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NRC Docket No. STN 50-454

Subject: Byron Station Unit 1 Reactor Coolant System Pressure and Temperature Limits Report

In accordance with Technical Specification 5.6.6, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)", we are submitting the Unit 1 PTLR.

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

Respectfully,

John J. Kowalski
Site Vice President
Byron Generating Station

Attachment: Byron Station Unit 1 RCS PTLR

JJK/LZ/mf

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

BYRON UNIT 1

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

(October 2020)

BYRON - UNIT 1
PRESSURE AND TEMPERATURE LIMITS REPORT

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BYRON - UNIT 1
PRESSURE AND TEMPERATURE LIMITS REPORT

1.0 Introduction

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-A, Revision 4 (Reference 1) was used with the following exceptions:

a) Elimination of the flange requirements documented in WCAP-16143-P.
b) The initial reference temperature of the outlet nozzle forging to shell weld (WF-419) is determined using BAW-2308 in lieu of the ASME NB-2300 requirements.

WCAP-18371-NP, Revision 0, Reference 4, 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 "Master Curve" fracture toughness properties from BAW-2308 Revision 1-A Safety Evaluation (SE) and Revision 2-A SE (Reference 2) are used for one outlet nozzle to upper shell forgings weld. WCAP-16143-P (Reference 5) documents the technical basis for the elimination of the flange requirements. These exceptions to the methodology in WCAP-14040-A, Revision 4 have been reviewed and accepted by the NRC in References 6, 7, and 10.

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 4 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 were developed in WCAP-18371-NP, Rev. 0 (Reference 4) using the limiting material between Byron Units 1 and 2. This approach is conservative. 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, 1998 Edition through 2000 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: BYRON UNIT 1 INTERMEDIATE SHELL FORGING 5P-5933
LIMITING ART VALUES AT 57 EFFECTIVE FULL POWER YEARS (EFPY): 1/4T, 102°F
3/4T, 87°F

