Degrees)	Capsule Lead Factor	Removal Time(a) (EFPY)	Estimated Capsule Fluence (n/cm^2)
U	58.5°	4.22	1.15 (Removed)	4.04×10^{18}
X	238.5°	4.27	5.64 (Removed)	1.57×10^{19}
W	121.5°	4.20	9.24 (Removed)	2.43×10^{19} (b)
Z	301.5°	4.20	Standby (c)	3.27×10^{19} (c)
V	61.0°	3.97	Standby (c)	------
Y	241.0°	3.97	Standby (c)	------

a) Effective Full Power Years (EFPY) from plant startup.
b) Maximum end of license (32 EFPY) inner vessel wall fluence is estimated to be 2.02×10^{19} n/cm^2.
c) Standby capsule to be used for future license renewal (Derived from WCAP 15132, Rev. 1).

4.0 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 4.1 shows the calculation of the surveillance material chemistry factors using surveillance capsule data.

Table 4.2 provides the reactor vessel material properties table.

Table 4.3 provides a summary of the Byron Unit 1 adjusted reference temperature (ARTs) at the 1/4T and 3/4T locations for 17.6 EFPY.

Table 4.4 shows the calculation of ARTs at 17.6 EFPY for the limiting Byron Unit 1 reactor vessel material (Intermediate Shell Forging 5P-5933).

Table 4.5 provides RT_{PTS} values for Byron Unit 1 for 48 EFPY obtained from Reference 9.

Table 4.1
Calculation of Chemistry Factors Using Surveillance Capsule Data (a)

Material	Capsule	Fluence (n/cm^2, E>1.0 Mev), f	FF[(a)]	Measured ΔRT_{NDT}	FF*ΔRT_{NDT}	$(FF)^2$
Inter. Shell Forging 5P-5933 (Tangential)	U	4.04×10^{18}	0.748	28.55	21.36	0.560
	X	1.57×10^{19}	1.124	9.82	11.04	1.263
	W	2.43×10^{19}	1.239	49.20	60.96	1.535
Inter. Shell Forging 5P-5933 (Axial)	U	4.04×10^{18}	0.748	18.52	13.85	0.560
	X	1.57×10^{19}	1.124	53.03	59.61	1.263
	W	2.43×10^{19}	1.239	29.34	36.35	1.535
				Sum:	203.17	6.716
				Chemistry Factor = 203.17 ÷ 6.716=		**30.3°F**
Byron 1 Weld Metal WF-336 (Heat #442002)	U	4.04×10^{18}	0.749	11.22 (5.61) [(b)]	8.40	0.561
	X	1.57×10^{19}	1.125	80.22 (40.11) [(b)]	90.25	1.266
	W	2.43×10^{19}	1.239	102.68 (51.34) [(b)]	127.22	1.535
Byron 2 Weld Metal WF-447 (Heat #442002)	U	4.05×10^{18}	0.749	16.88 (8.44) [(b)]	12.64	0.561
	W	1.27×10^{19}	1.067	57.76 (28.88) [(b)]	61.63	1.138
	X	2.30×10^{19}	1.225	108.02 (54.01) [(b)]	132.32	1.500
				Sum:	432.46	6.561
				Chemistry Factor = 432.46÷6.561=		**65.9°F**

a) Reference 17, Table 4-8
b) Adjusted ΔRT_{NDT} per Ratio Procedure of Regulatory Guide 1.99, Rev. 2 (Ref. 12). Ratio = 2.0. See Table 4.8 of WCAP 15178, Rev. 0. (Ref. 22).

Table 4.2:
Reactor Vessel Beltline Material Unirradiated Toughness Properties (a)

Material Description	Cu (%)	Ni (%)	Initial RT_{NDT}[(a)]
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 17

Table 4.3
Summary of Adjusted Reference Temperature (ART) at 1/4T and 3/4T Location for 17.6 EFPY (a)

Material	17.6 EFPY	
	1/4T ART	3/4T ART
Intermediate Shell Forging 5P-5933	84	70
- Using Surveillance Data[(b)]	100[(c)]	92[(c)]
Lower Shell Forging 5P-5951	54	40
Circumferential Weld WF-336	62	33
- Using Credible Surveillance Data[(d)]	47	32
Circumferential Weld WF-501	54	37
- Using Credible Surveillance Data form Braidwood 1 and 2	28	21
Nozzle Shell Forging 123J218	64	51

(a) Fluence, f, is based upon f_{surf} (E>1.0 MeV) = 9.85×10^{18} at 17.6 EFPY (Ref. 23).

(b) Calculated using a chemistry factor based on Regulatory Guide (RG) 1.99, Revision 2, Position 2 along with a full margin since it was determined that this data was not credible and the Table chemistry factor was non conservative (Ref. 19).

(c) These ART values were used to generate the Byron Unit 1 17.6 EFPY heatup and cooldown curves (Ref. 17).

