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

Byron Station, Unit 2
Facility Operating License No. NPF- 66
NRC Docket No. STN 50-455

Subject: Byron Station Unit 2 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 2 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 2 PTLR

DMH/JEL/rah

ATTACHMENT

Byron Station Unit 2 Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)

BYRON UNIT 2

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

(December 2006)

BYRON - UNIT 2
PRESSURE AND TEMPERATURE LIMITS REPORT

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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 1 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 32 EFPY: 1/4T, 107°F (N-588) & 52°F ('96 App. G)

3/4T, 89°F (N-588) & 37°F ('96 App. G)

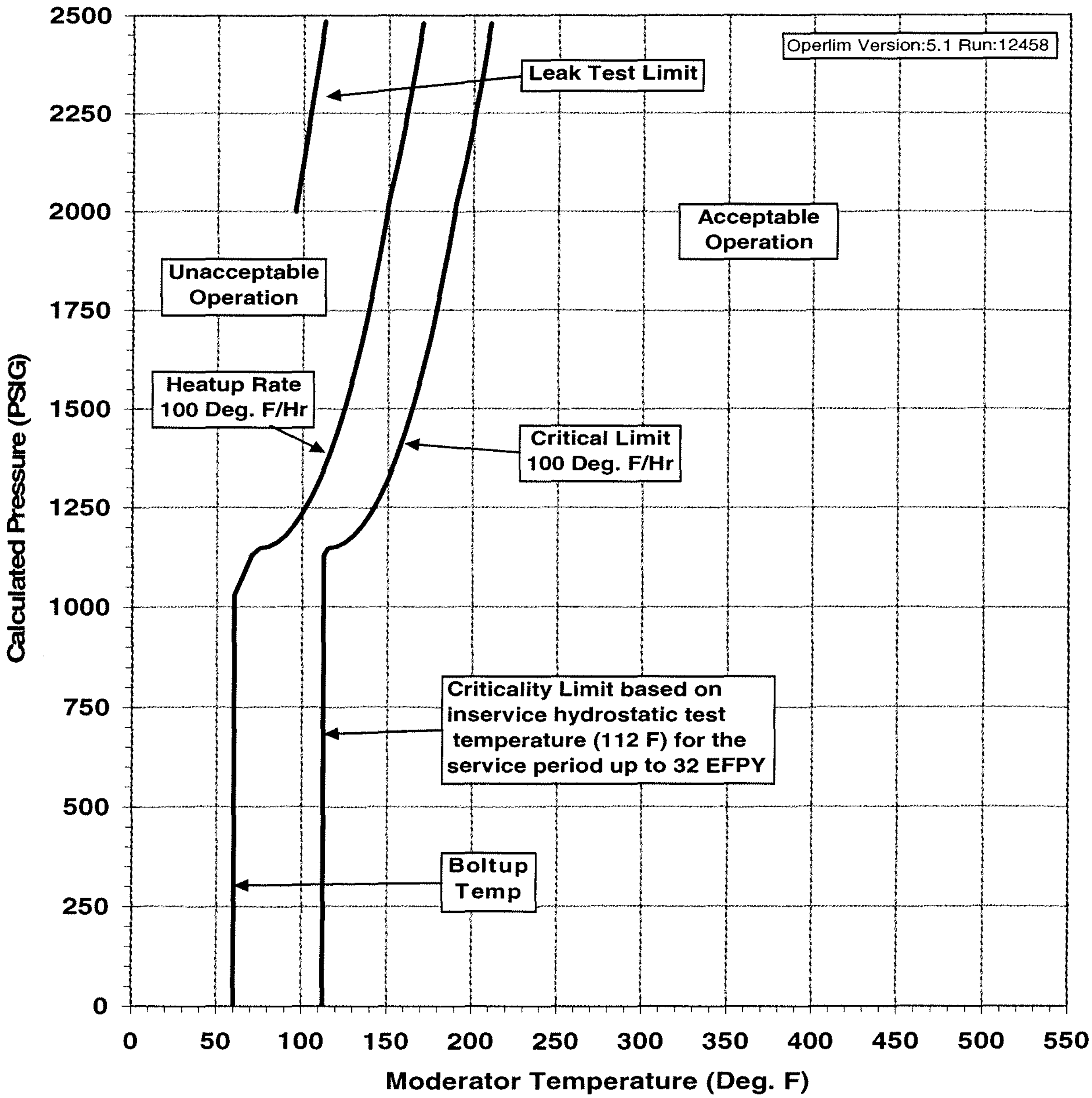

Figure 2.1
Byron Unit 2 Reactor Coolant System Heatup Limitations (Heatup rates of 100°F/hr) Applicable for 32 EFPY (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: CIRCUMFERENTIAL WELD WF-447 & NOZZLE SHELL FORGING
LIMITING ART VALUES AT 32 EFPY: 1/4T, 107°F (N-588) & 52°F ('96 App. G)
3/4T, 89°F (N-588) & 37°F ('96 App. G)

