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

Byron Station, Unit 1
Facility Operating License No. NPF- 37
NRC Docket No. STN 50-454

Subject: Byron Station Unit 1 Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)

In accordance with Technical Specification 5.6.6, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," we are submitting the December 2006 revision to the Unit 1 PTLR. The PTLR was revised to extend the applicability of the heatup and cooldown curves out to 32 Effective Full Power Years.

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

Respectfully,

David M Hoots
Site Vice President
Byron Nuclear Generating Station

Attachment: Byron Station Unit 1 PTLR

DMH/JEL/rah

ATTACHMENT

Byron Station Unit 1 Reactor Coolant System (RCS)
Pressure and Temperature Limits Report (PTLR)
(December 2006)

BYRON UNIT 1

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

(December 2006)

BYRON - UNIT 1
PRESSURE AND TEMPERATURE LIMITS REPORT

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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 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-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

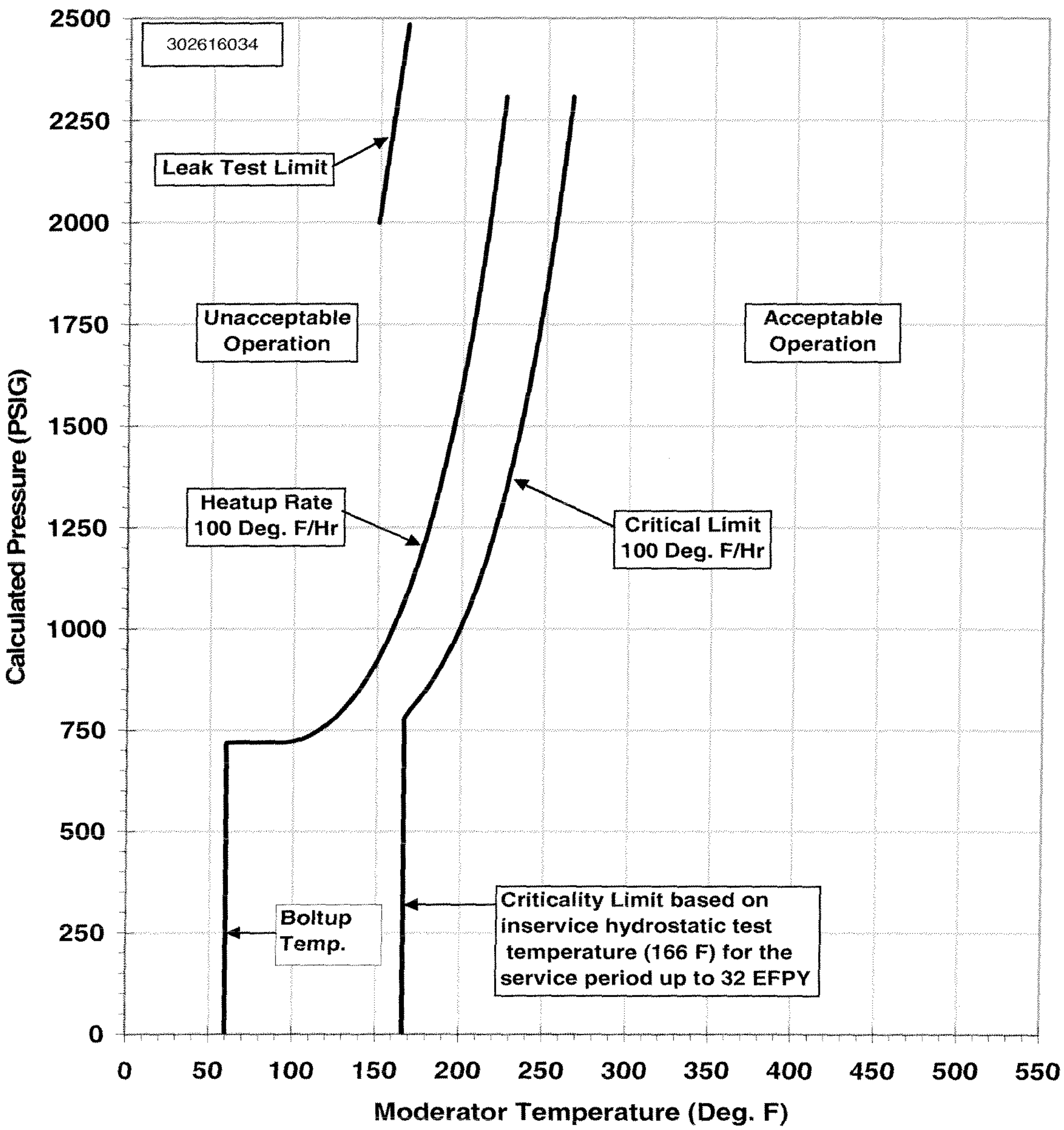

Figure 2.1
Byron Unit 1 Reactor Coolant System Heatup Limitations (Heatup rates of 100°F/hr) Applicable for 32 EFPY (Without Margins for Instrumentation Errors)

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

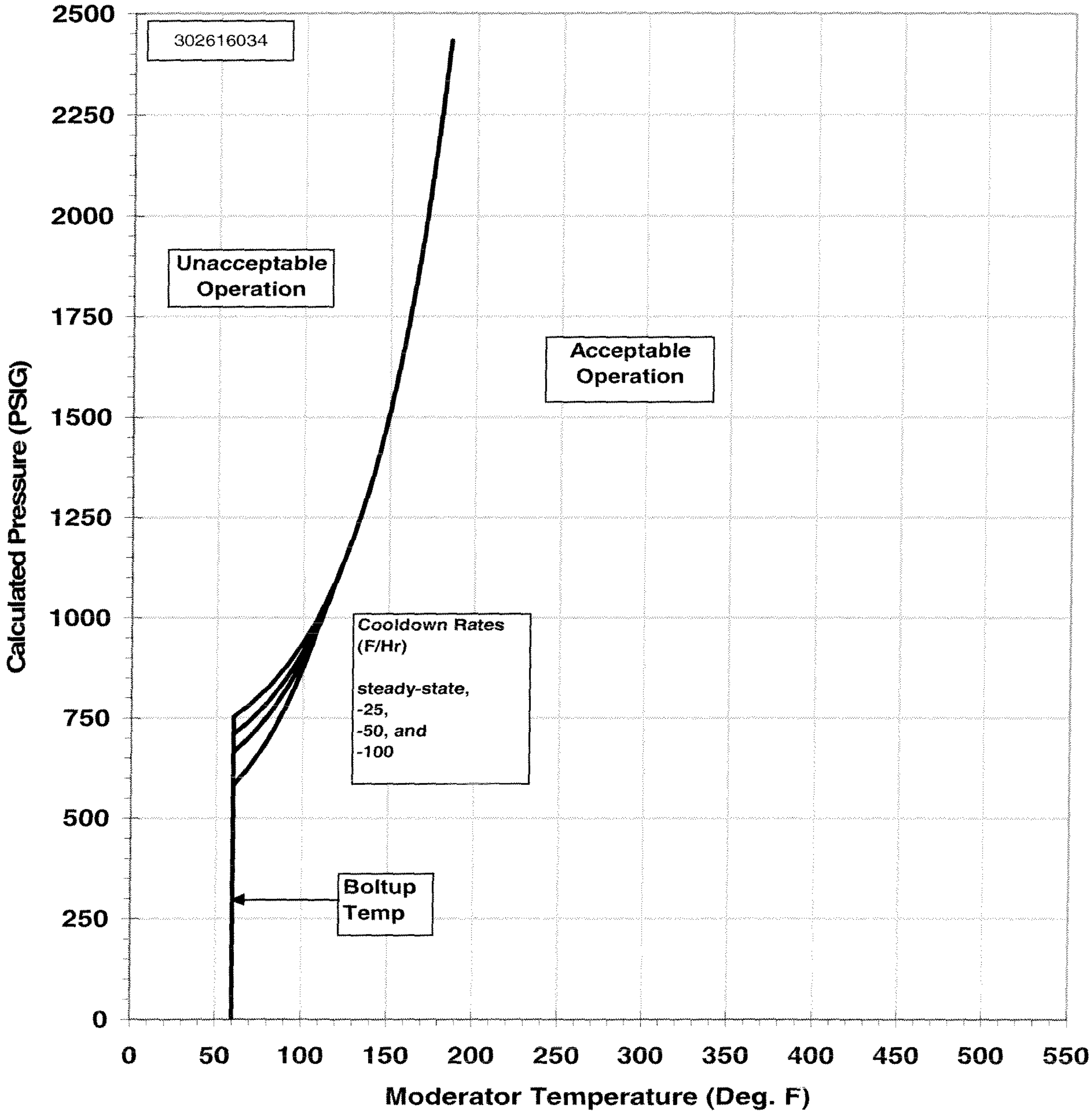

Figure 2.2
Byron Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown rates of 0, 25, 50 and 100°F/hr) Applicable for 32 EFPY (Without Margins for Instrumentation Errors)

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	0	166	0	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	0	60	0	60	0	60	0
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).