(d) Calculated using the chemistry factor from the Byron Unit 1 and 2 integrated surveillance data as reported in WCAP-15178 (Reference 22)

Table 4.4

Byron Unit 1 Calculation of Adjusted Reference Temperature (ART) at 17.6 EFPY at the Limiting Reactor Vessel Material, Intermediate Shell Forging 5P-5933 (Conservatively Based on Surveillance Capsule Data) (c)

Parameter	Values	
Operating Time	17.6 EFPY	
Location[b]	1/4T ART	3/4T ART
Chemistry Factor, CF (°F)	30.3	30.3
Fluence(f), n/cm^2 (E>1.0 Mev))[a]	5.91×10^{18}	2.13×10^{18}
Fluence Factor, FF	0.853	0.585
ΔRT_{NDT}= CFxFF(°F)	25.8	17.7
Initial RT $_{NDT,}$, I(°F)	40	40
Margin, M(°F)	34	34
ART= I+(CF*FF)+M, °F per RG 1.99, Revision 2	100	92

a) Fluence, f, is based upon f_{surf} (E>1.0 Mev) = 9.85×10^{18} at 17.6 EFPY (Ref. 23).
b) The Byron Unit 1 reactor vessel wall thickness is 8.5 inches at the beltline region.
c) WCAP 15123

Table 4.5:
RT_{PTS} for Byron Unit 1 Beltline Region Materials at Life Extension (48 EFPY) (f) (g)

Material	Fluence (n/cm^2, E>1.0 MeV)	FF (h)	CF (°F)	ΔRT_{PTS} [c] (°F)	Margin (°F)	$RT_{NDT(U)}$ [a] (°F)	RT_{PTS} [b] (°F)
Intermediate Shell Forging 5P-5933	2.91×10^{19}	1.28	26.0	33.3	33.3	40	107
Intermediate Shell Forging 5P-5933 using S/C Data[d]	2.91×10^{19}	1.28	30.3	38.8	34	40	113
Lower shell Forging 5P-5951	2.91×10^{19}	1.28	26.0	33.3	33.3	10	77
Inter. To Lower Shell Circ. Weld Metal WF-336 (442002)	2.91×10^{19}	1.28	54.0	69.1	56	-30	95
Inter. To Lower Shell Circ. Weld Metal (442002) using S/C Data[e]	2.91×10^{19}	1.28	65.9	84.4	28	-30	82
Nozzle Shell Forging 123J218	8.70×10^{18}	0.961	31.0	29.8	29.8	30	90
Nozzle Shell to Inter. Shell Circ. Weld Metal WF-501 (442011)	8.70×10^{18}	0.961	41.0	39.4	39.4	10	89
Nozzle Shell to Inter. Shell Circ. Weld Metal (442011) using S/C Data	8.70×10^{18}	0.961	16.7	16.0	16.0	10	42

Notes:

a) Initial RT_{NDT} values are measured values (See Table 4.2)

b) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS} + \text{Margin}$ (°F)

c) $\Delta RT_{PTS} = CF * FF$

d) Surveillance data is considered not credible, however, since the chemistry factor (CF) from the Reg. Guide Tables (Pos. 1.1) is lower (i.e. CF via Pos. 2.1 > CF via Pos. 1.1), then the Pos. 2.1 CF is used to determine PTS with a full σ_Δ margin term, i.e. 17 °F.

e) Based on Byron Unit 1 and 2 integrated surveillance data chemistry factor from WCAP-15178 (Reference 22).

f) The fluence for 48 EFPY (Ref. 9) did not incorporate the 5% increase. However, this fluence value is greater than the end-of-life fluence (32 EFPY).

g) Limiting RT_{PTS} is significantly less than the PTS Screening Criteria of 270 °F.

h) FF (Fluence Factor) = $f^{(0.28 - 0.10 * \log f)}$

5.0 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. WCAP-13880, "Analysis of Capsule X from the Commonwealth Edison Company Byron Unit 1 Reactor Vessel Radiation Surveillance Program", P.A. Peter, et. al., January 1994.

4. WCAP-12685, "Analysis of Capsule U from the Commonwealth Edison Company Byron Unit 1 Reactor Vessel Radiation Surveillance Program", E. Terek, et. al., August 1990.

5. Westinghouse Letter to Commonwealth Edison Company, CAE-96-106, "Byron Unit 1 and 2 LTOPS Setpoints Based on 10 and 12 EFPY P/T Limits", January 17, 1996.

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

7. Westinghouse Letter Report to Commonwealth Edison Company, FDRT/SPRO-009(94), "Byron Unit 1 Heatup and Cooldown Limit Curves for Normal Operation", P.A. Peter, January 1994.

8. WCAP-14044, "Westinghouse Surveillance Capsule Neutron Fluence Reevaluation", E.P. Lippencott, April 1994.

9. WCAP-15125, "Evaluation of Pressurized Thermal Shock for Byron Unit 1", Revision 0, T. J. Laubham et al., November 1998.

10. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements", Federal Register, Volume 60, No. 243, dated December 19, 1995.

11. 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events", May 15, 1991. (PTS Rule)

12. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials", U.S. Nuclear Regulatory Commission, May 1988.

13. ComEd Calculation BRW-96-906I/BYR 96-293, Rev. 0 "Channel Accuracy for Power Operated Relief Valve (PORV) Setpoints and Wide Range RCS Temperature Indication (Unit 1 Original Steam Generators and Replacement Steam Generators)".

5.0 References (continued)

14. ComEd Nuclear Fuel Services Department NDIT No. 960186, Revision 1 "Maximum Allowable LTOPS PORV Setpoints for Byron Unit 1 with RSGs".