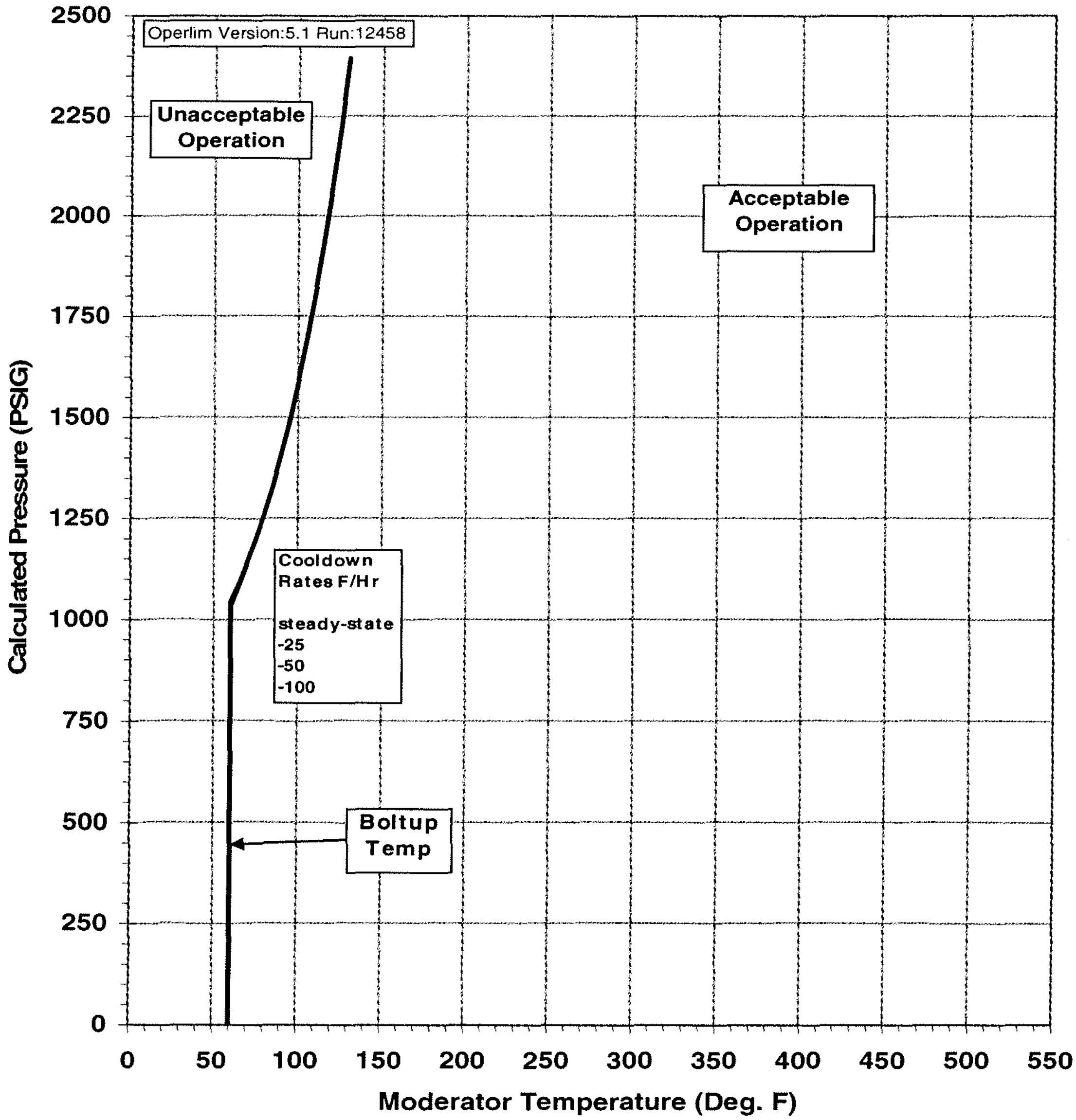

Figure 2.2
Byron Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50 and 100°F/hr) Applicable for 32 EFPY (Without Margins for Instrumentation Errors)

Table 2.1a
Byron Unit 2 Heatup Data Points at 32 EFPY
(Without Margins for Instrumentation Errors)

Heatup Curve					
100 F Heatup		Criticality Limit		Leak Test Limit	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
60	1030	112	0	95	2000
65	1078	112	1078	112	2485
70	1128	112	1128		
75	1148	115	1148		
80	1152	120	1152		
85	1162	125	1162		
90	1180	130	1180		
95	1205	135	1205		
100	1237	140	1237		
105	1276	145	1276		
110	1323	150	1323		
115	1377	155	1377		
120	1440	160	1440		
125	1511	165	1511		
130	1592	170	1592		
135	1682	175	1682		
140	1784	180	1784		
145	1897	185	1897		
150	2023	190	2023		
155	2120	195	2120		
160	2227	200	2227		
165	2347	205	2347		
170	2480	210	2480		

Table 2.1b
Byron Unit 2 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	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).