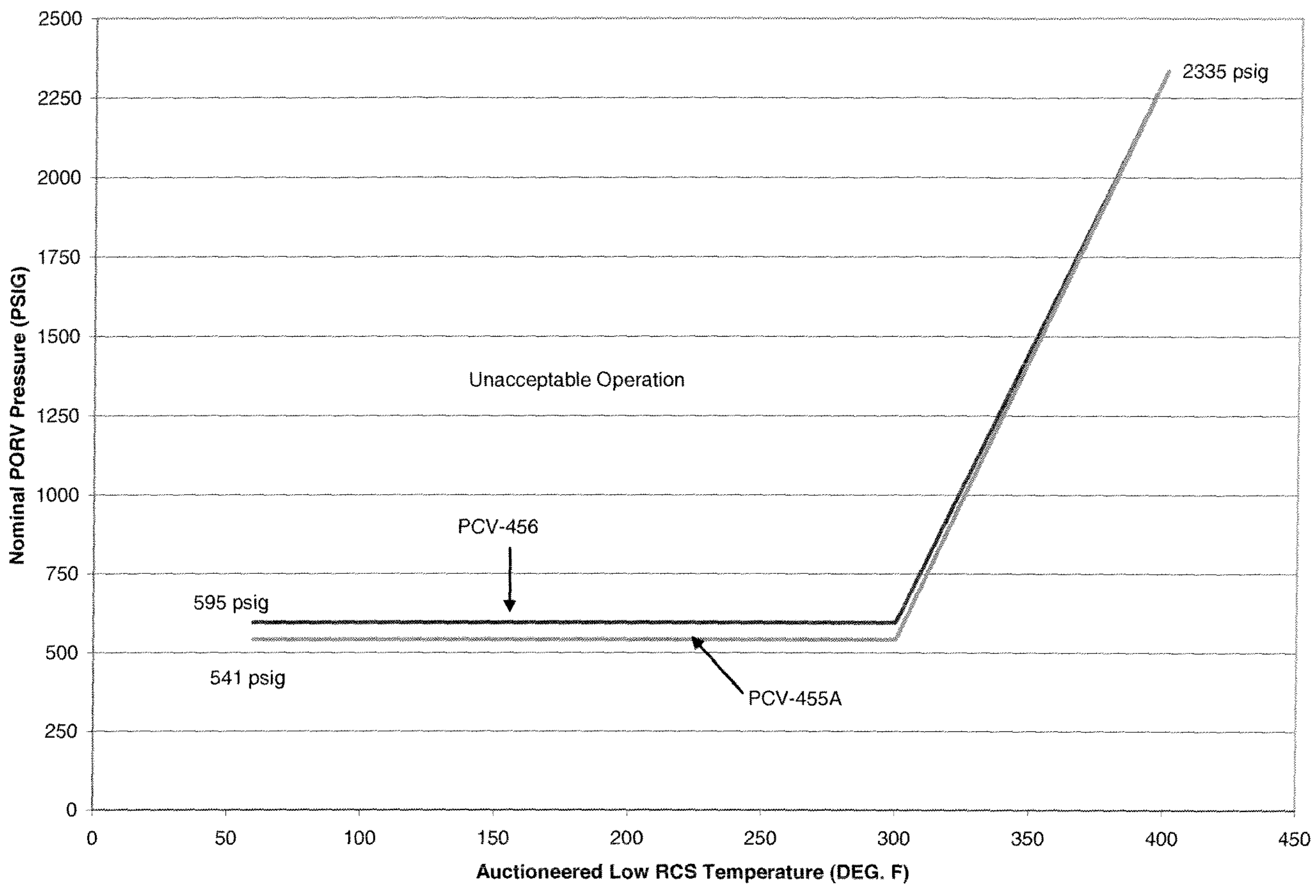

Figure 3.1
Byron Unit 1 Nominal PORV Setpoints for the Low Temperature Overpressure Protection (LTOP) System Applicable for 32 EFPY (Includes Instrumentation Uncertainty)

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 14) 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 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 4.1. The time of specimen withdrawal may be modified to coincide with those refueling outages or reactor shutdowns most closely approaching the withdrawal schedule.