15. Westinghouse Letter to ComEd, CAE-97-211/CCE-97-290, "Byron and Braidwood Units 1 and 2 ΔT Metal Evaluation," November 7, 1997.

16. 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.

17. WCAP- 15124, Revision 0, "Byron Unit 1 Heatup and Cooldown Limit Curves for Normal Operation," T. J. Laubham, et al., November 1998.

18. 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.

19. WCAP-15183, Revision 0, "Commonwealth Edison Company Byron Unit 1 Surveillance Program Credibility Evaluation," T. J. Laubham, et al., June 1999.

20. 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.

21. WCAP-15180, Revision 0, "Commonwealth Edison Company Byron Unit 2 Surveillance Program Credibility Evaluation," T. J. Laubham, et al., June 1999.

22. WCAP- 15178, Revision 0, "Byron Unit 2 Heatup and Cooldown Limit Curves for Normal Operation," T. J. Laubham, et al., June 1999

23. Westinghouse Calculation CN-EMT-01-8, "Braidwood Unit 1 and 2, Development of New Pressure Temperature Limit Curves and Evaluation of Byron Units 1 and 2 PT Curve EFPY

24. 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

BYRON UNIT 2

PRESSURE TEMPERATURE LIMITS REPORT (PTLR)

(October 2004)

BYRON - UNIT 2
PRESSURE AND TEMPERATURE LIMITS REPORT
(October 2004)

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BYRON - UNIT 2
PRESSURE AND TEMPERATURE LIMITS REPORT
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1.0 Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)

This PTLR for Unit 2 has been prepared in accordance with the requirements of 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 Operating Limits

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 (Reference 1) was used with the following exception:

a) Use of ASME Code Section XI, Appendix G, Article G-2000, 1996 Addenda,

This exception to the methodology in WCAP-14040-NP-A has been reviewed and accepted by the NRC in Reference 17.

WCAP-15178, Reference 14, 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. Reference 20 evaluated the effect of higher fluence from the 5% uprate on the existing PT curves. The weld metal data integration for Byron and Braidwood Units 1 and 2 is documented in Reference 2.

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

2.1.1 The RCS temperature rate-of-change limits defined in Reference 14 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 2.1. The RCS P/T limits for cooldown are shown in Figure 2.2 and Table 2.1. These limits are defined in Reference 14. 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 Code Section XI, Appendix G, Article G-2000, 1996 Addenda. The criticality limit curve specifies pressure-temperature limits for core

2.0 Operating Limits (continued)

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.

2.2 Low Temperature Overpressure Protection (LTOP) System Setpoints TS-LCO 3.4.12.

The power operated relief valves (PORVs) shall each have maximum lift settings in accordance with Figure 2.3 and Table 2.2. These limits are based on References 5, 13 and 15. The Residual Heat Removal (RH) Suction Relief Valves are also analyzed to individually provide low temperature overpressure protection. This analysis for the RH Suction Relief Valves remains valid with the current Appendix G limits contained in this PTLR document and will be reevaluated in the future as the Appendix G limits are revised.

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 maximum lift settings shown in Figure 2.3 and Table 2.3 account for appropriate instrument error.

2.3 LTOP Enable Temperature

The as-analyzed LTOP enable temperature is 200°F (References 14 and 16).

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 2.2 extends to 450°F where the pressure setpoint is 2350 psig. This is intended to prohibit PORV lift for an inadvertent LTOP system arming at power.

2.4 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 2).

2.5 Reactor Vessel Minimum Pressurization Temperature (Non-Technical Specification)

Based on the steady-state limits specified in Table 2.1, the minimum temperature at which the Reactor Vessel may be pressurized (i.e., in an unvented condition) shall be ≥ 60°F, plus an allowance for the uncertainty of the temperature instrument, determined using a technique consistent with ISA-S67.04-1994.