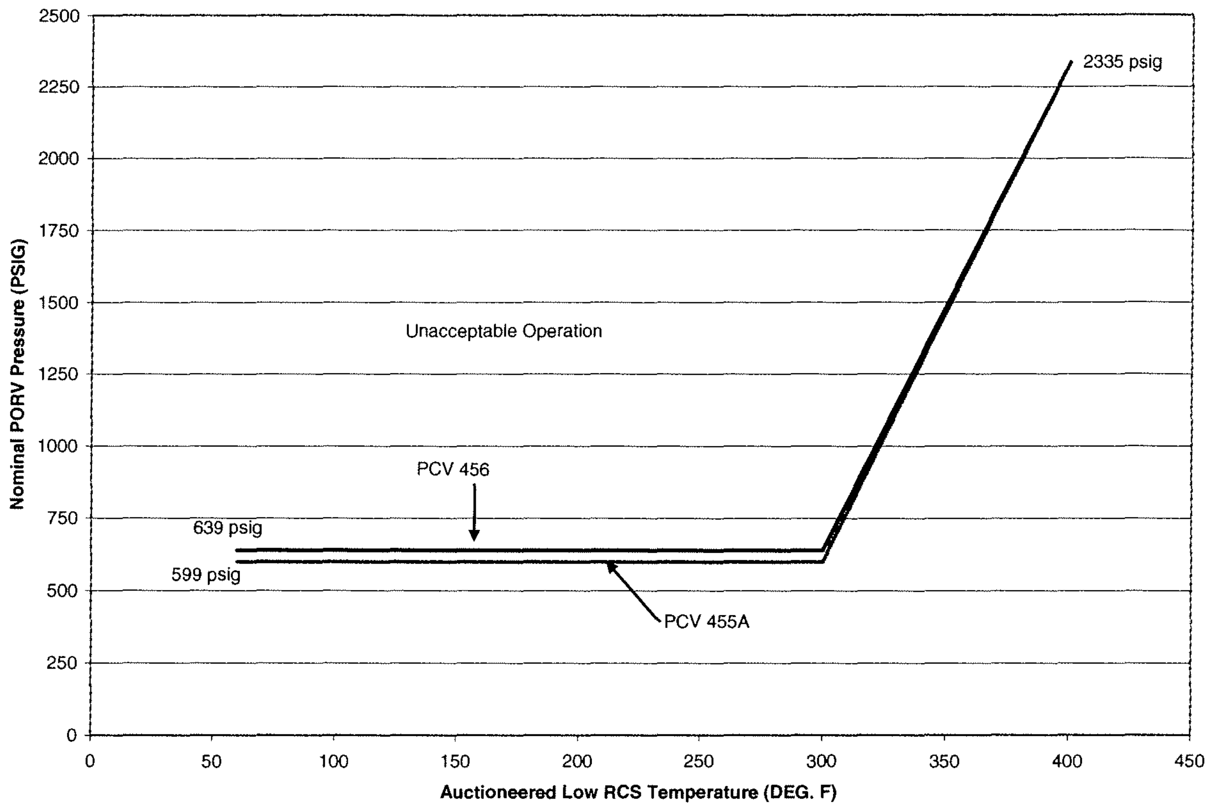

Figure 3.1
Byron Unit 2 Nominal PORV Setpoints for the Low Temperature Overpressure Protection (LTOP) System Applicable for the 32 EFPY (Includes Instrumentation Uncertainty)

Table 3.1
Data Points for Byron Unit 2 Nominal PORV Setpoints for the LTOP System Applicable for 32 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 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 2 Capsule Withdrawal Schedule				
Capsule	Vessel Location (Degrees)	Capsule Lead Factor	Removal Time[(a)] (EFPY)	Estimated Capsule Fluence (n/cm^2)
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)	2.30×10^{19} [(b)]
Z	301.5°	4.21	B2R11 [(c)]	-- [(d)]
V	61.0°	3.97	B2R11 [(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 2.06×10^{19} n/cm^2.
c) Standby capsule to be used for future license renewal (derived from Table 7-1 of WCAP 15176, Reference 9).
d) Capsule has been removed and stored in the spent fuel pool. Since the capsule has not been analyzed the 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 2 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 2 reactor vessel material, i.e. weld metal HT # 442002, (Based on Surveillance Capsule Data).

Table 5.5 provides RT_{PTS} values for Byron Unit 2 for 32 EFPY obtained from Reference 5.

Table 5.6 provides RT_{PTS} values for Byron Unit 2 for 48 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]	FF[c]	ΔRT_{NDT} [d]	$FF*\Delta RT_{NDT}$	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
	$CF_{Forging} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (119.66) \div (6.398) = \mathbf{18.7°F}$					
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
	$CF = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (432.46) \div (6.561) = \mathbf{65.9°F}$					

a) Reference 7, Table 4-8
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 taken from Ref. 9
e) Adjusted ΔRT_{NDT} per Ratio Procedure of Regulatory Guide 1.99, Rev. 2. Ratio = 2.0. See Table 4-9 of WCAP 15392, Revision 2. (Reference 7).

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 49D329-1-1/49C297-1-1	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 7.
b) Initial RT_{NDT} values are based on measured data.

