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L-PI-19-038
TS 5.6.6.c

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Prairie Island Nuclear Generating Plant, Units 1 and 2
Docket Nos. 50-282 and 50-306
Renewed Facility Operating License Nos. DPR-42 and DPR-60

Prairie Island Nuclear Generating Plant Units 1 and 2 Revised Pressure and Temperature Limits Report

References: 1) NRC letter to S. Sharp, PINGP, Prairie Island Nuclear Generating Plant, Units 1 and 2 – Reactor Vessel Material Surveillance Capsule Withdrawal Schedules dated July 3, 2019

Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy (hereafter "NSPM"), hereby submits the enclosed revised Prairie Island Nuclear Generating Plant Units 1 and 2 Pressure and Temperature Limits Report (PTLR) Revision 6, in accordance with Technical Specification Section 5.6.6.c.

Revision 6 modifies the surveillance capsule withdrawal schedule for Capsule N (Prairie Island Unit 1) and Capsule N (Prairie Island Unit 2). The changes to the PTLR are found in Table 6.3.

Summary of Commitments

This letter makes no new commitments and no revisions to existing commitments.

Scott Sharp
Site Vice President, Prairie Island Nuclear Generating Plant
Northern States Power Company – Minnesota

Enclosure (1)

cc: Administrator, Region III, USNRC
Project Manager, Prairie Island, USNRC
Resident Inspector, Prairie Island, USNRC
State of Minnesota

ENCLOSURE 1

PRESSURE AND TEMPERATURE LIMITS REPORT, REVISION 6

(21 pages follow)

RECORD OF REVISION

Revision No.	Approval Date	Remarks
0	5/5/98	Original Pressure and Temperature Limits Report. Issued after May 4, 1998, approval of License Amendment Request dated March 6, 1998, as Amendment 135/127. Distribution with Technical Specification Revision 135.
1	4/6/00	Revised discussion of surveillance data credibility. Revisions to References 5.6 and 5.7 identified which incorporate findings from Comprehensive Revised Response to GL 92-01. Revised Table 6.5 data to reflect the data incorporated into the updated References 5.6 and 5.7. Changed title for the operating limit "Temperature for Disabling both Safety Injection Pumps" to the terminology "Safety Injection (SI) Pump Disable Temperature" in preparation for ITS. Changed titles of Table 6.1 and 6.2 to match Table of Contents. Changed titles of Figures 6.1 and 6.2 in the Table of Contents to match the titles on the figures. Distributed with Technical Specification Revision 153.
2	10/12/2002	This revision makes the PTLR consistent with the license amendments 158/149. Details: Revised Table of Contents to reflect changes in page numbering and the addition of 2 new subsections in section 3: "Pressurizer Temperature Limits" and "Steam Generator Temperature/Pressure Limit". Revised the wording in section 1.0 to be consistent with license amendments 158/149 5.6.6 as to the items contained in the PTLR document. Revised the list of Technical Specifications LCO/SRs that reference items in the PTLR. These changes are due to the different license amendments 158/149 numbering. Revised the "Referenced in" portions of the section 3.0 subparagraphs to reflect the different license amendments 158/149 numbering. Added subsection "Pressurizer Temperature Limits" to section 3.0. These limits were moved to the PTLR for license amendments 158/149 versus an LCO in the previous Technical Specification. Added subsection "Steam Generator Temperature/Pressure Limit" to section 3.0. This limit was moved to the PTLR for license amendments 158/149 versus an LCO in the previous Technical Specification revisions.

RECORD OF REVISION

Revision No.	Approval Date	Remarks
3	10/21/2002	This revision corrects errors in references to the TRM, a transcription error for the Maximum Pressurizer Cooldown rate and adds Site Director of Engineering as approver. Details: On cover page, added Approval of Site Director of Engineering. On page 1, changed TRM reference 3.10.1 to TRM references 3.4.4 and 3.4.5. On page 4, subsection “Pressurizer Temperature Limits,” revised the Maximum Pressurizer Cooldown Rate to 200°F per hour versus 100°F per hour, to correct a transcription error in the last revision. On page 4, subsection “Pressurizer Temperature Limits,” revised “Referenced in” specification to TRM 3.4.4 to make the PTLR consistent with the TRM. On page 4, subsection “Steam Generator Temperature/Pressure Limit,” revised the “Referenced in” specification to TRM 3.4.5 to make the PTLR consistent with the TRM.
4	3/30/2015	Updated section 4.0 discussion of ART, RT_{PTS}, fluence, and CF with values applicable to 54 EFPY. Throughout, changed the effective until from 35 EFPY to 54 EFPY. Updated tables 6.4 and 6.5 with ART values at 54 EFPY. Added references supporting change to effective until 54 EFPY.
5	8/5/2015	Remove reference to TS section 3.5.3 in section 3.0 of the PTLR which discusses SI pump disable temperature. Refer to License Amendment 213/201.
6	8/2/2019	Update Table 6.3 for newly approved material surveillance removal schedule. Refer to NRC Safety Evaluation of 7/3/19. (ML 19177A380)