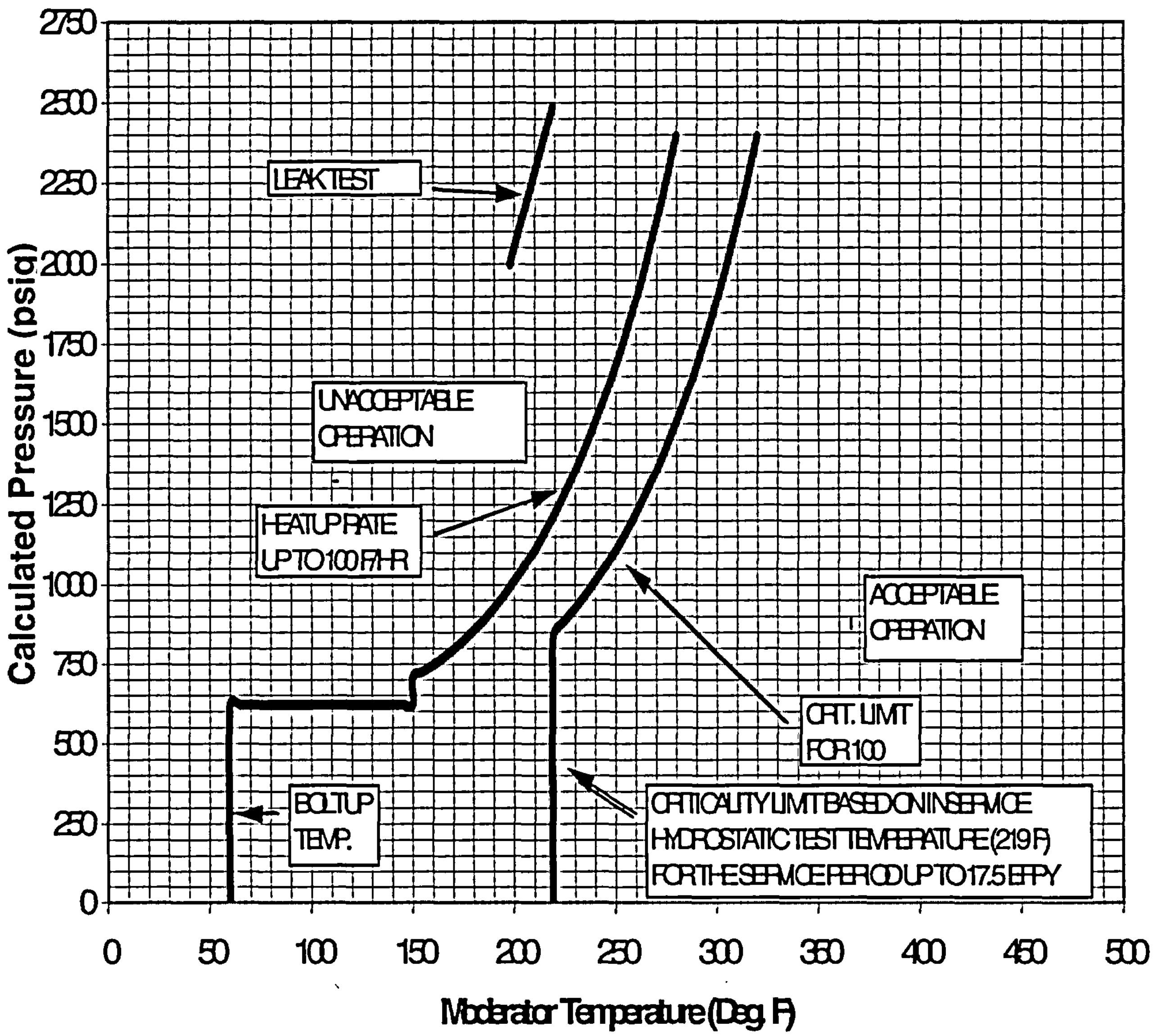

Figure 2.1:
Byron Unit 2 Reactor Coolant System Heatup Limitations (Heatup rates up to 100 °F/hr) Applicable for 17.5 EFPY (Without margins for instrumentation errors and using 1996 Appendix G Methodology)