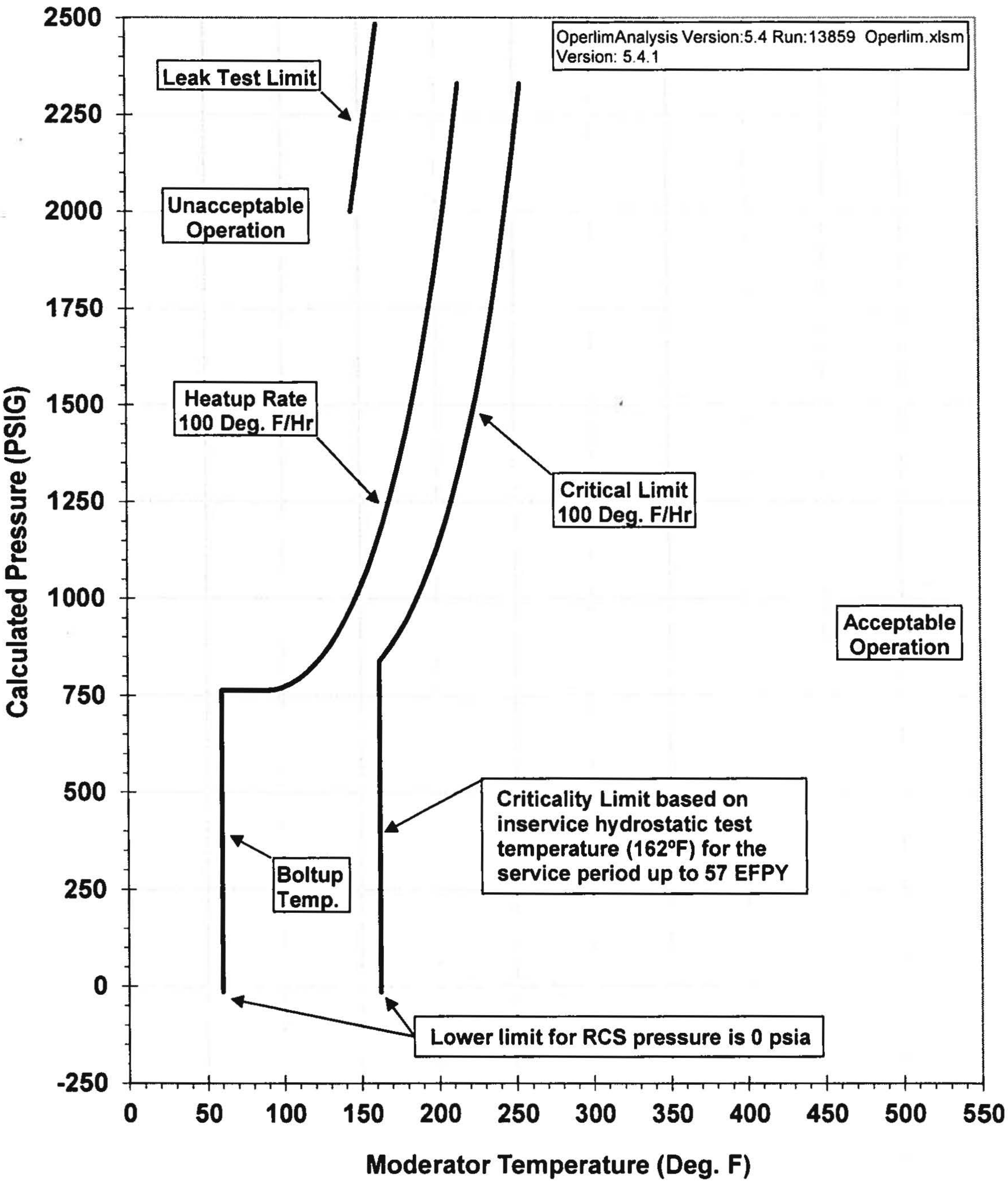

Figure 2.1
Byron Unit 1 Reactor Coolant System Heatup Limitations (Heatup rates of 100°F/hr) Applicable for 57 EFPY (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: BYRON UNIT 1 INTERMEDIATE SHELL FORGING 5P-5933