Prairie Island Nuclear Generating Plant
Units One and Two

Pressure and Temperature Limits Report
Revision 6 (Effective until 54 EFPY)

PORC REVIEW DATE:	APPROVAL:
NR	PCR #: 604000000323

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1.0 PURPOSE

The purpose of the Prairie Island Nuclear Generating Station Pressure and Temperature Limits Report (PTLR) is to present operating limits for Units 1 and 2 relating to; (1) RCS pressure and temperature during Heatup, Cooldown and low temperature operation; (2) RCS heatup and cooldown rates; (3) the Over Pressure Protection System (OPPS) arming temperature; (4) OPPS lift settings; (5) Safety Injection Pump disable temperature as well as (6) thermal stress related temperature limitations for the pressurizer and steam generators. This report has been prepared in accordance with the requirements with Technical Specification 5.6.6.

2.0 APPLICABILITY

This report is applicable to both Units 1 and 2 until 54 Effective Full Power Years (EFPY) is reached on that particular units' Reactor Pressure Vessel. The Technical Specifications that are affected by the information contained in this report are:

TS 3.4.3	RCS Pressure and Temperature (P/T) Limits
TS 3.4.6	RCS Loops – MODE 4
TS 3.4.7	RCS Loops – MODE 5, Loops Filled
TS 3.4.10	Pressurizer Safety Valves
TS 3.4.12	Low Temperature Overpressure Protection (LTOP) – Reactor Coolant System Cold Leg Temperature (RCSCLT) > Safety Injection (SI) Pump Disable Temperature
TS 3.4.13	Low Temperature Overpressure Protection (LTOP) – Reactor Coolant System Cold Leg Temperature (RCSCLT) ≤ Safety Injection (SI) Pump Disable Temperature
TS 3.5.3	ECCS – Shutdown
	Miscellaneous Specifications – Technical Requirements Manual
TRM 3.4.4	PTLR Compliance – Pressurizer
TRM 3.4.5	PTLR Compliance – Steam Generator(s)

3.0 OPERATING LIMITS

All limits are valid until 54 EFPY, which is projected to be beyond the expiration of the operating license for each of Prairie Island Units 1 and 2.

Over Pressure Protection System (OPPS) Enable Temperature

310 °F*

Referenced in:	TS 3.4.6, TS 3.4.7, TS 3.4.10, TS 3.4.12, TS 3.4.13, SR 3.4.12.4, SR 3.4.13.5

*Analytical limit [225 °F] plus indicating instrument channel uncertainty [18 °F] (Reference 5.11) plus additional margin for operational simplicity.

Safety Injection (SI) Pump Disable Temperature

218 °F *

Referenced in:	TS 3.4.12, TS 3.4.13

*Analytical limit [200 °F] plus indicating instrument channel uncertainty [18 °F] (Reference 5.11).

RCS Pressure/Temperature (P/T) Limits

Figure 6.1* RCS P/T limits for heatup

Figure 6.2* RCS P/T limits for cooldown

Referenced in:	TS 3.4.3, TS 3.4.12, TS 3.4.13, SR 3.4.3.1

*Figures are analytical limits and do not include instrumentation uncertainty.

NOTE:	**Tables 6.1 and 6.2 contain a tabulated version of the curves.**

Instrumentation Uncertainty for P/T Curves

124 psig Pressure Uncertainty

18 °F Temperature Uncertainty

NOTE:	**These values must be applied to the P/T limit curves in operating procedures (Reference 5.10 and 5.11).**

RCS Heatup/Cooldown Rate Limits

100 °F per hour Maximum RCS Heatup Rate

100 °F per hour Maximum RCS Cooldown Rate

Referenced in:	TS 3.4.3, SR 3.4.3.1

Over Pressure Protection System (OPPS) PORV Setpoint

500 psig*

Referenced in:	TS 3.4.12, TS 3.4.13

*This setpoint accounts for instrument channel uncertainty (Reference 5.8).