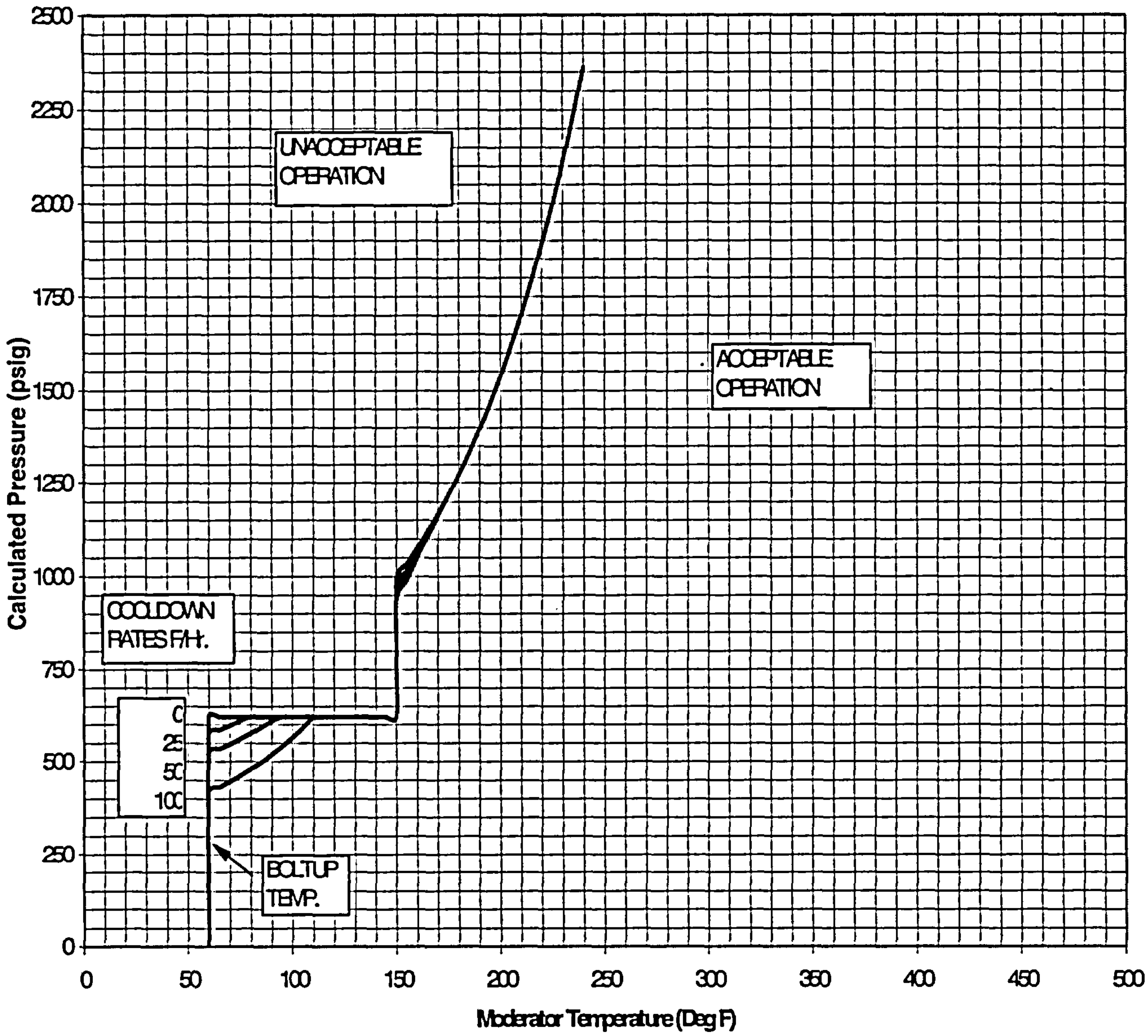

Figure 2.2:

Byron Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100 °F/hr) Applicable for 17.5 EFPY (Without Margins for Instrumentation Errors and using 1996 Appendix G Methodology)

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(October 2004)

Table 2.1: Byron Unit 2 Heatup and Cooldown Data Points at 17.5 EFPY
(Without Margins for Instrumentation Errors and Using the 1996 Appendix G Methodology)

Heatup Curve						Cooldown Curves							
100 F Heatup		Criticality Limit		Leak Test Limit		Steady State		25 F Cooldown		50 F Cooldown		100 F Cooldown	
T	P	T	P	T	P	T	P	T	P	T	P	T	P
60	0	219	0	198	2000	60	0	60	0	60	0	60	0
60	621	219	635	219	2485	60	621	60	574	60	523	60	418
65	621	219	674			65	621	65	585	65	534	65	431
85	621	219	660			70	621	70	597	70	547	70	446
90	621	219	650			75	621	75	610	75	561	75	462
95	621	219	643			80	621	80	621	80	576	80	480
100	621	219	638			85	621	85	621	85	592	85	498
105	621	219	637			90	621	90	621	90	609	90	519
110	621	219	637			95	621	95	621	95	621	95	541
115	621	219	641			100	621	100	621	100	621	100	564
120	621	219	646			105	621	105	621	105	621	105	590
125	621	219	654			110	621	110	621	110	621	110	618
130	621	219	664			115	621	115	621	115	621	115	621
135	621	219	676			120	621	120	621	120	621	120	621
140	621	219	690			125	621	125	621	125	621	125	621
145	621	219	707			130	621	130	621	130	621	130	621
150	621	219	725			135	621	135	621	135	621	135	621
150	707	219	746			140	621	140	621	140	621	140	621
155	725	219	770			145	621	145	621	145	621	145	621
160	746	219	796			150	621	150	621	150	621	150	621
165	770	219	824			150	995	150	974	150	957	150	935
170	796	220	855			155	1033	155	1016	155	1002	155	989
175	824	225	889			160	1075	160	1061	160	1051	160	1048
180	855	230	926			165	1119	165	1109	165	1104	165	1112
185	889	235	966			170	1166	170	1161	170	1161		
190	926	240	1010			175	1217	175	1217				
195	966	245	1057			180	1272						
200	1010	250	1107			185	1331						
205	1057	255	1162			190	1395						
210	1107	260	1221			195	1463						
215	1162	265	1285			200	1536						
220	1221	270	1353			205	1615						
225	1285	275	1427			210	1700						
230	1353	280	1506			215	1791						
235	1427	285	1591			220	1889						
240	1506	290	1683			225	1995						
245	1591	295	1781			230	2108						
250	1683	300	1887			235	2230						
255	1781	305	2001			240	2361						
260	1887	310	2123										
265	2001	315	2254										
270	2123	320	2395										
275	2254												
280	2395												

Note 1: Heatup and Cooldown data includes the vessel flange requirements of 180 °F and 621 psig per 10CFR50, Appendix G.

Note 2: For each cooldown rate, the steady-state pressure values shall govern the temperature where no allowable pressure values are provided.

Note 3: Temperatures and pressures are given in ° F and psig, respectively.

