Exelon Generation.

Technical Specification 6.6.7

NMP1L 3144
April 13, 2017

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-001

Nine Mile Point Nuclear Station, Unit 1
Renewed Facility Operating License No. DPR-63
NRC Docket No. 50-220

Subject: Nine Mile Point, Unit 1, Pressure and Temperature Limits Report

Enclosed is a copy of the Pressure and Temperature Limits Report, PTLR-1, Revision 3 for Nine Mile Point Unit 1 (NMP1). This report is being submitted pursuant to NMP1 Technical Specification 6.6.7.b.

Should you have any questions regarding the information in this submittal, please contact Dennis M. Moore, Manager Site Regulatory Assurance at (315) 349-5219.

Sincerely,

Robert E. Kreider, Jr.
Plant Manager, Nine Mile Point Nuclear Station
Exelon Generation Company, LLC

REK/RSP

Enclosures: (1) Pressure and Temperature Limits Report for Nine Mile Point Unit 1 PTLR-1, Revision 3

cc: NRC Regional Administrator, Region I
NRC Project Manager
NRC Senior Resident Inspector

AD01
NRR

Enclosure 1

Pressure and Temperature Limits Report

For

Nine Mile Point Unit 1, PTLR-1, Revision 3

Exelon Generation.

NINE MILE POINT NUCLEAR STATION

NINE MILE POINT UNIT 1

Pressure and Temperature Limits Report (PTLR)

PTLR-1, Revision 03.00

Prepared by: Roy Corieri / [signature]
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Date: 3/15/17

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Date: 3/15/2017

Approved by: Mohamood Khan
Senior Manager
Design Engineering
Date: 3/15/17

This Controlled Document provides reactor pressure vessel pressure and temperature limits for use in conjunction with the Nine Mile Point Unit 1 Technical Specifications. Document pages may only be changed through the re-issue of a revision to the entire document.

NMP1 Pressure and Temperature Limits Report

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1.0 PURPOSE

The purpose of the Nine Mile Point Nuclear Station Unit 1 (NMP1) Pressure and Temperature Limits Report (PTLR) is to present operating limits relating to:

1. Reactor Coolant System (RCS) Pressure versus Temperature limits during Heatup, Cooldown and Hydrostatic/Class 1 Leak Testing.
2. RCS Heatup and Cooldown rates.
3. Reactor Pressure Vessel (RPV) head flange bolt-up temperature limits.

This report has been prepared in accordance with the requirements of Technical Specification (TS) Section 6.6.7, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," and the template provided in Licensing Topical Report SIR-05-044, Revision 0 (Reference 6.1).

2.0 APPLICABILITY

This report is applicable to the NMP1 RPV until the end of operating cycle 25. The following TS sections are affected by the information contained in this report:

- Limiting Condition for Operation Section 3.2.1, "Reactor Vessel Heatup and Cooldown Rates."
- Limiting Condition for Operation Section 3.2.2, "Minimum Reactor Vessel Temperature for Pressurization."
- Surveillance Requirement Section 4.2.2, "Minimum Reactor Vessel Temperature for Pressurization."

3.0 METHODOLOGY

The limits in this report were derived as follows:

(1) The methodology used to calculate the pressure and temperature limits is in accordance with Reference 6.1, which has been approved for BWR use by the NRC. The pressure and temperature limit calculations are documented in Reference 6.2.

(2) The neutron fluence is calculated in accordance with NRC Regulatory Guide (RG) 1.190 (Reference 6.3) based on the wetted surface fluence that is documented in Reference 6.23. The methodology used to calculate the RPV neutron fluence has been approved by the NRC in Reference 6.5.

(3) The adjusted reference temperature (ART) values for the limiting beltline materials are calculated in accordance with NRC Regulatory Guide 1.99, Revision 2 (Reference 6.6), as documented in Reference 6.7 as amended by Reference 6.24.

(4) This revision of the pressure and temperature limits is to incorporate the following changes:

- Rev. 0 - Initial issue of PTLR.
- Rev. 01.00:
  - Removed 28 EFPY curves/tables which are no longer valid
  - Limited the use of the 36 EFPY curves/tables until the end of operating cycle (EOC) 22
  - Removed 46 EFPY curves/tables which are not used during the current fluence period
  - Added a footnote 4 and clarification to fourth bullet in Section 4.0 to make the described operating limit consistent with the applicable PTLR Figure.
  - Previous Technical Specification Pressure-Temperature Limit Figures only had one curve per Figure. The PTLR Figures include several curves. A note was added to the bottom of each PTLR figure indicating plant operation shall remain to the right of all of the curves shown in each Figure.
  - Changed the legend for the temperature axis on each curve from “metal” temperature to “coolant” temperature to be consistent with the sample Figures 2-2 and 2-3 in Reference 6.1.
  - Added a footnote 1 to the last two columns of the Curve A and B Tables to make more evident that the data in the last two columns was used to plot the respective curves with instrument uncertainty and static head corrections added.
  - Removed the 28 EFPY and 46 EFPY ART Calculation Tables corresponding to the removal of the 28 EFPY and 46 EFPY PT curves/tables.