RCS Minimum Temperature When Not Vented

86 °F*

Referenced in:	TS 3.4.3, TS 5.5.6

* Analytical limit [68°F] plus indicating instrument channel uncertainty [18°F] (Reference 5.11)

Minimum Boltup Temperature

60 °F**

Referenced in:	TS 5.5.6

**No instrument uncertainty included.

Pressurizer Temperature Limits

100 °F per hour Maximum Pressurizer Heatup Rate

200 °F per hour Maximum Pressurizer Cooldown Rate

320 °F Maximum Temperature Difference Between the Pressurizer and the Spray Fluid for which the Pressurizer Spray can be used.

Referenced in:	TRM Specification 3.4.4

Steam Generator Temperature/Pressure Limit

200 psig Maximum secondary side Pressure if the temperature of the steam generator is below **70 °F.**

Referenced in:	TRM Specification 3.4.5

4.0 DISCUSSION

This PTLR for Prairie Island Units 1 and 2 has been prepared in accordance with the requirements contained in Technical Specification 5.6.6. Periodic adjustments to the curves, limits and setpoints based on new irradiation fluences of the reactor vessel or changes in instrument uncertainty can be made under the conditions of 10CFR50.59, with the updated PTLR submitted to the NRC upon issuance.

Changes to the curves, limits, setpoints or parameters in the PTLR resulting from new or additional analysis of either beltline or weld material properties (e.g. additional capsule data) must be submitted to the NRC prior to issuance of an updated PTLR.

The results of the analysis of the Units 1 and 2 reactor vessel material surveillance capsule tests show that the limitations for Unit 1 are the most restrictive and conservative. For simplicity these results have been applied to both units.

The following parameters were used in the development of the curves, limits, and setpoints given in section 3.0 of this report. These values were obtained from Prairie Island Units 1 and 2 Reactor Vessel Radiation Surveillance Program Data. The surveillance program capsules were removed as indicated in Table 6.3.

Adjusted Reference Temperature (ART)

The adjusted reference temperature is the reference temperature (as defined in the ASME Boiler and Pressure Vessel Code, Section XI, Appendix G for Nil- ductility transition) that has been adjusted for radiation effects. This temperature was determined for all beltline materials for both Prairie Island Units 1 and 2 at the 1/4T and 3/4T thicknesses from the reactor vessel clad/base metal interface radius, where T is the reactor vessel thickness. Comparison of ARTs for all materials shows that the limiting material at 54 EFPY is the Unit 1 intermediate to lower shell forging circumferential weld material (Table 6.4 and 6.5). The limiting ARTs at 54 EFPY for this material are 150°F for 1/4T, and 133°F for 3/4T.

The Heatup and Cooldown limitations and curves remain unchanged from those developed for 35 EFPY. Because the ART values at 54 EFPY are lower than the values calculated for 35 EFPY in Reference 5.3, the ART values from Reference 5.3 remain as bounding values for development of the Pressure/Temperature limits. The limiting ARTs used to develop the Pressure/Temperature limits are as follows:

1/4T = 154 °F

3/4T = 136 °F

References:	5.3, 5.6

End of Life Fluence Reference Temperature (RT_{pts})

The RT_{pts} reference temperature is the end of life reference temperature determined at the clad/base metal interface radius of the reactor vessel and adjusted for radiation effect to the projected end of plant life. The reference temperature has been obtained for all beltline materials in both Prairie Island Units 1 and 2. The projected end of life for both units is 54 Effective Full Power Years (54 EFPY). Comparison of RT_{pts} for all materials indicates that the limiting material is the Unit 1 upper to intermediate shell forging circumferential weld material. The limiting RT_{pts} is as follows:

$$RT_{pts} = 157\ °F$$

Reference:	5.12

Neutron Fluences (f)

The ARTs are determined, in part, based on neutron fluence that is determined by using analytical techniques and passive neutron flux monitoring devices included within the Reactor Vessel Material Surveillance Program. Neutron fluence is determined for the present and future condition of the reactor vessel. Unit 1 intermediate to lower shell weld neutron fluences used in determining the 54 EFPY limiting ART for the reactor vessels are as follows:

Units are 10^{19} n/cm^2, for energies > 1.0 MeV at 54 EFPY

Clad/Base Metal Interface = 4.97

1/4T = 3.33

3/4T = 1.49

References:	5.12, 5.13

NOTE:	**These values are not the highest fluences that were obtained in the reactor vessels, but are the values determined for the most limiting material. The highest fluences were obtained at the unit 2 intermediate shell forging. (References 5.12 and 5.13).**

Chemistry Factor (CF)

Chemistry Factors are parameters used in the development of the ARTs for the beltline materials and account for the Copper and Nickel content in the reactor vessel beltline materials. The chemistry factors determined for the limiting ARTs, corresponding to the Unit 1 intermediate to lower shell circumferential weld, are as follows.