LIMITING ART VALUES AT 57 EFPY: 1/4T, 102°F
3/4T, 87°F

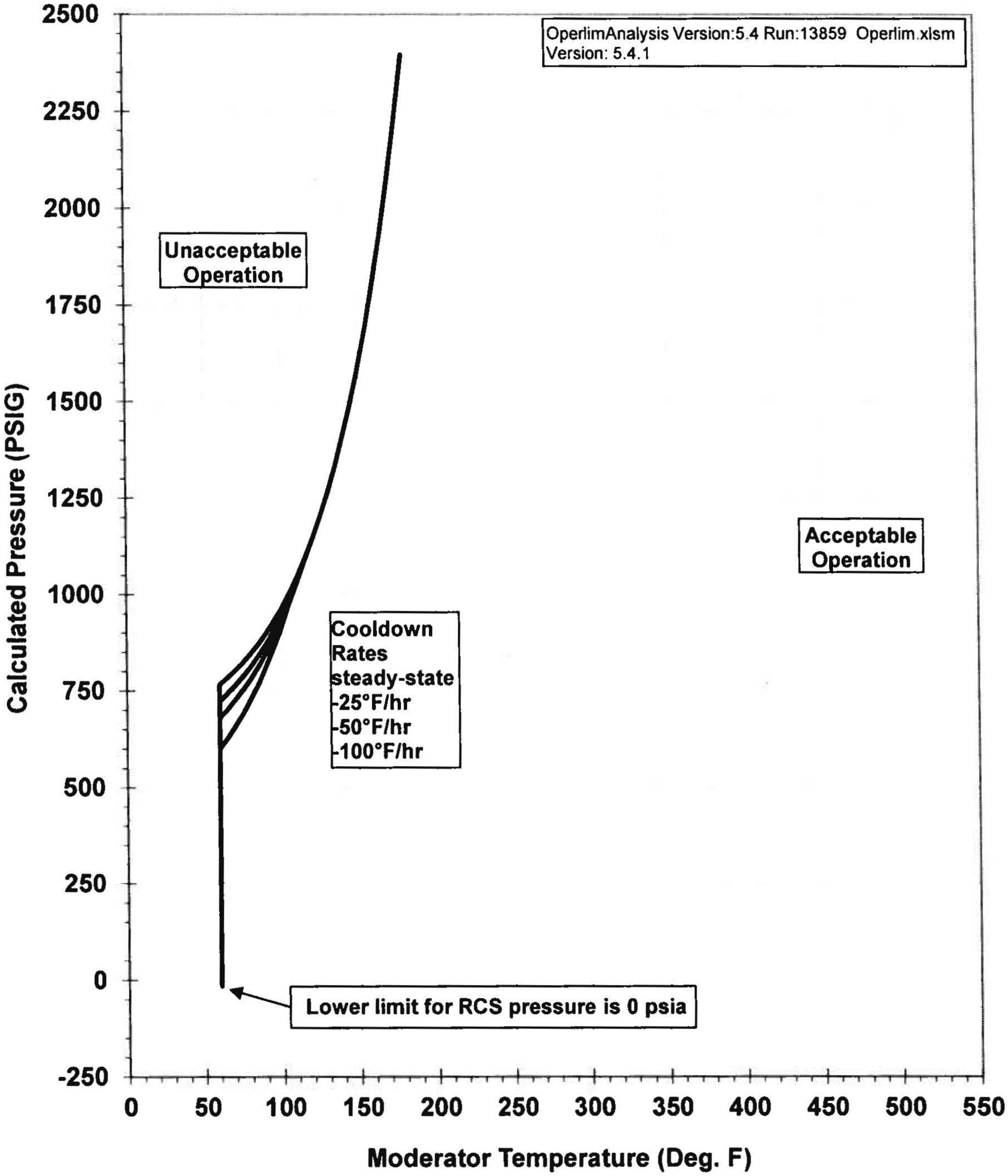

Figure 2.2
Byron Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown rates of 0, 25, 50 and 100°F/hr) Applicable for 57 EFPY (Without Margins for Instrumentation Errors)

Table 2.1a
Byron Unit 1 Heatup Data Points at 57 EFPY
(Without Margins for Instrumentation Errors)

Heatup Curve					
100 F Heatup		Criticality Limit		Leak Test Limit	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
60	Note 1	162	Note 1	145	2000
60	764	162	839	162	2485
65	764	165	851		
70	764	170	877		
75	764	175	908		
80	764	180	943		
85	764	185	982		
90	764	190	1027		
95	766	195	1076		
100	772	200	1132		
105	781	205	1194		
110	793	210	1263		
115	809	215	1339		
120	828	220	1424		
125	851	225	1518		
130	877	230	1622		
135	908	235	1737		
140	943	240	1864		
145	982	245	2004		
150	1027	250	2159		
155	1076	255	2330		
160	1132				
165	1194				
170	1263				
175	1339				
180	1424				
185	1518				
190	1622				
195	1737				
200	1864				
205	2004				
210	2159				
215	2330				

Note:

1. The minimum acceptable pressure is 0 psia.

Table 2.1b
Byron Unit 1 Cooldown Data Points at 57 EFPY
(Without Margins for Instrumentation Errors)

Cooldown Curves							
Steady State		25 °F Cooldown		50 °F Cooldown		100 °F Cooldown	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
60	Note 1	60	Note 1	60	Note 1	60	Note 1
60	766	60	723	60	681	60	601
65	783	65	742	65	702	65	628
70	802	70	763	70	726	70	658
75	823	75	786	75	752	75	691
80	846	80	812	80	780	80	727
85	871	85	840	85	812	85	768
90	900	90	872	90	848	90	814
95	931	95	907	95	887	95	864
100	965	100	945	100	930	100	920
105	1003	105	988	105	979	105	979
110	1045	110	1035	110	1032	110	1032
115	1092	115	1088	115	1088	115	1088
120	1143	120	1143	120	1143	120	1143
125	1200	125	1200	125	1200	125	1200
130	1263	130	1263	130	1263	130	1263
135	1332	135	1332	135	1332	135	1332
140	1409	140	1409	140	1409	140	1409
145	1494	145	1494	145	1494	145	1494
150	1587	150	1587	150	1587	150	1587
155	1691	155	1691	155	1691	155	1691
160	1805	160	1805	160	1805	160	1805
165	1932	165	1932	165	1932	165	1932
170	2071	170	2071	170	2071	170	2071
175	2226	175	2226	175	2226	175	2226
180	2396	180	2396	180	2396	180	2396