- Rev. 02.00:
  - Included newly calculated ART values from Reference 6.24 to account for increased fluence values due to updated cycle calculations and GNF2 fuel introduction.
  - Included newly developed PT Curves A, B, and C which were produced in Reference 6.25 using the new ART values.
- Rev. 03.00:
  - Replaced the Table 4 calculated 36 EFPY ART values with 46 EFPY ART values previously calculated in Reference 6.24.
  - Replaced End-of-Cycle 22 PT Curves A, B, and C with End-of-Cycle 25 PT Curves A, B and C, which were previously produced in Reference 6.25.

Changes to the curves, limits, or parameters within this PTLR, based upon new irradiation fluence data of the RPV, or other plant design assumptions in the Updated Final Safety Analysis Report (UFSAR), can be made pursuant to 10 CFR 50.59, provided the above methodologies are utilized. The revised PTLR shall be submitted to the NRC upon issuance, in accordance with TS Section 6.6.7.

4.0 OPERATING LIMITS

The pressure-temperature (P-T) limit curves included in this report represent steam dome pressure versus minimum vessel coolant temperature (as measured from recirculation loop suction) and incorporate the appropriate non-beltline limits and irradiation embrittlement effects in the beltline region.

The operating limits for pressure and temperature are required for three categories of operation: (a) hydrostatic pressure tests and leak tests, referred to as Curve A; (b) core not critical operation (heatup and cooldown), referred to as Curve B; and (c) core critical operation (heatup and cooldown), referred to as Curve C.

Complete P-T limit curves were developed for 28, 36 and 46 EFPY for NMP1, as documented in Reference 6.2. Only the NMP1 P-T limit curves for the current fluence period as documented in References 6.25 and 6.26 are included in this report. The applicable NMP1 P-T limit curves for this fluence period are included in Figures 1 through 3, and a tabulation of the curves is included in Tables 1 through 3.

Other conditions applicable to the NMP1 RPV are:

- Heatup and Cooldown rate limit during Hydrostatic and Class 1 Leak Testing (Figure 1: Curve A): **≤ 25°F/hour**[1].
- Normal Operating Heatup and Cooldown rate limit (Figure 2: Curve B – non-nuclear heating, and Figure 3: Curve C – nuclear heating): **≤ 100°F/hour**[2].
- RPV head installation temperature (i.e., bolt-up) and core not critical limit (Figure 1: Curve A – Hydrostatic and Class 1 Leak Testing; Figure 2: Curve B – non-nuclear heating): **≥ 70°F**[3].
- RPV flange and adjacent shell temperature core critical limit (Figure 3: Curve C – nuclear heating): **≥ 100°F**[4].

5.0 DISCUSSION

The adjusted reference temperature (ART) of the limiting beltline material is used to adjust beltline P-T curves to account for irradiation effects. Regulatory Guide 1.99, Revision 2 (Reference 6.6) provides the methods for determining the ART. The RG 1.99 methods for determining the limiting material and adjusting the P-T curves using ART are discussed in this section.

The vessel beltline copper (Cu) and nickel (Ni) values were obtained from the evaluation of the NMP1 vessel plate and weld materials (Reference 6.7). The Cu and Ni values were used with Tables 1 and 2 of RG 1.99 to determine a chemistry factor (CF) per Paragraph 1.1 of RG 1.99 for welds and plates, respectively.

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[1] Interpreted as: The temperature change in any 1-hour period is less than or equal to 25°F.

[2] Interpreted as: The temperature change in any 1-hour period is less than or equal to 100°F.

[3] A higher minimum bolt-up temperature of 70°F was applied to these curves, as compared to the 60°F value determined in Reference 6.2, in order to be consistent with the minimum bolt-up temperature value used in previous studies.

[4] With water level within the normal range for power operation, the minimum criticality temperature of 100°F is determined from the RT_{NDT} of the closure flange region + 60°F.

The peak RPV inside diameter (ID) fluence value of 1.74 x 10^{18} n/cm^2 at 46 EFPY used in the P-T curve evaluation was obtained from Reference 6.23 for the limiting ART. Neutron fluence values were calculated using methods that conform to the guidelines of RG 1.190 (Reference 6.3). This fluence value applies to the limiting beltline lower shell plate (Heat No. P2112 for NMP1). The fluence value for the lower shell plate is based upon an attenuation factor of 0.652 for a postulated 1/4T flaw. As a result the 1/4T fluence for 46 EFPY for the limiting lower shell plate is 1.14 x 10^{18} n/cm^2. The peak RPV ID fluence validity period for the 46 EFPY curves remains until the end of operating cycle 25 (< 46 EFPY) as documented in Reference 6.26.

The P-T limit curves for the core not critical and core critical operating conditions at a given EFPY apply for both the 1/4T and 3/4T locations. When combining pressure and thermal stresses, it is usually necessary to evaluate stresses at the 1/4T location (inside surface flaw) and the 3/4T location (outside surface flaw). This is because the thermal gradient tensile stress of interest is in the inner wall during cooldown and is in the outer wall during heatup. However, as a conservative simplification, the thermal gradient stresses at the 1/4T location are assumed to be tensile for both heatup and cooldown. This results in the approach of applying the maximum tensile stresses at the 1/4T location. This approach is conservative because irradiation effects cause the allowable toughness at the 1/4T location to be less than that at the 3/4T location for a given metal temperature. This approach causes no operational difficulties, since the boiling water reactor is at steam saturation conditions during normal operation, which is well above the P-T curve limits.