1/4T = 80.8 °F

3/4T = 80.8 °F

References:	5.13

Reactor Vessel Material Surveillance Program

The Reactor Vessel Material Surveillance Program is described in the USAR (Reference 5.9). The schedule for removal of the Units 1 and 2 capsules is contained in Table 6.3 of this report.

References:	5.2, 5.5, 5.9

Supplemental Data Tables

Tables 6.4 and 6.5 contain the development of all of the ARTs for the beltline materials for Unit 1 and Unit 2 respectfully, including all the parameters.

References:	5.13

Surveillance Data Credibility

The credibility of surveillance capsule data is determined as specified in Regulatory Guide 1.99, Revision 2, Section B. Four radiation surveillance capsules have been removed from each of the Prairie Island Reactor Vessels, as shown in Table 6.3, and the credibility of these capsule data is analyzed in references 5.2 and 5.5. The credibility of the surveillance data effects how it is applied in the development of the materials' ARTs.

When two or more credible surveillance data sets become available, the data sets may be used to determine the ART values as described in Regulatory Guide 1.99, Revision 2, Position 2.1. If the ART values based on surveillance capsule data are larger than those calculated per Regulatory Guide 1.99, Revision 2, Position 1.1, the surveillance data must be used. If the surveillance capsule data gives lower values, either may be used. In the case of the Prairie Island limiting material, the Unit 1 intermediate to lower shell forging circumferential weld, surveillance data was available but considered non-credible. This resulted in the use of the full σ_{Δ} margin of 28°F. The ART calculated using surveillance capsule data is larger than that calculated using position 1.1. For comparison Tables 6.4 and 6.5 contains the ARTs for all those materials in the surveillance programs using both Regulatory Guide 1.99, Revision 2, development methods: Position 1.1 and Position 2.1.

RCS Minimum Temperature When Not Vented

This is the RCS lower temperature limit until the system is vented with at least a 3 square inch vent.

Minimum Boltup Temperature

The Minimum Boltup Temperature is the minimum temperature of the reactor vessel flange metal required any time reactor vessel flange is under tensioning stress.

5.0 REFERENCES

5.1 WCAP-14040-NP-A, Methodology Used to Develop Cold Overpressure Mitigation, Revision 2, January 1996.

5.2 WCAP-14779, Analysis of Capsule S from the Northern States Power Company Prairie Island Unit 1 Reactor Vessel Radiation Surveillance Program, Revision 2, February 1998.

5.3 WCAP-14780, Prairie Island Unit 1 Heatup and Cooldown Limit Curves Normal Operation, Revision 3, February 1998.

5.4 WCAP-14781, Evaluation of Pressurized Thermal Shock for Prairie Island Unit 1, Revision 3, February 1998.

5.5 WCAP-14613, Analysis of Capsule P from the Northern States Power Company Prairie Island Unit 2 Reactor Vessel Radiation Surveillance Program, Revision 2, February 1998.

5.6 WCAP-14637, Prairie Island Unit 2 Heatup and Cooldown Limit Curves Normal Operation, Revision 3, December 1999.

5.7 WCAP-14638, Evaluation of Pressurized Thermal Shock for Prairie Island Unit 2, Revision 3, December 1999.

5.8 Westinghouse Letter NSP-98-0120, “Prairie Island Units 1 and 2 COMS Setpoint Analysis,” Revision 2, February 1998.

5.9 USAR Section 4.7.2, “Reactor Vessel Material Surveillance Program”

5.10 NSP Calculation No. SPCRC002, “Unit 1 Reactor Coolant Hot Leg Pressure Control Room Indication at 1PR-420 (0-750 psig scale) with 2 RC Pumps Running,” Revision 0.

5.11 NSP Calculation No. SPCRC003, “Unit 1 Wide Range RCS Cold Leg Temperature Control Room Indication Loop 1T-450B Uncertainty with Streaming Effects,” Revision 0.

5.12 Calculation CN-MRCDA-07-59, “Prairie Island Units 1 and 2 Measurement Uncertainty Recapture: Reactor Vessel Integrity Evaluation”.