Table 5.3

Summary of Byron Unit 2 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 49D329-1/49C297-1 (RG Position 1 [(b)])	22	11
Lower Shell Forging 49D330-1/49C298-1 (RG Position 1[(b)])	53	37
Using capsule data (RG Position 2[(b)])	17	9
Circumferential Weld WF-447 (HT# 442002) (RG Position 1[(b)])	123	93
Using credible surveillance capsule data (RG Position 2[(b)])	107	89
Nozzle Shell Forging 4P-6107 (RG Position 1 [(b)])	52[(c)]	37[(c)]
Nozzle Shell to Intermediate Shell Weld WF-562 (HT # 442011)	96	76
Using credible surveillance capsule data (RG Position 2[(b)])	75	63

(a) Fluence, f, is based upon f_{surf} (E>1.0 Mev) = 2.06×10^{19} at 32 EFPY, Reference 7.
(b) Calculated using a chemistry factor based on Regulatory Guide (RG) 1.99, Positions 1 and 2, as reported in WCAP-15392, Revision 2 (Reference 7).
(c) These ART values were used to generate the Byron Unit 2 Heatup and Cooldown Curves, WCAP-15392 Revision 2 (Reference 7).

Table 5.4

Byron Unit 2 Calculation of Adjusted Reference Temperatures (ARTs) at 32 EFPY at the Limiting Reactor Vessel Material, Nozzle Shell Forging 4P-6107 (a)

Parameter	Values	
Operating Time	32 EFPY	
Location(b)	1/4T ART(°F)	3/4T ART(°F)
Chemistry Factor, CF (°F)	31.0	31.0
Fluence(f), n/cm^2 (E>1.0 Mev)(c)	$3.13x10^{18}$	$1.13x10^{18}$
Fluence Factor, FF	0.681	0.442
ΔRT_{NDT}= CFxFF (°F)	21.1	13.7
Initial RT $_{NDT,}$, I (°F)	10	10
Margin, M (°F)	21.1	13.7
ART= I+(CF*FF)+M,°F per RG 1.99, Revision 2	52	37

(a) WCAP 15392, Revision 2, Reference 7.
(b) The Byron Unit 2 reactor vessel wall thickness is 8.5 inches at the beltline region.
(c) Fluence, f, is based upon f_{surf} (E>1.0 Mev) = $2.06x10^{19}$ at 32 EFPY, Reference 7.

Table 5.5

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

Material	Fluence (10^{19}n/cm^2, 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	$2.06 * 10^{19}$	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[(f)]	$2.06 * 10^{19}$	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[(f)]	$2.03 * 10^{19}$	1.19	65.9	77.8	28	10	116
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[(f)]	$5.22 * 10^{18}$	0.818	16.7	13.5	13.5	40	67

(a) Fluence projections for 32 EFPY from Byron 2 PTS report, WCAP-15177 (Reference 5)
(b) Limiting RT_{PTS} is significantly less than the PTS Screening Criteria of 300 °F.
(c) $\Delta RT_{PTS} = CF * FF$
(d) Initial RT_{NDT} values are measured values (See Table 5.2)
(e) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS} + \text{Margin}$ (°F)
(f) Calculated using a CF based on surveillance capsule data per RG 1.99, Position 2.

Table 5.6

RT_{PTS} Calculation for Byron Unit 2 Beltline Region Materials at Life Extension (48 EFPY) [a,b]

Material	Fluence (10^{19} n/cm^2, 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	$2.98 * 10^{19}$	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[f]	$2.98 * 10^{19}$	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[f]	$2.93 * 10^{19}$	1.29	65.9	85	28	10	123
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[f]	$7.53*10^{18}$	0.920	16.7	15.4	15.4	40	71

(a) The fluence for 48 EFPY (Reference 5) 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 300 °F.
(c) $\Delta RT_{PTS} = CF * FF$
(d) Initial RT_{NDT} values are measured values (See Table 5.2)
(e) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS} + \text{Margin}$ (°F)
(f) Calculated using a CF based on surveillance capsule data per RG 1.99, Position 2.

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

5. WCAP-15177, "Evaluation of Pressurized Thermal Shock for Byron Unit 2," Revision 0, T. J. Laubham, et al., September 2000.

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.