For the core not critical curve (Curve B) and the core critical curve (Curve C), the P-T curves are applicable for a coolant heatup and cooldown temperature rate of **≤ 100°F/hr**. However, the core not critical and the core critical curves were also developed to bound transients defined on the RPV thermal cycle diagram and the nozzle thermal cycle diagrams. For the hydrostatic pressure and leak test curve (Curve A), a coolant heatup and cooldown temperature rate of **≤ 25°F/hr** must be maintained. The P-T limits and corresponding limits of either Curve A or B may be applied, if necessary, while achieving or recovering from test conditions. Thus, although Curve A applies during pressure testing, the limits of Curve B may be conservatively used during pressure testing if the pressure test heatup/cooldown rate limits cannot be maintained.

The initial nil-ductility transition reference temperature (RT_{NDT}), the chemistry (weight-percent copper and nickel), and ART at the 1/4T location for all RPV beltline materials significantly affected by fluence (i.e., fluence > 10^{17} n/cm^2 for E > 1MeV) are shown in Table 4 for 46 EFPY, based on Reference 6.24. The initial RT_{NDT} values were determined and reported to the NRC in the NMP1 responses to NRC Generic Letter (GL) 92-01, Revision 1 (Reference 6.8) and GL 92-01, Revision 1, Supplement 1 (Reference 6.9). The NRC acknowledged these GL responses in letters dated March 30, 1994, August 26, 1996, and June 25, 1999 (References 6.10, 6.11, and 6.12, respectively). The initial RT_{NDT} values shown in Table 4 have previously been used in establishing the current TS P-T limit curves (license amendment approved by the NRC in Reference 6.5) and in evaluations contained in the License Renewal Application (approved by the NRC in Reference 6.13).

Per Reference 6.7 and in accordance with Appendix A of Reference 6.1, the NMP1 representative weld and plate surveillance materials data from the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) were reviewed. The representative heats of plate materials (P2112 and P2130) in the ISP are the same as the lower shell plate material in the vessel beltline region of NMP1. For plate heat P2112, since the scatter in the fitted results exceeds 1-sigma (17°F), the full 2-sigma margin term has been utilized in calculating the ART value for this plate in the vessel. For plate heat P2130, since the surveillance data was found to be credible, the margin term (σ_Δ = 17°F) is divided by two for the plate material when calculating the ART. Therefore, the CFs from the NRC's Reactor Vessel Integrity Database (Reference 6.14) and Reference 6.6 were used in the determination of ART for all NMP1 materials except for plate heat P2130.

The only computer code used in the determination of the NMP1 P-T curves was the ANSYS/Mechanical Release 6.1 (with Service Packs 2 and 3) finite element computer program (Reference 6.15) for the feedwater nozzle (non-beltline) stresses. This analysis was performed to determine through-wall thermal and pressure stress distributions for the NMP1 feedwater nozzles due to a step-change thermal transient (Reference 6.16). The ANSYS program was controlled under the vendor's 10 CFR 50 Appendix B Quality Assurance Program for nuclear quality-related work. Benchmarking consistent with NRC Generic Letter 83-11, Supplement 1 (Reference 6.17), was performed as a part of the computer program verification by comparing the solutions produced by the computer code to hand calculations for several problems. The following inputs were used in the finite element analysis:

- With respect to operating conditions, stress distributions were developed for a thermal shock of 450°F, which represents the maximum thermal shock for the feedwater nozzle during normal operating conditions. The stress results for a 450°F shock are appropriate for use in developing the non-beltline P-T curves based on the limiting feedwater nozzle, as a shock of 450°F is representative of the Turbine Roll transient that occurs in the feedwater nozzle as part of the 100°F/hr startup transient. Therefore, these stresses represent the bounding stresses in the feedwater nozzle associated with 100°F/hr heatup/cooldown limits associated with the P-T curves for the upper vessel feedwater nozzle region.
- Heat transfer coefficients were calculated from the governing design basis stress report for the NMP1 feedwater nozzle and from a model of the heat transfer coefficient as a function of flow rate, as shown in Table 5 (Reference 6.16). The heat transfer coefficients were evaluated at flow rates that bound the actual operating conditions in the feedwater nozzles at NMP1.
- A two-dimensional, axisymmetric finite element model of the feedwater nozzle was constructed (Figure 4) using the same modeling techniques that were employed to evaluate the feedwater nozzle in the governing design basis report. In order to properly model the feedwater nozzle, the analysis was performed as a penetration in a sphere and not in a cylinder. To make up for this difference in geometry, a conversion factor of 3.2 times the cylinder radius was used to model the sphere (Reference 6.16). Material properties were evaluated at 325°F (Table 6) to conservatively bound the 100°F condition where the maximum stress occurred.