5.13 Calculation ENG-ME-819, “Adjusted Reference Temperature for Unit 1 and Unit 2 Reactor Vessel Materials at 54 EFPY”.

6.0 ATTACHMENTS

6.1 Table 6.1 – 54 EFPY Heatup Data Points

6.2 Table 6.2 – 54 EFPY Cooldown Data Points

6.3 Table 6.3 – Reactor Vessel Material Surveillance Capsule Removal Schedule.

6.4 Table 6.4 – Prairie Island Unit 1 1/4T and 3/4T ART Calculations at 54 EFPY

6.5 Table 6.5 – Prairie Island Unit 2 1/4T and 3/4T ART Calculations at 54 EFPY

6.6 Figure 6.1 – Prairie Island Reactor Coolant System Heatup Limitations Applicable to 54 EFPY.

6.7 Figure 6.2 – Prairie Island Reactor Coolant System Cooldown Limitations Applicable to 54 EFPY.

Table 6.1 54 EFPY Heatup Data Points
(Without Instrumentation Uncertainty Margins)

Heatup Curves									
60 Heatup		Critical. Limit		100 Heatup		Critical. Limit		Leak Test Limit	
T	P	T	P	T	P	T	P	T	P
60	0	273	0	60	0	273	0	251	2000
60	584	273	594	60	560	273	560	273	2485
65	584	273	587	65	560	273	560		
85	584	273	584	85	560	273	560		
90	584	273	584	90	560	273	560		
95	584	273	586	95	560	273	560		
100	586	273	591	100	560	273	560		
105	591	273	597	105	560	273	560		
110	597	273	604	110	560	273	562		
115	604	273	613	115	562	273	566		
120	613	273	622	120	566	273	571		
125	622	273	633	125	571	273	577		
130	633	273	645	130	577	273	585		
135	645	273	658	135	585	273	594		
140	658	273	672	140	594	273	604		
145	672	273	687	145	604	273	615		
150	687	273	704	150	615	273	627		
155	704	273	722	155	627	273	641		
160	722	273	741	160	641	273	656		
165	741	273	761	165	656	273	672		
170	761	273	784	170	672	273	690		
175	784	273	808	175	690	273	709		
180	808	273	833	180	709	273	730		
185	833	273	861	185	730	273	752		
190	861	273	891	190	752	273	777		
195	891	273	923	195	777	273	802		
200	923	273	957	200	802	273	831		
205	957	273	994	205	831	273	861		
210	994	273	1033	210	861	273	893		
215	1033	273	1076	215	893	273	928		
220	1076	273	1121	220	928	273	966		
225	1121	273	1170	225	966	273	1006		
230	1170	275	1223	230	1006	275	1049		
235	1223	280	1279	235	1049	280	1096		
240	1279	285	1339	240	1096	285	1149		
245	1339	290	1404	245	1146	290	1199		
250	1404	295	1473	250	1199	295	1257		
255	1473	300	1548	255	1257	300	1318		
260	1548	305	1628	260	1318	305	1384		
265	1628	310	1713	265	1384	310	1455		
270	1713	315	1805	270	1455	315	1531		
275	1805	320	1903	275	1531	320	1612		
280	1903	325	2007	280	1612	325	1699		
285	2007	330	2119	285	1699	330	1792		
290	2119	335	2231	290	1792	335	1892		
295	2231	340	2347	295	1892	340	1998		
300	2347	345	2471	300	1998	345	2112		
305	2471			305	2112	350	2233		
				310	2233	355	2363		
				315	2363				

Table 6.2 54 EFPY Cooldown Data Points (Without Margins for Instrumentation Uncertainty)

Cooldown Curves									
Steady State		20 deg F		40 deg F		60 deg F		100 deg F	
T	P	T	P	T	P	T	P	T	P
60	0	60	0	60	0	60	0	60	0
60	590	60	563	60	537	60	510	60	455
65	594	65	568	65	542	65	515	65	460
70	599	70	573	70	547	70	520	70	465
75	605	75	579	75	552	75	526	75	471
80	611	80	585	80	558	80	532	80	478
85	617	85	591	85	565	85	539	85	485
90	621	90	598	90	572	90	546	90	493
95	621	95	605	95	580	95	554	95	502
100	621	100	613	100	588	100	563	100	511
105	621	105	621	105	597	105	572	105	520
110	621	110	621	110	607	110	582	110	531
115	621	115	621	115	617	115	592	115	543
116	621	116	621	120	628	120	604	120	555
116	668	116	644	125	640	125	616	125	568
120	676	120	652	130	653	130	630	130	583
125	687	125	664	135	667	135	644	135	599
130	699	130	676	140	682	140	660	140	615
135	712	135	690	145	698	145	676	145	634
140	726	140	704	150	715	150	695	150	653
145	741	145	720	155	734	155	714	155	674
150	757	150	736	160	754	160	735	160	697
155	774	155	754	165	776	165	757	165	722
160	793	160	773	170	799	170	782	170	748
165	813	165	794	175	824	175	808	175	777
170	834	170	816	180	851	180	836	180	808
175	857	175	841	185	880	185	866	185	841
180	882	180	866	190	911	190	899	190	876
185	909	185	894	195	945	195	934	195	915
190	937	190	924	200	981	200	972	200	956
195	968	195	956	205	1019	205	1012	205	1001
200	1001	200	990	210	1061	210	1056	210	1048
205	1036	205	1027	215	1106	215	1102	215	1100
210	1075	210	1067	220	1154	220	1153	220	1155
215	1115	215	1110	225	1205				
220	1159	220	1156						
225	1206	225	1205						
230	1257								
235	1311								
240	1370								
245	1432								
250	1500								
255	1572								
260	1649								
265	1732								
270	1820								
275	1915								
280	2017								
285	2126								
290	2243								
295	2367								