Note:

1. The minimum acceptable pressure is 0 psia.

3.0 Low Temperature Overpressure Protection and Boltup

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

3.1 LTOP System Setpoints (LCO 3.4.12)

Two PORVs shall have maximum lift settings in accordance with Figure 3.1 and Table 3.1. These settings are based on the LTOP calculation in Reference 3.

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 lift settings shown in Figure 3.1 and Table 3.1 account for appropriate instrument error.

3.2 LTOP Enable Temperature

Byron Unit 1 procedures governing the heatup and cooldown of the RCS require the arming of the LTOP system for RCS temperature less than 350°F and disarming of the LTOP system for RCS temperature of 350°F and above.

Note that the last LTOP PORV segment in Table 3.1 extends to 400°F where the pressure setpoint is 2335 psig. This is intended to prohibit PORV lift for an inadvertent LTOP system arming at power.

3.3 Reactor Vessel Boltup Temperature (Non-Technical Specification)

The minimum boltup temperature for the Reactor Vessel Flange shall be ≥ 60°F. Boltup is a condition in which the Reactor Vessel head is installed with tension applied to any stud, and with the RCS vented to atmosphere.

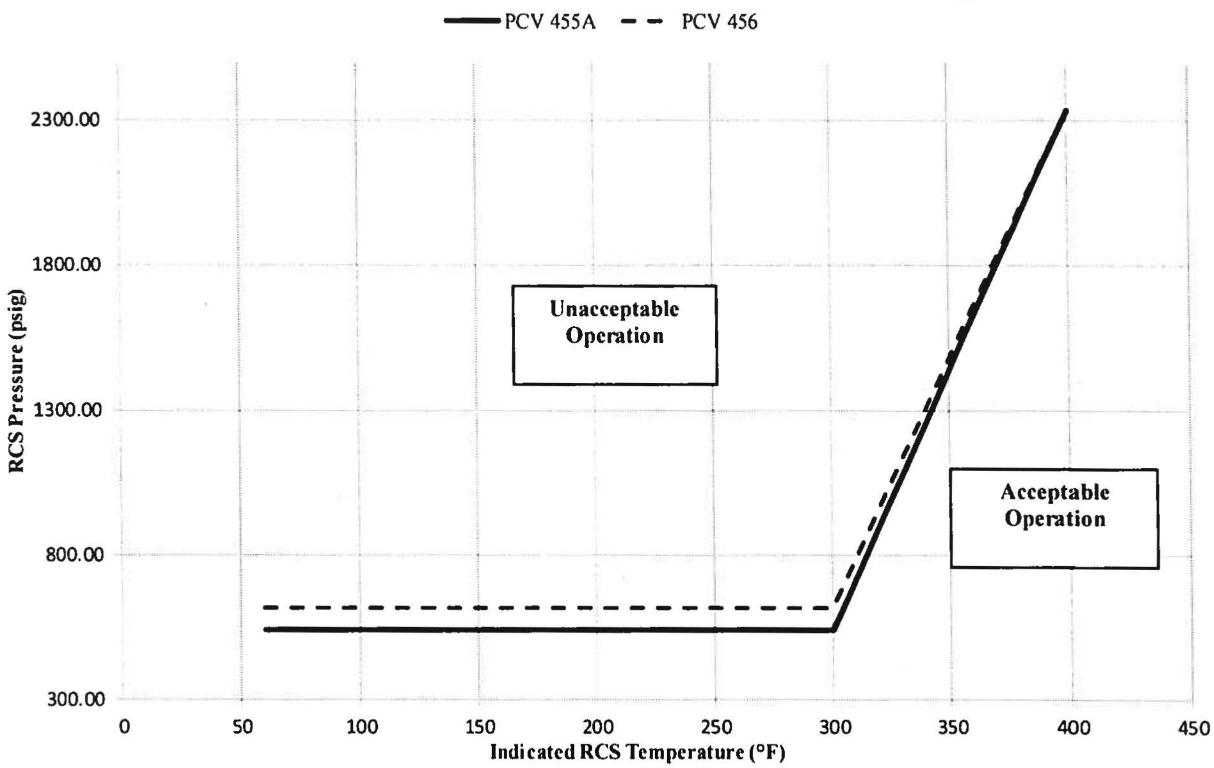

Figure 3.1
Byron Unit 1 Nominal PORV Setpoints for the Low Temperature Overpressure Protection (LTOP) System Applicable for 57 EFPY (Includes Instrumentation Uncertainty)

Table 3.1
Data Points for Byron Unit 1 Nominal PORV Setpoints for the LTOP System Applicable for 57 EFPY (Includes Instrumentation Uncertainty)

PCV-455A

(1TY-0413M)

AUCTIONEERED LOW RCS TEMP. (DEG. F)	RCS PRESSURE (PSIG)
60	541
300	541
310	720
320	899
330	1079
340	1258
350	1438
360	1617
400	2335

PCV-456

(1TY-0413P)

AUCTIONEERED LOW RCS TEMP. (DEG. F)	RCS PRESSURE (PSIG)
60	618
300	618
310	789
320	961
330	1133
340	1304
350	1476
360	1648
400	2335