6.0 REFERENCES

1. Structural Integrity Associates Report No. SIR-05-044-A, Revision 0, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," April 2007.
2. Structural Integrity Associates Calculation No. 0800297.301, Revision 1, "Revised Pressure-Temperature Curves," January 2009.
3. NRC Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001.
4. "Neutron Transport Analysis for Nine Mile Point Unit 1," Report Number MPM-405778, MPM Technologies, May 2006.

5. NRC Letter to NMPNS dated October 27, 2003, "Nine Mile Point Nuclear Station, Unit No. 1, Issuance of Amendment Re: Pressure-Temperature Limit Curves (TAC No. MB6687)."
6. NRC Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.
7. Structural Integrity Associates Calculation No. 0800297.300, Revision 1, "Evaluation of Adjusted Reference Temperature Shifts," August 2008.
8. NRC Generic Letter 92-01, Revision 1, "Reactor Vessel Structural Integrity," March 6, 1992.
9. NRC Generic Letter 92-01, Revision 1, Supplement 1, "Reactor Vessel Structural Integrity," May 19, 1995.
10. NRC Letter to Niagara Mohawk Power Corporation (NMPC) dated March 30, 1994, "Generic Letter (GL) 92-01, Revision 1, 'Reactor Vessel Structural Integrity,' Nine Mile Point Nuclear Station Unit No. 1 (NMP-1) (TAC No. M83486)."
11. NRC Letter to NMPC dated August 26, 1996, "Closeout for Niagara Mohawk Power Corporation (NMPC) Response to Generic Letter 92-01, Revision 1, Supplement 1 for the Nine Mile Point Nuclear Station, Unit Nos. 1 & 2 (TAC Nos. M92700 and M927001)."
12. NRC Letter to NMPC dated June 25, 1999, "Response to Request for Additional Information Regarding Generic Letter 92-01, Revision 1, Supplement 1, 'Reactor Vessel Structural Integrity,' Nine Mile Point Nuclear Station, Unit Nos. 1 & 2 (TAC Nos. MA1200 and MA1201)."
13. NUREG-1900, "Safety Evaluation Report Related to the License Renewal of Nine Mile Point Nuclear Station, Units 1 and 2," September 2006.
14. U. S. Nuclear Regulatory Commission, "Reactor Vessel Integrity Database Version 2.0.1," September 7, 2000.
15. ANSYS/Mechanical Release 6.1 (w/Service Packs 2 and 3), ANSYS, Inc., April 2002.
16. Structural Integrity Associates Calculation No. NMP-09Q-302, Revision 0, "Feedwater Nozzle Green's Functions for Nine Mile Point Unit 1."
17. NRC Generic Letter 83-11, Supplement 1, "Licensee Qualification for Performing Safety Analyses," June 24, 1999.
18. NRC Letter to NMPNS dated November 8, 2004, "Nine Mile Point Nuclear Station Unit Nos. 1 and 2 – Issuance of Amendments Re: Implementation of the Reactor Pressure Vessel Integrated Surveillance Program (TAC Nos. MC1758 and MC1759)."
19. G.E. Drawing No. 237E434, "Loadings Reactor Vessel."

20. "Neutron Transport Analysis for Nine Mile Point Unit 1," Report Number MPM-1209877, MPM Tecnologies, December 2009.
21. Engineering Change Notice, ECN No. N1-09-022 0800297.300-01.00 Rev. 000.
22. Calculation Change Notice, CCN No. N1-09-022 0800297.301-01.00 Rev. 000
23. "Neutron Transport Analysis for Nine Mile Point Unit 1," Report Number MPM-611914, MPM Tecnologies, December 2011.
24. Engineering Change Notice, ECN No. ECP-10-000337-CN-006 0800297.300-01.00 Rev. 000.
25. Calculation Change Notice, CCN No. ECP-10-000337-CN-007 0800297.301-01.00 Rev. 000
26. Calculation Change Notice No. ECP-16-000510-CN-001 0800297.301-01.00 Rev. 0000.