Table 6.3 Reactor Vessel Material Surveillance Capsule Removal Schedule

Recommended Surveillance Capsule Removal Schedule for Unit 1				
Capsule	**Capsule Location (degree)**	**Lead Factor[(a)]**	**Withdrawal EFPY[(b)]**	**Fluence (n/cm^2, E> 1.0 MeV)**
V	77	2.94	1.34	5.630×10^{18}(a)
P	247	1.72	4.60	1.318×10^{19}(a)
R	257	2.99	8.56	4.478×10^{19}(a)
S	57	1.77	18.12	4.017×10^{19}(a)
T	67	1.89	Standby	- - -
N	237	1.77	39.90 to 43.60	7.54×10^{19} to 8.11×10^{19}

Recommended Surveillance Capsule Removal Schedule for Unit 2				
Capsule	**Capsule Location (degree)**	**Lead Factor[(d)]**	**Withdrawal EFPY[(b)]**	**Fluence (n/cm^2, E> 1.0 MeV)**
V	77	2.95	1.39	6.206×10^{18}(d)
T	67	1.75	4.00	1.199×10^{19}(d)
R	257	2.99	8.81	4.376×10^{19}(d)
P	247	1.84	17.24	4.165×10^{19}(d)
N	237	1.72	40.59 to 44.29	7.32×10^{19} to 7.88×10^{19}
S	57	1.72	Standby	- - -

Notes:

(a) Updated in Capsule S dosimetry analysis.
(b) Effective Full Power Years (EFPY) from plant startup.
(c) Not used.
(d) Updated in Capsule P dosimetry analysis.

Table 6.4 Prairie Island Unit 1 1/4T and 3/4T ART Calculations at 54 EFPY

Material	CF	f @ 54 EFPY[(a)]	1/4T f 3/4T f	1/4T FF[(d)] 3/4T FF	I[(e)] (°F)	M[(g)] (°F)	ΔRT_{NDT} (°F)	ART (°F)
1/4T Calculations								
Upper Shell Forging B	51	1.770	1.185	1.047	-4	34	53.4	84
Upper to Inter. Shell Circ Weld W2	79.5	1.770	1.185	1.047	0[(c)]	66	83.3	149
Intermediate Shell Forging C	44.0	5.162	3.455	1.324	14	34	58.2	107
Using S/C Data	54.7	5.162	3.455	1.324	14	34[(b)]	72.4	121
Inter. to Lower Shell Weld W3	69.7	4.969	3.326	1.315	-13	56	91.6	135
Using S/C Data	80.8	4.969	3.326	1.315	-13	56[(b)]	106.2	150
Lower Shell Forging D	44.0	5.026	3.364	1.318	-4	34	58.0	88
3/4T Calculations								
Upper Shell Forging B	51	1.770	0.5307	0.823	-4	34	42.0	72
Upper to Inter. Shell Circ Weld W2	79.5	1.770	0.5307	0.823	0[(c)]	66	65.4	131
Intermediate Shell Forging C	44.0	5.162	1.548	1.121	14	34	49.3	98
Using S/C Data	54.7	5.162	1.548	1.121	14	34[(b)]	61.3	110
Inter. to Lower Shell Weld W3	69.7	4.969	1.490	1.110	-13	56	77.4	121
Using S/C Data	80.8	4.969	1.490	1.110	-13	56[(b)]	89.7	133
Lower Shell Forging D	44.0	5.026	1.507	1.114	-4	34	49.0	79

NOTE:

(a) Fluence values (f) are x 10^{19} n/cm^2 (E > 1.0 MeV). (Ref. 5.13)
(b) The full σ_Δ margin of 17°F for the forging and 28°F for the weld was used since the surv. data was deemed not credible (Ref. 5.3).
(c) Estimated per Standard Review Plan Section 5.3.2 (Ref. 5.3).
(d) FF, Fluence Factor = f(0.28-0.1*logf) (Ref. 5.13)
(e) I is the unirradiated material reference temperature. (Ref. 5.3)
(g) M is a margin term required for conservative results. (Ref. 5.3)