Note: 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 8) 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 fourth reactor vessel material irradiation surveillance specimens (Capsule Y) have been removed and analyzed to determine changes in the reactor vessel material properties. The surveillance capsule testing has been completed for the licensed operating period. The remaining two capsules, V 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°	4.03	1.18 (EOC 1)[d]	0.409×10^{19}
X	238.5°	4.08	5.67 (EOC 5)	1.49×10^{19}
W	121.5°	4.08	9.27 (EOC 8)	2.26×10^{19}
Y	241.0°	3.87	18.81 (EOC 15)	3.97×10^{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}

Notes:

(a) Source document is WCAP-18054-NP (Reference 9), Table 7-1.
(b) EFPY from plant startup.
(c) Standby Capsules Z and V were removed and placed in the spent fuel pool. No testing or analysis has been performed on these capsules.
(d) EOC = end-of-cycle.

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 57 EFPY.

Table 5.4 provides the Reference Temperature for Pressurized Thermal Shock (RT_{PTS}) values for Byron Unit 1 for 57 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	0.409×10^{19}	0.752	28.7	21.58	0.57
	X	1.49×10^{19}	1.110	18.3	20.32	1.23
	W	2.26×10^{19}	1.221	49.5	60.42	1.49
	Y	3.97×10^{19}	1.355	27.8	37.66	1.83
Intermediate Shell Forging (Axial)	U	0.409×10^{19}	0.752	18.6	13.99	0.57
	X	1.49×10^{19}	1.110	54.6	60.63	1.23
	W	2.26×10^{19}	1.221	29.5	36.01	1.49
	Y	3.97×10^{19}	1.355	11.7	15.85	1.83
				SUM:	266.46	10.25
	$CF_{IS\ Forging} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (266.46) \div (10.25) = \mathbf{26.0°F}$					
Byron Unit 1 Surveillance Weld Material (Heat #442002)	U	0.409×10^{19}	0.752	10.4 (5.2)	7.82	0.57
	X	1.49×10^{19}	1.110	80.2 (40.1)	89.06	1.23
	W	2.26×10^{19}	1.221	101.2 (50.6)	123.54	1.49
	Y	3.97×10^{19}	1.355	153.4 (76.7)	207.79	1.83
Byron Unit 2 Surveillance Weld Material (Heat #442002)	U	0.406×10^{19}	0.750	17.4 (8.7)	13.05	0.56
	W	1.21×10^{19}	1.053	57.6 (28.8)	60.66	1.11
	X	2.18×10^{19}	1.211	108.4 (54.2)	131.32	1.47
	Y	4.19×10^{19}	1.366	117.4 (58.7)	160.39	1.87
				SUM:	793.63	10.13
	$CF_{Weld\ Metal} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (793.63) \div (10.13) = \mathbf{78.3°F}$					