**Figure 1: NMP1 Pressure Test (Curve A)
Curve is Valid Until End of Operation Cycle 25**

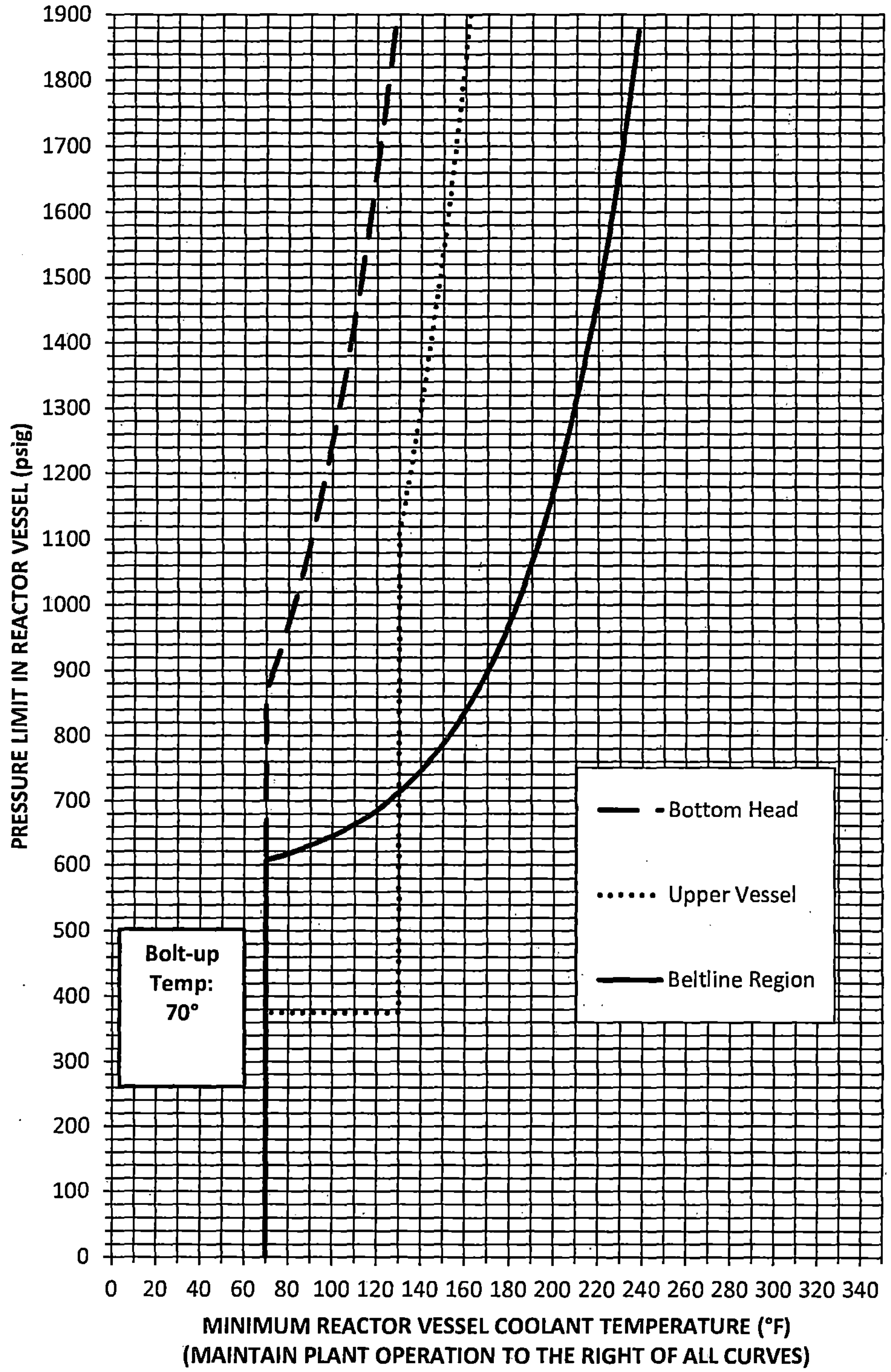


FIGURE 2: NMP1 Normal Operation (Heatup and Cooldown)

Core Not Critical (Curve B)