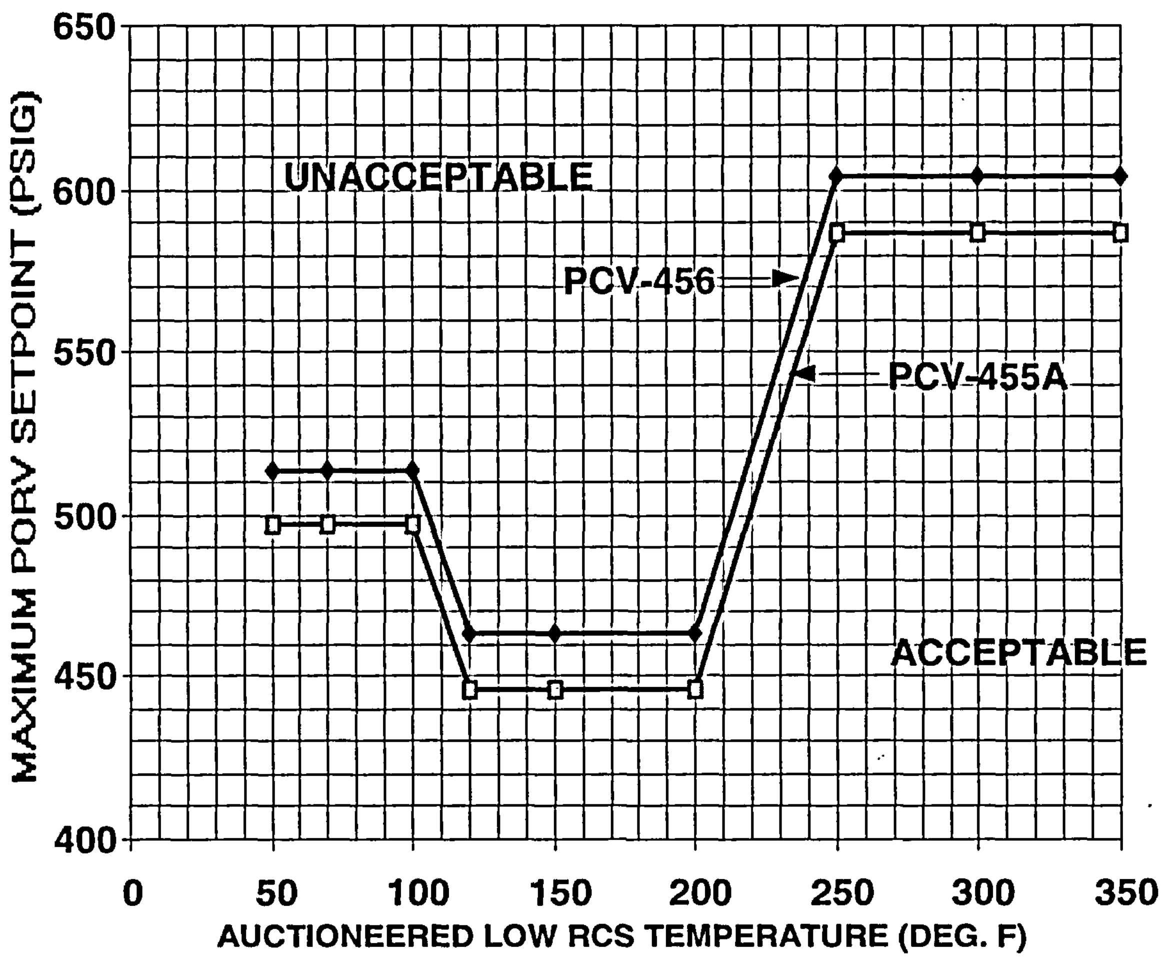

Figure 2.3: Byron Unit 2 Maximum Allowable Nominal PORV Setpoints for the Low Temperature Overpressure Protection (LTOP) System Applicable for the First 17.5 EFPY

Table 2.2: Data Points for Byron Unit 2 Maximum Allowable PORV Setpoints for the LTOP System Applicable for 17.5 EFPY

PCV-455A

(2TY-0413M)	
AUCTIONEERED LOW RCS TEMP. (DEG. F)	RCS PRESSURE (PSIG)
50	497
70	497
100	497
120	446
150	446
200	446
250	587
300	587
350	587
450	2350

PCV-456

(2TY-0413P)	
AUCTIONEERED LOW RCS TEMP. (DEG. F)	RCS PRESSURE (PSIG)
50	514
70	514
100	514
120	462
150	462
200	462
250	604
300	604
350	604
450	2350

Note: To determine maximum allowable lift setpoints for RCS Pressure and RCS Temperatures greater than 350°F, linearly interpolate between the 350°F and 450°F data points shown above. (Setpoints extend to 450°F to prevent PORV liftoff from an inadvertent LTOP system arming while at power.)

3.0 Reactor Vessel Material Surveillance Program

The pressure vessel material surveillance program (Reference 6) 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 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 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. Other capsules will be removed to avoid excessive fluence accumulation should they be needed to support life extension. The removal schedule is provided in Table 3.1. The time of specimen withdrawal may be modified to coincide with those refueling outages or reactor shutdowns most closely approaching the withdrawal schedule.

Table 3.1: Byron Unit 2 Capsule Withdrawal Schedule

Capsule	Vessel Location (Degrees)	Capsule Lead Factor	Removal Time(a) (EFPY)	Estimated Capsule Fluence (n/cm^2) (c)
U	58.5°	4.40	1.15 (Removed)	4.05×10^{18}
W	121.5°	4.25	4.634 (Removed)	1.27×10^{19}
X	238.5°	4.25	8.573 (Removed at EOL Wall)	2.30×10^{19} (b)
Z	301.5°	4.21	Standby (c)	3.35×10^{19} (c)
V	61.0°	3.97	Standby (c)	-------
Y	241.0°	3.97	Standby (c)	-------

a) Effective Full Power Years (EFPY) from plant startup.
b) Maximum end of license (32 EFPY) inner vessel wall fluence is estimated to be 2.06×10^{19} n/cm^2.
c) Standby capsule to be used for future license renewal (derived from Table 7-1 of WCAP 15176, Ref. 18).

4.0 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 4.1 shows the calculation of the surveillance material chemistry factors using surveillance capsule data.

Table 4.2 provides the reactor vessel material properties table.