Notes:

a) Source document is WCAP-18371-NP (Reference 4), Table 5-1 and Table 5-3.

b) f = fluence; ΔRT_{NDT} values are the measured 30 ft-lb shift values taken from References 9 and 11. ΔRT_{NDT} values for the surveillance weld data are adjusted by a ratio of 2.0 to account for chemistry differences between the surveillance weld and the vessel weld. (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 (%)	Chemistry Factor	Initial RT_{NDT} (°F)[b]
Closure Head Flange, Heat # 124K358VA1	--	0.74	--	60
Vessel Flange, Heat # 123J219VA1	--	0.73	--	10
Inlet Nozzle 03-001, Heat # 1V4684-3V1320	0.12	0.82	86[d]	-10
Inlet Nozzle 03-002, Heat # 1V4684-3V1320	0.12	0.82	86[d]	-20
Inlet Nozzle 04-001, Heat # 1V4695	0.13	0.79	95.8[d]	-20
Inlet Nozzle 04-002, Heat # 1V4695	0.12	0.78	85.7[d]	-20
Outlet Nozzle 01-001, Heat # 1V4656	0.11	0.84	77[d]	0
Outlet Nozzle 01-002, Heat # 1V4656	0.11	0.84	77[d]	-20
Outlet Nozzle 02-001, Heat # 2V2557	0.11	0.85	77[d]	-20
Outlet Nozzle 02-002, Heat # 2V2557	0.11	0.84	77[d]	-10
Nozzle Shell Forging, Heat # 123J218	0.05	0.72	31[d]	30
Intermediate Shell Forging, Heat # 5P-5933	0.04	0.74	26[d]	40
Lower Shell Forging, Heat # 5P-5951	0.04	0.64	26[d]	10
Intermediate to Lower Shell Forging Circ. Weld Seam WF-336 (Heat # 442002)	0.04	0.63	54[d], 78.3[e]	-30
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-501 (Heat # 442011)	0.03	0.67	41[d], 31.2[e][f]	10
Byron Unit 1 Surveillance Program Weld Metal (Heat # 442002)	0.02	0.69	27[d]	--
Byron Unit 2 Surveillance Program Weld Metal (Heat # 442002)	0.02	0.71	27[d]	--
Braidwood Units 1 & 2 Surveillance Program Weld Metals (Heat # 442011)	0.03	0.67, 0.71	41[d]	--
Inlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-337 (Heat # 442002)	0.15	0.56	139.2[d][g]	-10
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-419 (Heat # 1P5412)	0.178	0.69	168.3	-48.6[c]
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-406 (Heat # 504)	0.054	0.80	73.6[d]	10

Notes contained on the following page:

Notes:

a) Data taken from Reference 4.
b) The initial RT_{NDT} values for the forgings and welds are based on measured data.
c) Generic value taken from BAW-2308 (Reference 2).
d) Chemistry Factor calculated for Cu and Ni values per Regulatory Guide 1.99, Rev. 2, Position 1.1.
e) Chemistry Factor calculated per Regulatory Guide 1.99, Rev. 2, Position 2.1.
f) The Position 2.1 CF uses credible surveillance data from Braidwood in WCAP-18370-NP (Reference 12).
g) The surveillance weld material is not representative of this weld material even though the two welds share the same material heat number, flux type, and lot number. The reactor vessel beltline and surveillance welds have low weight-percent copper values due to restrictions on copper content in the beltline region; whereas, the nozzle circumferential weld seams do not have the low weight percent copper restrictions. The embrittlement behavior for these two welds would not be the same. Therefore, surveillance weld data will not be applied to this weld.