Curve is Valid Until End of Operating Cycle 25

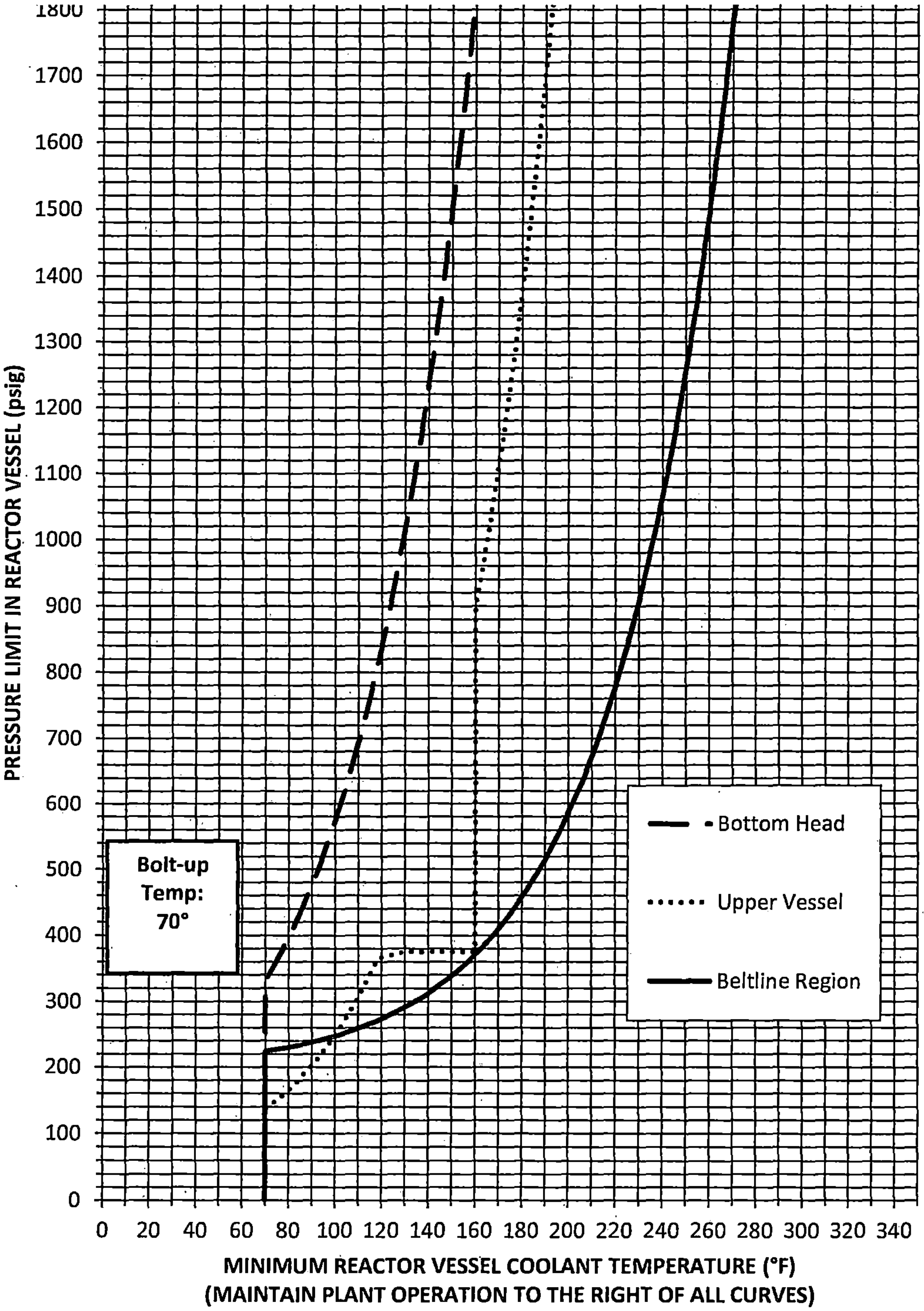
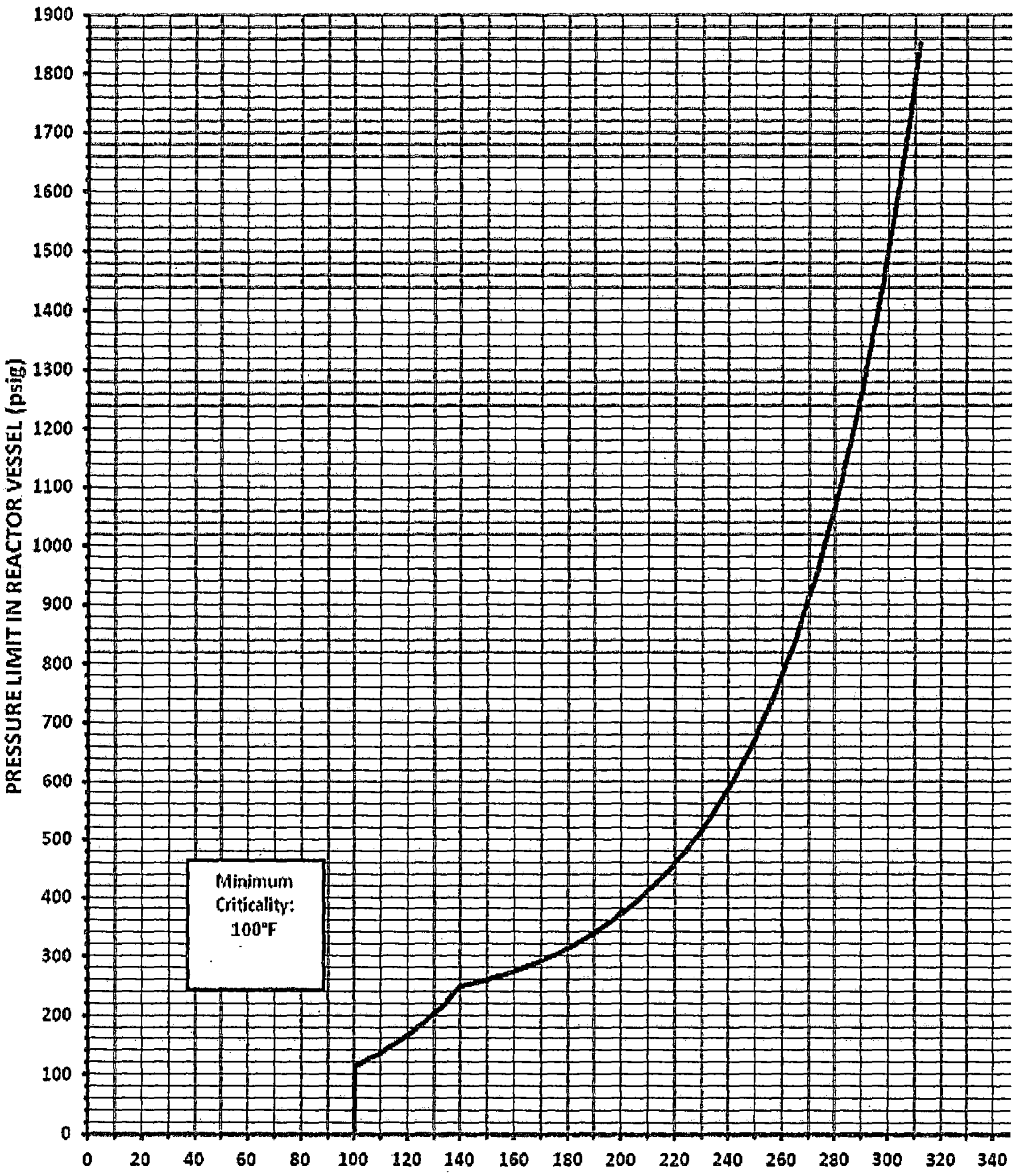