Table 4.3 provides a summary of the Byron Unit 2 adjusted reference temperature (ARTs) at the 1/4T and 3/4T locations for 17.5 EFPY.

Table 4.4 shows the calculation of ARTs at 17.5 EFPY for the limiting Byron Unit 2 reactor vessel material, i.e. weld metal HT # 442002, (Based on Surveillance Capsule Data).

Table 4.5 provides RT_{PTS} values for Byron Unit 2 for 32 EFPY obtained from Reference 9.

Table 4.6 provides RT_{PTS} values for Byron Unit 2 for 48 EFPY obtained from Reference 9.

Table 4-1: Calculation of Chemistry Factors Using Surveillance Capsule Data (a)

Material	Capsule	Fluence (n/cm^2, E>1.0MeV)	$FF^{(a)}$	Measured $\Delta RT_{NDT}{}^{(b)}$	$FF*\Delta RT_{NDT}$	FF^2
Lower Shell Forging 49D330/49C298-1-1 (Tangential)	U	$4.05*10^{18}$	0.749	0.0	0	0.561
	W	$1.27*10^{19}$	1.067	3.65	3.89	1.138
	X	$2.30*10^{19}$	1.225	15.75	19.29	1.500
Lower Shell Forging 49D330/ 49C298-1-1	U	$4.05*10^{18}$	0.749	19.76	14.80	0.561
	W	$1.27*10^{19}$	1.067	31.88	34.02	1.138
	X	$2.30*10^{19}$	1.225	38.91	47.66	1.500
				SUM:	119.66	6.398
	$CF_{Forging} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (119.66) \div (6.398) =$					**18.7°F**
Byron Unit 1 Surv. Weld Material (Heat # 442002)	U	$4.04*10^{18}$	0.749	11.22 (5.61)$^{(b)}$	8.40	0.561
	X	$1.57*10^{19}$	1.125	80.22 (40.11)$^{(b)}$	90.25	1.266
	W	$2.43*10^{19}$	1.239	102.68 (51.34)$^{(b)}$	127.22	1.535
Byron Unit 2 Surv. Weld Material (Heat # 442002)	U	$4.05*10^{18}$	0.749	16.88 (8.44)$^{(b)}$	12.64	0.561
	W	$1.27*10^{19}$	1.067	57.76 (28.88)$^{(b)}$	61.63	1.138
	X	$2.30*10^{19}$	1.225	108.02 (54.01)$^{(b)}$	132.32	1.500
				SUM:	432.46	6.561
	$CF_{Surv. Weld, 442002} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (432.46) \div (6.561) =$					**65.9°F**$^{(f)}$

a) Reference 14, Table 4-8
b) ΔRT_{NDT} values are the measured 30 ft-lb shift values taken from Ref. 18.

Table 4.2: Reactor Vessel Beltline Material Unirradiated Toughness Properties (a)

Material Description	Cu (%)	Ni (%)	Initial RT_{NDT} (°F)
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 49D329-1-1/49C297-1-1	0.01	0.70	-20
Lower Shell Forging 49D330-1-1/49C298-1-1	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 # 442002)	0.03	0.67, 0.71	----

a) Reference 14.

Table 4.3:Summary of Adjusted Reference Temperature (ART) at 1/4T and 3/4TLocation for 17.5 EFPY (a)

Material Description	17.5 EFPY	
	1/4T ART(°F)	3/4T ART(°F)
Intermediate Shell Forging 49D329-1/49C297-1 (RG Position 1 (b))	14	4
Lower Shell Forging 49D330-1/49C298-1 (RG Position 1(b))	43	24
Using capsule data (RG Position 2(b))	12	2
Circumferential Weld WF-447 (HT# 442002) (RG Position 1(b))	102	73
Using credible surveillance capsule data (RG Position 2(b))	94(c)	77(c)
Nozzle Shell Forging 4P-6107 (RG Position 1 (b))	41	29
Nozzle Shell to Intermediate Shell Weld WF-562 (HT # 442011)	82	65
Using credible surveillance capsule data (RG Position 2(b))	57	50

(a) Fluence, f, is based upon f_{surf} (E>1.0 Mev) = 9.86×10^{18} at 17.5 EFPY, Reference 20.
(b) Calculated using a chemistry factor based on Regulatory Guide (RG) 1.99, Positions 1 and 2, Reference 12, as reported in WCAP-15178, Reference 14.
(c) These ART values were used to generate the Byron Unit 2 Heatup and Cooldown Curves, WCAP-15178 (Reference 14).

Table 4.4:Byron Unit 2 Calculation of Adjusted Reference Temperature (ART) at 17.5 EFPY at the Limiting Reactor Vessel Material Weld Metal (Based on Surveillance Capsule Data) (c)

Parameter	Values	
Operating Time	17.5 EFPY	
Location(b)	1/4T ART	3/4T ART
Chemistry Factor, CF (°F)	65.9	65.9
Fluence(f), n/cm^2 (E>1.0 Mev)(a)	5.92×10^{18}	2.14×10^{18}
Fluence Factor, FF	0.853	0.586
ΔRT_{NDT}= CFxFF(°F)	56.2	38.6
Initial RT_{NDT}, I(°F)	10	10
Margin, M (°F)	28.0	28.0
ART= I+(CF*FF)+M,°F per RG 1.99, Revision 2	94	77