Table 4.1

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	B1R12 [(c)]	-- [(d)]
V	61.0°	3.97	B1R12 [(c)]	-- [(d)]
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 15123, Rev. 1) (Reference 15).
d) Capsule removed and is stored in the spent fuel pool. Capsule has not been analyzed and therefore capsule fluence has not been estimated.

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 (ARTs) 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 Byron Unit 1 reactor vessel material (Intermediate Shell Forging 5P-5933).

Table 5.5 provides RT_{PTS} values for Byron Unit 1 for 32 EFPY obtained from Reference 4.

Table 5.6 provides RT_{PTS} values for Byron Unit 1 for 48 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]	FF [c]	ΔRT_{NDT} [d]	$FF*\Delta RT_{NDT}$	FF^2
Inter. Shell Forging 5P-5933 (Tangential)	U	$4.04x10^{18}$	0.748	28.55	21.36	0.560
	X	$1.57x10^{19}$	1.124	9.82	11.04	1.263
	W	$2.43x10^{19}$	1.239	49.20	60.96	1.535
Inter. Shell Forging 5P-5933 (Axial)	U	$4.04x10^{18}$	0.748	18.52	13.85	0.560
	X	$1.57x10^{19}$	1.124	53.03	59.61	1.263
	W	$2.43x10^{19}$	1.239	29.34	36.35	1.535
				Sum:	203.17	6.716
	$CF_{Forging} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (203.17) \div (6.716) =$ **30.3°F**					
Byron 1 Weld Metal WF-336 (Heat #442002)	U	$4.04x10^{18}$	0.749	11.22 (5.61) [e]	8.40	0.561
	X	$1.57x10^{19}$	1.125	80.22 (40.11) [e]	90.25	1.266
	W	$2.43x10^{19}$	1.239	102.68 (51.34) [e]	127.22	1.535
Byron 2 Weld Metal WF-447 (Heat #442002)	U	$4.05x10^{18}$	0.749	16.88 (8.44) [e]	12.64	0.561
	W	$1.27 x10^{19}$	1.067	57.76 (28.88) [e]	61.63	1.138
	X	$2.30 x 10^{19}$	1.225	108.02 (54.01) [e]	132.32	1.500
				SUM:	432.46	6.561
	$CF = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (432.46) \div (6.561) =$ **65.9°F**					

a) Reference 7, Table 4-9
b) f = Calculated fluence, (x 10^{19} n/cm^2, E > 1.0 MeV)
c) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$.
d) ΔRT_{NDT} values are the measured 30 ft-lb shift values
e) Adjusted ΔRT_{NDT} per Ratio Procedure of Regulatory Guide 1.99, Rev. 2. Ratio = 2.0. See Table 4-9 of WCAP 15391, Rev. 1. (Reference 7).

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 Temperatures (ARTs) at 1/4T and 3/4T Locations for 32 EFPY [(a)]		
Material Description	32 EFPY	
	1/4T ART(°F)	3/4T ART(°F)
Intermediate Shell Forging 5P-5933	95	80
- Using Surveillance Data[(b)]	106[(c)]	97[(c)]
Lower Shell Forging 5P-5951	65	50
Circumferential Weld WF-336	82	52
- Using Credible Surveillance Data[(d)]	67	48
Circumferential Weld WF-501	69	49
- Using Credible Surveillance Data from Braidwood 1 and 2	47	34
Nozzle Shell Forging 123J218	75	59

(a) Fluence, f, is based upon f_{surf} (E>1.0 MeV) = 2.02×10^{19} at 32 EFPY (Reference 7).

(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 (Reference 8).

(c) These ART values were used to generate the Byron Unit 1 32 EFPY heatup and cooldown curves (Reference 7).

(d) Calculated using the chemistry factor from the Byron Unit 1 and 2 integrated surveillance data as reported in WCAP-15391 (Reference 7).