Figure 3: NMP1 Normal Operation (Heatup and Cooldown) – Core Critical (Curve C)

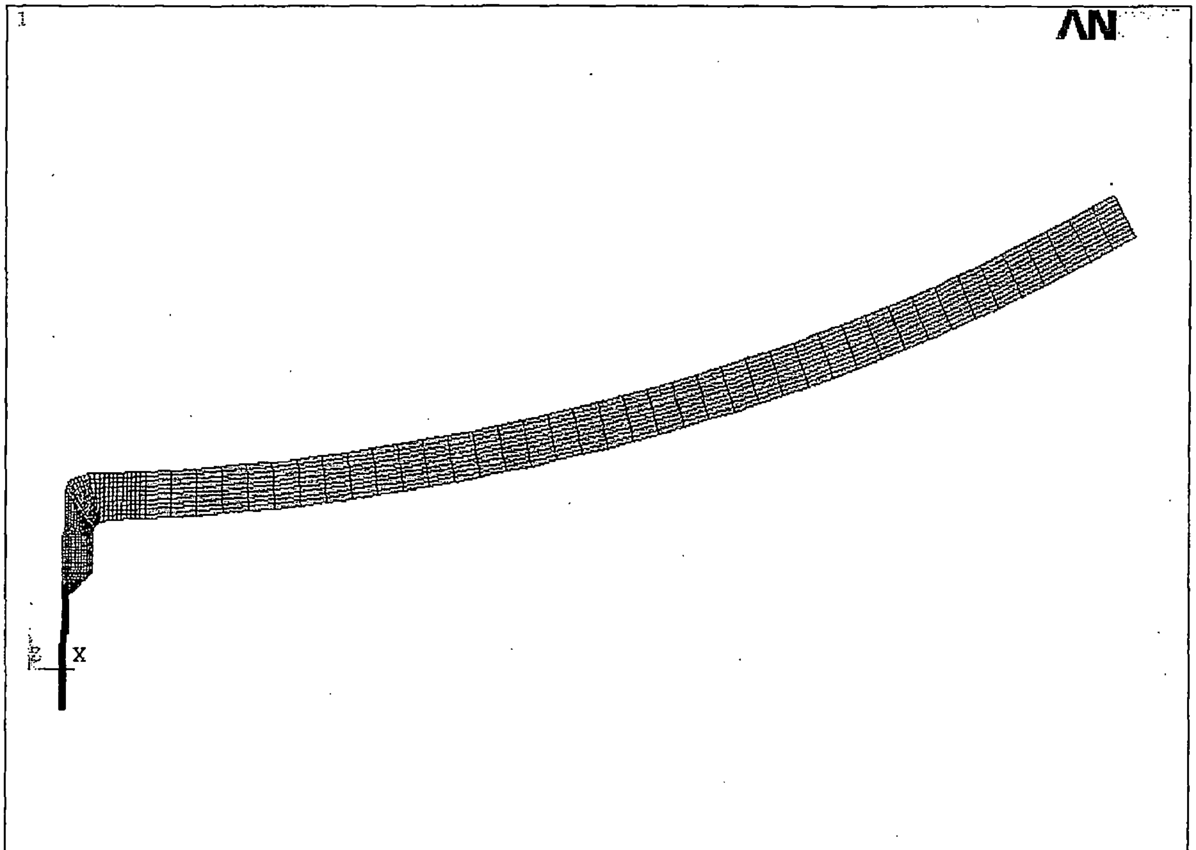
Curve is ValidUntil End of Cycle 25



MINIMUM REACTOR VESSEL COOLANT TEMPERATURE (°F)

MAINTAIN PLANT OPERATION TO RIGHT OF CURVE

Figure 4: NMP1 Feedwater Nozzle Finite Element Model



Feedwater Nozzle Finite Element Model

NMP1 Pressure and Temperature Limits Report

Table 1: NMP1 Pressure Test (Curve A) – Beltline Region, EOC 25

Plant =	NMP-1	
Component =	Beltline	(penetrations portion)
Vessel thickness, t =	7.125	inches
Vessel Radius, R =	106.5	inches
ART =	171.2	°F =====> 46 EFPY
K_{It} =	0	(no thermal effects)
Safety Factor =	1.5	
M_m =	2.472	
Temperature Adjustment =	4.0	°F (instrument uncertainty)
Pressure Adjustment =	27.7	
Pressure Adjustment =	10.0	psig (instrument uncertainty)