(a) Fluence, f, is based upon f_{surf} (E>1.0 Mev) = 9.86×10^{18} at 17.5 EFPY, Reference 20.
(b) The Byron Unit 2 reactor vessel wall thickness is 8.5 inches at the beltline region.
(c) WCAP 15178

Table 4.5: RT_{PTS} for Byron Unit 2 Beltline Region Materials - 32 EFPY

Material	Fluence[a] (n/cm^2, E>1.0 MeV)	FF[b]	CF (°F)	ΔRT_{PTS}[d] (°F)	Margin (°F)	$RT_{NDT(U)}$[e] (°F)	RT_{PTS}[f] (°F)
Intermediate Shell Forging	$2.06 * 10^{19}$	1.20	20	23.8	23.8	-20	28
Lower Shell Forging	$2.06 * 10^{19}$	1.20	37	44.0	34	-20	58
Lower Shell Forging Using S/C Data[c]	$2.06 * 10^{19}$	1.20	18.7	22.3	17	-20	19
Nozzle Shell Forging	$5.22 * 10^{18}$	0.818	31	25.0	25	10	60
Inter. to Lower Shell Circ. Weld	$2.03 * 10^{19}$	1.19	54	63.7	56	10	130
Inter. to Lower Shell Circ. Weld Using S/C Data[c]	$2.03 * 10^{19}$	1.19	65.9	77.8	28	10	**116**[g]
Nozzle Shell to Inter. Shell Circ. Weld	$5.22 * 10^{18}$	0.818	41	33.1	33.1	40	106
Nozzle Shell to Inter. Shell Circ. Weld Using S/C Data[c]	$5.22 * 10^{18}$	0.818	16.7	13.5	13.5	40	67

(a) Fluence projections for 32 EFPY from Byron 2 PTS report, WCAP-157177 (Reference 9)
(b) FF (Fluence Factor) = $f^{(0.28-0.10*\log f)}$
(c) Calculated using a CF based on surveillance capsule data per RG 1.99, Position 2 (Reference 12).
(d) ΔRT_{PTS} = CF * FF
(e) Initial RT_{NDT} values are measured values (See Table 4.2)
(f) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS}$ + Margin (°F)
(g) Limiting RT_{PTS} is significantly less than the PTS Screening Criteria of 300 °F.

Table 4.6: RT_{PTS} for Byron Unit 2 Beltline Region Materials at Life Extension (48 EFPY) (a) (g)

Material	Fluence[a] (n/cm^2, E>1.0 MeV)	FF[b]	CF (°F)	ΔRT_{PTS}[d] (°F)	Margin (°F)	$RT_{NDT(U)}$[e] (°F)	RT_{PTS}[f] (°F)
Intermediate Shell Forging	$2.98 * 10^{19}$	1.29	20	25.8	25.8	-20	32
Lower Shell Forging	$2.98 * 10^{19}$	1.29	37	47.7	34	-20	62
Lower Shell Forging Using S/C Data[c]	$2.98 * 10^{19}$	1.29	18.7	24.1	17	-20	21
Nozzle Shell Forging	$7.53*10^{18}$	0.920	31	28.5	28.5	10	67
Inter. to Lower Shell Circ. Weld	$2.93 * 10^{19}$	1.29	54	69.7	56	10	136
Inter. to Lower Shell Circ. Weld Using S/C Data[c]	$2.93 * 10^{19}$	1.29	65.9	85	28	10	123[g]
Nozzle Shell to Inter. Shell Circ. Weld	$7.53*10^{18}$	0.920	41	37.7	37.7	40	115
Nozzle Shell to Inter. Shell Circ. Weld Using S/C Data[c]	$7.53*10^{18}$	0.920	16.7	15.4	15.4	40	71

(a) The fluence for 48 EFPY (Ref. 9) did not incorporate the 5% increase. However, this fluence value is greater than the end-of-life fluence (32 EFPY).
(b) FF (Fluence Factor) = $f^{(0.28-0.10*\log f)}$
(c) Calculated using a CF based on surveillance capsule data per RG 1.99, Position 2 (Reference 12).
(d) ΔRT_{PTS} = CF * FF
(e) Initial RT_{NDT} values are measured values (See Table 4.2)
(f) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS}$ + Margin (°F)
(g) Limiting RT_{PTS} is significantly less than the PTS Screening Criteria of 300 °F.

5.0 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. Deleted

3. WCAP-14064, "Analysis of Capsule W from the Commonwealth Edison Company Byron Unit 2 Reactor Vessel Radiation Surveillance Program," Malone, M.J., et al., July 1994.

4. WCAP-12431, "Analysis of Capsule U from the Commonwealth Edison Company Byron Unit 2 Reactor Vessel Radiation Surveillance Program," Terek, E., et al., October 1989.

5. Westinghouse Letter to Commonwealth Edison Company, CAE-96-106, "Byron Unit 1 and 2 LTOPS Setpoints Based on 10 and 12 EFPY P/T Limits," January 17, 1996.

6. WCAP-10398, "Commonwealth Edison Company, Byron Station Unit 2 Reactor Vessel Radiation Surveillance Program," Singer, L.R., December 1983.

7. WCAP-14063, "Commonwealth Edison Company, Byron Unit 2 Heatup and Cooldown Limit Curves for Normal Operation," Peter, P.A., November 1994.

8. Deleted

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