Table 5.4

Byron Unit 1 Calculation of Adjusted Reference Temperatures (ARTs) at 32 EFPY at the Limiting Reactor Vessel Material, Intermediate Shell Forging 5P-5933 (Conservatively Based on Surveillance Capsule Data)[a]

Parameter	Values	
Operating Time	32 EFPY	
Location[b]	1/4T ART(°F)	3/4T ART(°F)
Chemistry Factor, CF (°F)	30.3	30.3
Fluence(f), n/cm^2 (E>1.0 Mev)[c]	1.21×10^{19}	4.37×10^{18}
Fluence Factor, FF	1.053	0.770
ΔRT_{NDT}= CFxFF (°F)	31.9	23.3
Initial RT $_{NDT,}$, I (°F)	40	40
Margin, M (°F)	34	34
ART= I+(CF*FF)+M,°F per RG 1.99, Revision 2	106	97

a) WCAP 15123 (Reference 15)
b) The Byron Unit 1 reactor vessel wall thickness is 8.5 inches at the beltline region.
c) Fluence, f, is based upon f_{surf} (E>1.0 Mev) = 2.02×10^{19} at 32 EFPY (Reference 7).

Table 5.5

RT_{PTS} Calculation for Byron Unit 1 Beltline Region Materials at EOL (32 EFPY) [(a)]

Material	Fluence (10^{19} n/cm^2, E>1.0 MeV)	FF	CF (°F)	ΔRT_{PTS} [(b)] (°F)	Margin (°F)	$RT_{NDT(U)}$ [(c)] (°F)	RT_{PTS} [(d)] (°F)
Intermediate Shell Forging 5P-5933	1.95×10^{19}	1.18	26.0	30.7	30.7	40	101
Intermediate Shell Forging 5P-5933 using S/C Data[(e)]	1.95×10^{19}	1.18	30.3	35.8	34	40	110
Lower shell Forging 5P-5951	1.95×10^{19}	1.18	26.0	30.7	30.7	10	71
Inter. To Lower Shell Circ. Weld Metal WF-336 (442002)	1.95×10^{19}	1.18	54.0	63.7	56	-30	90
Inter. To Lower Shell Circ. Weld Metal (442002) using S/C Data[(f)]	1.95×10^{19}	1.18	65.9	67.6	28	-30	66
Nozzle Shell Forging 123J218	5.83×10^{18}	0.849	31.0	26.3	26.3	30	83
Nozzle Shell to Inter. Shell Circ. Weld Metal WF-501 (442011)	5.83×10^{18}	0.849	41.0	34.8	34.8	10	80
Nozzle Shell to Inter. Shell Circ. Weld Metal (442011) using S/C Data	5.83×10^{18}	0.849	16.7	14.2	14.2	10	38

(a) Limiting RT_{PTS} is significantly less than the PTS Screening Criteria of 270 °F.
(b) ΔRT_{PTS} = CF * FF
(c) Initial RT_{NDT} values are measured values.
(d) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS}$ + Margin (°F)
(e) 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.
(f) Based on Byron Unit 1 and 2 integrated surveillance data chemistry factor from WCAP-15178 (Reference 9).

Table 5.6

RT_{PTS} Calculation for Byron Unit 1 Beltline Region Materials at Life Extension (48 EFPY)[a,b]

Material	Fluence (10^{19} n/cm², E>1.0 MeV)	FF	CF (°F)	ΔRT_{PTS}[c] (°F)	Margin (°F)	$RT_{NDT(U)}$[d] (°F)	RT_{PTS}[e] (°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[f]	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[g]	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

(a) The fluence for 48 EFPY (Reference 4) did not incorporate the 5% increase. However, this fluence value is greater than the end-of-life fluence (32 EFPY).
(b) Limiting RT_{PTS} is significantly less than the PTS Screening Criteria of 270 °F.
(c) $\Delta RT_{PTS} = CF * FF$
(d) Initial RT_{NDT} values are measured values.
(e) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS}$ + Margin (°F)
(f) 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.
(g) Based on Byron Unit 1 and 2 integrated surveillance data chemistry factor from WCAP-15178 (Reference 9).

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 Commonwealth Edison Company, CAE-06-90/CCE-06-86, "Transmittal of Byron and Braidwood Units 1 and 2 Revision 1 LTOPS Setpoints Analysis Reports for 22 and 32 EFPY (LTR-SCS-03-87, Revision 1 Attachment A) (LTR-SCS-03-87, Revision 1 Attachment B)," August 28, 2006.

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

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

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

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

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