Table 5.3

Summary of Byron Unit 1 Adjusted Reference Temperature (ART) Values at 1/4T and 3/4T Locations for 57 EFPY [(a)]

Reactor Vessel Material	Surface Fluence (n/cm^2, E > 1.0 MeV)	57 EFPY	
		1/4T ART (°F)	3/4T ART (°F)
Inlet Nozzle 03-001	1.33×10^{17}	12.8[(b)]	
Inlet Nozzle 03-002	1.33×10^{17}	2.8[(b)]	
Inlet Nozzle 04-001	1.33×10^{17}	5.4[(b)]	
Inlet Nozzle 04-002	1.33×10^{17}	2.7[(b)]	
Outlet Nozzle 01-001	1.01×10^{17}	17.0[(b)]	
Outlet Nozzle 01-002	1.01×10^{17}	-3.0[(b)]	
Outlet Nozzle 02-001	1.01×10^{17}	-3.0[(b)]	
Outlet Nozzle 02-002	1.01×10^{17}	7.0[(b)]	
Nozzle Shell Forging	1.15×10^{19}	85.6	68.6
Intermediate Shell Forging	3.19×10^{19}	101.2	86.6
→ Using non-credible surveillance data	3.19×10^{19}	101.2	86.6
Lower Shell Forging	3.18×10^{19}	71.2	56.6
Nozzle to Intermediate Shell Forging Circ. Weld Seam (Heat # 442011)	1.15×10^{19}	83.5	61.1
→ Using credible Braidwood Units 1 and 2 surveillance data	1.15×10^{19}	66.0	48.9
Intermediate to Lower Shell Forging Circ. Weld Seam (Heat # 442002)	3.07×10^{19}	89.1	65.6
→ Using credible surveillance data	3.07×10^{19}	89.4	67.3
Inlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-337 (Heat # 442002)	1.33×10^{17}	26.9[(b)]	
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-419 (Heat # 1P5412)	1.01×10^{17}	36.6[(b)]	
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-406 (Heat # 504)	1.01×10^{17}	26.2[(b)]	

Notes contained on the following page:

Notes:

(a) The source document containing detailed calculations is WCAP-18371-NP (Reference 4), Table 7-1, Table 7-3, Table 7-4 and Table 7-5.

(b) The ART values for the extended beltline materials are conservatively calculated at the surface, i.e. without attenuation of the fluence.

Table 5.4

RT_{PTS} Calculation for Byron Unit 1 Beltline Region Materials at End-of-Life Extension (EOLE) (57 EFPY) [a,b]

Reactor Vessel Material	R.G. 1.99, Rev. 2 Position	CF (°F)	Fluence (n/cm^2, E > 1.0 MeV)	FF	IRT_{NDT} [c] (°F)	ΔRT_{NDT} (°F)	σ_u [c] (°F)	σ_Δ [d] (°F)	Margin	RT_{PTS} (°F)
Inlet Nozzle 03-001	1.1	86	1.33×10^{17}	0.133	-10	11.4	0	5.7	11.4	12.8
Inlet Nozzle 03-002	1.1	86	1.33×10^{17}	0.133	-20	11.4	0	5.7	11.4	2.8
Inlet Nozzle 04-001	1.1	95.8	1.33×10^{17}	0.133	-20	12.7	0	6.4	12.7	5.4
Inlet Nozzle 04-002	1.1	85.7	1.33×10^{17}	0.133	-20	11.4	0	5.7	11.4	2.7
Outlet Nozzle 01-001	1.1	77	1.01×10^{17}	0.110	0	8.5	0	4.3	8.5	17.0
Outlet Nozzle 01-002	1.1	77	1.01×10^{17}	0.110	-20	8.5	0	4.3	8.5	-3.0
Outlet Nozzle 02-001	1.1	77	1.01×10^{17}	0.110	-20	8.5	0	4.3	8.5	-3.0
Outlet Nozzle 02-002	1.1	77	1.01×10^{17}	0.110	-10	8.5	0	4.3	8.5	7.0
Nozzle Shell Forging	1.1	31	1.15×10^{19}	1.039	30	32.2	0	16.1	32.2	94.4
Intermediate Shell Forging	1.1	26	3.19×10^{19}	1.305	40	33.9	0	17.0	33.9	107.9
→ Using non-credible surveillance data	2.1	26.0	3.19×10^{19}	1.305	40	33.9	0	17.0	33.9	107.9
Lower Shell Forging	1.1	26	3.18×10^{19}	1.304	10	33.9	0	17.0	33.9	77.8
Nozzle to Intermediate Shell Forging Circ. Weld Seam (Heat # 442011)	1.1	41	1.15×10^{19}	1.039	10	42.6	0	21.3	42.6	95.2
→ Using credible Braidwood Units 1 and 2 surveillance data	2.1	31.2	1.15×10^{19}	1.039	10	32.4	0	14.0	28.0	70.4
Intermediate to Lower shell Forging Circ Weld Seam (Heat # 442002)	1.1	54	3.07×10^{19}	1.296	-30	70.0	0	28.0	56.0	96.0
→ Using credible surveillance data	2.1	78.3	3.07×10^{19}	1.296	-30	101.5	0	14.0	28.0	99.5
Inlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-337 (Heat # 442002)	1.1	139.2	1.33×10^{17}	0.133	-10	18.5	0	9.2	18.5	26.9
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-419 (Heat # 1P5412)	1.1	168.3	1.01×10^{17}	0.110	-48.6[e]	18.6	18.0[e]	28.0[e]	66.6	36.6
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-406 (Heat # 504)	1.1	73.6	1.01×10^{17}	0.110	10	8.1	0	4.1	8.1	26.2