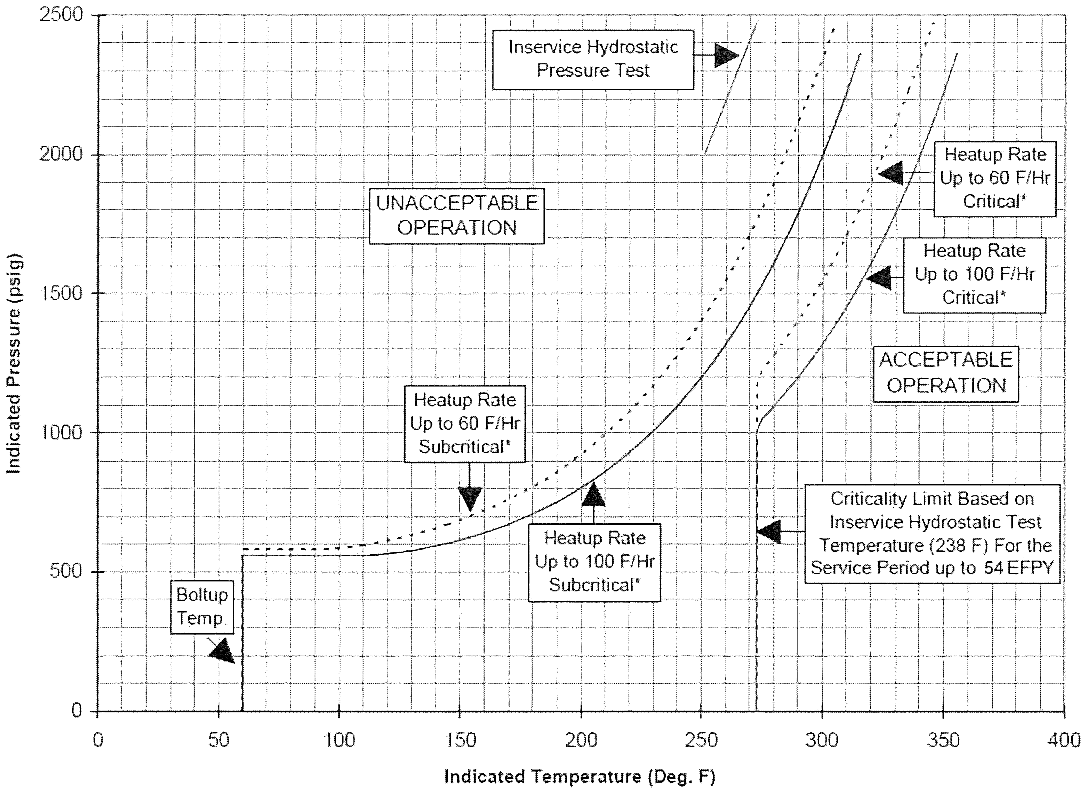
Table 6.5 Prairie Island Unit 2 1/4T and 3/4T ART Calculations at 54 EFPY

Material	CF	f @ 54 EFPY	1/4T f 3/4T f[(a)]	1/4T FF[(c)] 3/4T FF	I[(d)] (°F)	M[(e)] (°F)	ΔRT_{NDT} (°F)	ART (°F)
1/4T Calculations								
Upper Shell Forging B	44.0	1.743	1.167	1.043	-13	34	45.9	67
Upper to Inter. Shell Weld W2	70.0	1.743	1.167	1.043	-13	56	72.7	116
Using Unit 1 S/C Data[(b)]	80.8	1.743	1.167	1.043	-13	56	84.3	128
Intermediate Shell Forging C	44.0	5.196	3.478	1.325	14	34	58.3	107
Inter. to Lower Shell Weld W3	52.0	5.043	3.375	1.318	-31	56	68	93
Using S/C Data	80.0	5.043	3.375	1.318	-31	28	105.7	103
Lower Shell Forging D	51.0	5.112	3.421	1.321	-4	34	67.4	98
Using S/C Data	60.0	5.112	3.421	1.321	-4	34[(f)]	78.8	109
3/4T Calculations								
Upper Shell Forging B	44.0	1.743	0.5226	0.8187	-13	34	36.0	57
Upper to Inter. Shell Weld W2	70.0	1.743	0.5226	0.8187	-13	56	57.1	101
Using Unit 1 S/C Data[(b)]	80.8	1.743	0.5226	0.8187	-13	56	66.2	110
Intermediate Shell Forging C	44.0	5.196	1.558	1.123	14	34	49.4	98
Inter. to Lower Shell Weld W3	52.0	5.043	1.512	1.114	-31	56	57.5	83
Using S/C Data	80.0	5.043	1.512	1.114	-31	28	89.4	87
Lower Shell Forging D	51.0	5.112	1.533	1.118	-4	34	57.0	87
Using S/C Data	60.0	5.112	1.533	1.118	-4	34[(f)]	66.6	97