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 WCAP-18371-NP (Reference 4), Table E-1.
(c) Initial RT_{NDT} values are based on measured data, unless noted otherwise. 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 $0.5*\Delta RT_{NDT}$
(e) The IRT_{NDT} values are based on BAW-2308 (Reference 2). Use of BAW-2308 as an exemption to the 10 CFR 50.61 methodology was approved in Reference 10. BAW-2308 requires the use of $\sigma_I = 18°F$ and $\sigma_\Delta = 28°F$.

6.0 References

1. WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", J.D. Andrachek, et al., May 2004.

2. AREVA Document, BAW-2308, Revision 1-A and 2-A, "Initial RT_{NDT} of Linde 80 Weld Materials," August 2005 and March 2008.

3. LTR-SCS-19-13, Revision 0, "Byron Units 1 and 2 Low Temperature Overpressure Protection System (LTOPS) Analysis for 57 EFPY," December 10, 2019.

4. WCAP-18371-NP, Revision 0, "Byron Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," September 2019.

5. WCAP-16143-P, Revision 1, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2," W. Bamford, et al., October 2014.

6. 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 – Exemption from the Requirements of 10 CFR Part 50, Appendix G (TAC Nos. MC8697, MC8698, MC8699, and MC8700)," November 22, 2006. *[ADAMS Accession Number ML061890003]*

7. NRC Letter from J. S. Wiebe, NRR, to B.C. Hanson, Exelon Generation Company, LLC, "Braidwood Station, Units 1 and 2, and Byron Station, Unit Nos. 1 and 2 – Issuance of Amendments to Utilize WCAP-16143-P, Revision 1 "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2," Dated October 16, 2014 (CAC Nos. MF5033, MF5034, MF5035 and MF5036)," October 28, 2015. *[ADAMS Accession Number ML15232A441]*

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

9. WCAP-18054-NP, Revision 1, "Analysis of Capsule Y from the Exelon Generation Byron Unit 1 Reactor Vessel Radiation Surveillance Program," September 2018.

10. NRC Letter from J. S. Wiebe, NRR, to B.C. Hanson, Exelon Generation Company, LLC, "Braidwood Station, Units 1 and 2, and Byron Station, Unit Nos. 1 and 2- Issuance of Amendment Nos. 217, 217, 221, and 221 Regarding Reactor Coolant System Pressure and Temperature Limits Report Technical Specifications (EPID L-2019-LLA-0215)," September 18, 2020. [ADAMS Accession Number ML20163A046].

11. WCAP-18056-NP, Revision 1, "Analysis of Capsule Y from the Exelon Generation Byron Unit 2 Reactor Vessel Radiation Surveillance Program," September 2018.

12. WCAP-18370-NP, Revision 0, "Braidwood Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," June 2019.