NOTE:

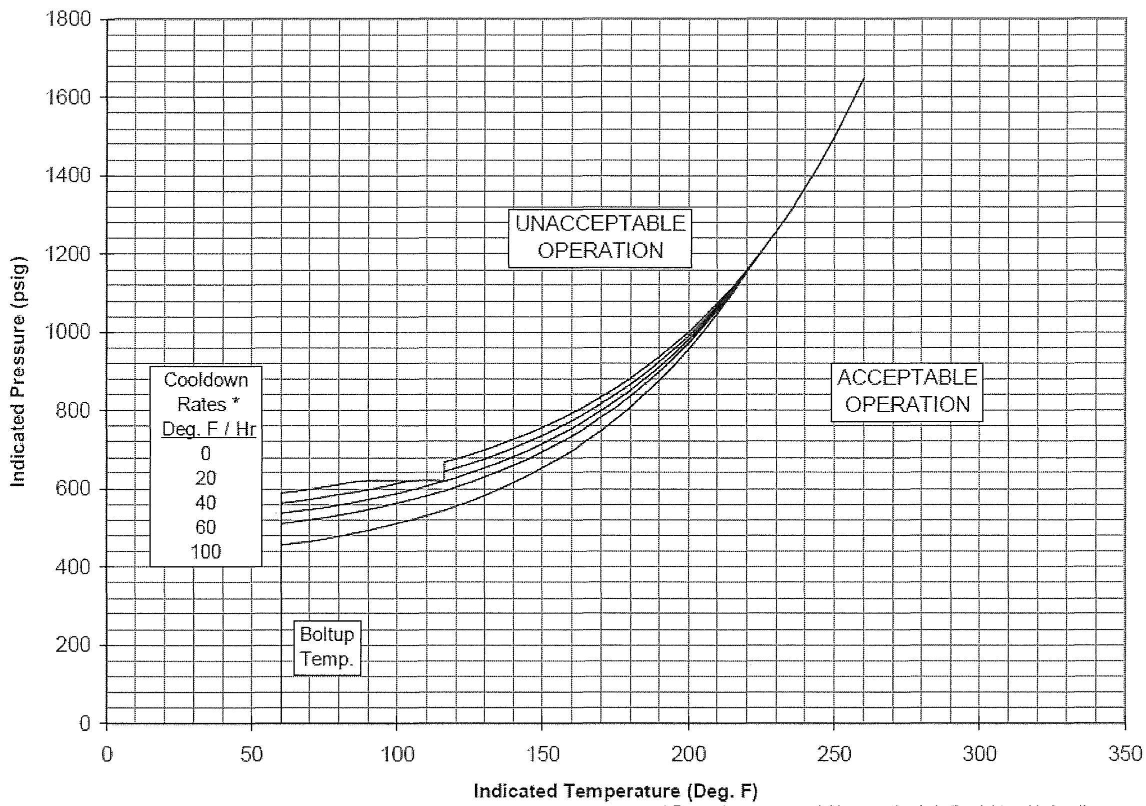
(a) Fluence values (f) are x 10^{19} n/cm^2 (E > 1.0 MeV). (Ref. 5.13)
(b) This calculation is using the chemistry factor based on the surveillance capsule data for the Prairie Island Unit 1 surveillance program. Per WCAP-14779 Rev. 1, the surveillance weld data is not credible, therefore, a full σ_Δ of 28°F was used in the margin term.
(c) FF, Fluence Factor = f(0.28-0.1*logf). (Ref. 5.13)
(d) I is the unirradiated material reference temperature. (Ref. 5.6)
(e) M is a margin term required for conservative results. (Ref. 5.6)
(f) The full σ_Δ margin of 17°F for the forging was used since the surveillance data was deemed not credible (Ref. 5.6).

Figure 6.1 Prairie Island Reactor Coolant System Heatup Limitations Applicable to 54 EFPY (w/o Margins for Instrument Uncertainty)



* For each curve, acceptable operation is to the right and below the curve.

Figure 6.2 Prairie Island Reactor Coolant System Cooldown Limitations to 54 EFPY (w/o Margins for Instrument Uncertainty)



* For each curve, acceptable operation is to the right and below the curve.