Gauge Fluid Temperature (°F)	K_{Ic} (ksi·√in)	K_{Im} (ksi·√in)	Gauge Pressure (psig)	(1) Temperature for P-T Curve (°F)	(1) Adjusted Pressure for P-T urve (psig)
56	35.27	23.51	0	70	0
56	35.27	23.51	636	70	599
58	35.35	23.57	638	70	600
60	35.44	23.63	640	70	602
62	35.53	23.69	641	70	603
64	35.63	23.75	643	70	605
66	35.73	23.82	645	70	607
68	35.83	23.89	647	72	609
70	35.94	23.96	649	74	611
72	36.05	24.03	651	76	613
74	36.17	24.11	653	78	615
76	36.29	24.19	655	80	617
78	36.41	24.28	657	82	619
80	36.55	24.36	659	84	622
82	36.68	24.46	662	86	624
84	36.82	24.55	664	88	627
86	36.97	24.65	667	90	629
88	37.13	24.75	670	92	632
90	37.29	24.86	673	94	635
92	37.45	24.97	676	96	638
94	37.63	25.08	679	98	641
96	37.81	25.21	682	100	645
98	38.00	25.33	686	102	648
100	38.19	25.46	689	104	651
102	38.40	25.60	693	106	655
104	38.61	25.74	697	108	659
106	38.83	25.89	701	110	663
108	39.06	26.04	705	112	667
110	39.30	26.20	709	114	671
112	39.55	26.36	714	116	676
114	39.80	26.54	718	118	681
116	40.07	26.72	723	120	685
118	40.35	26.90	728	122	690
120	40.65	27.10	733	124	696
122	40.95	27.30	739	126	701
124	41.27	27.51	745	128	707
126	41.60	27.73	751	130	713
128	41.94	27.96	757	132	719
130	42.30	28.20	763	134	725
132	42.67	28.44	770	136	732
134	43.05	28.70	777	138	739
136	43.46	28.97	784	140	746
138	43.87	29.25	792	142	754
140	44.31	29.54	800	144	762
142	44.76	29.84	808	146	770
144	45.23	30.16	816	148	779
146	45.73	30.48	825	150	787
148	46.24	30.82	834	152	797

Table 1 (Continued)

Gauge Fluid Temperature (°F)	Klc (ksi·√in)	Klm (ksi·√in)	Gauge Pressure (psig)	(1) Temperature for P-T Curve (°F)	(1) Adjusted Pressure for P-T urve (psig)
150	46.77	31.18	844	154	806
152	47.32	31.55	854	156	816
154	47.90	31.93	864	158	827
156	48.50	32.33	875	160	837
158	49.12	32.75	886	162	849
160	49.77	33.18	898	164	860
162	50.45	33.63	910	166	873
164	51.15	34.10	923	168	885
166	51.89	34.59	936	170	899
168	52.65	35.10	950	172	912
170	53.44	35.63	964	174	927
172	54.27	36.18	979	176	942
174	55.13	36.75	995	178	957
176	56.02	37.35	1011	180	973
178	56.95	37.97	1028	182	990
180	57.92	38.62	1045	184	1007
182	58.93	39.29	1063	186	1026
184	59.98	39.99	1082	188	1045
186	61.08	40.72	1102	190	1064
188	62.21	41.48	1123	192	1085
190	63.40	42.27	1144	194	1106
192	64.63	43.09	1166	196	1129
194	65.91	43.94	1189	198	1152
196	67.25	44.83	1213	200	1176
198	68.64	45.76	1239	202	1201
200	70.08	46.72	1265	204	1227
202	71.59	47.73	1292	206	1254
204	73.16	48.77	1320	208	1282
206	74.79	49.86	1349	210	1312
208	76.48	50.99	1380	212	1342
210	78.25	52.17	1412	214	1374
212	80.09	53.39	1445	216	1407
214	82.00	54.67	1480	218	1442
216	83.99	56.00	1516	220	1478
218	86.07	57.38	1553	222	1515
220	88.22	58.82	1592	224	1554
222	90.47	60.31	1632	226	1595
224	92.81	61.87	1675	228	1637
226	95.24	63.49	1719	230	1681
228	97.77	65.18	1764	232	1727
230	100.41	66.94	1812	234	1774
232	103.15	68.77	1861	236	1824
234	106.00	70.67	1913	238	1875

(1) DATA IN THESE COLUMNS WERE USED TO PLOT P-T CURVES AND INCLUDE INSTRUMENT UNCERTAINTIES/STATIC HEAD CORRECTION.

NMP1 Pressure and Temperature Limits Report

Table 2: NMP1 Normal Operation – Core Not Critical (Curve B) Beltline Region, EOC 25

Plant =	NMP-1		
Component =	Beltline	(penetrations portion)	
Vessel thickness, t =	7.125	inches	
Vessel Radius, R =	106.5	inches	
ART =	171.2	°F =====>	46 EFPY
K_{It} =	12.91		
Safety Factor =	2		
M_m =	2.472		
Temperature Adjustment =	12.2	°F (instrument uncertainty)	
Pressure Adjustment =	27.7	psig (hydrostatic pressure head for a full vessel at 70°F)	
Pressure Adjustment =	52.2	psig (instrument uncertainty)	
Heat Up and Cool Down Rate =	100	°F/Hr	

Gauge Fluid Temperature (°F)	K_{Ic} (ksi·√in)	K_{Im} (ksi·√in)	Gauge Pressure (psig)	(1) Temperature for P-T Curve (°F)	(1) Adjusted Pressure for P-T Curve (psig)
48	34.96	11.03	0	70	0
48	34.96	11.03	298	70	219
50	35.04	11.06	299	70	219
52	35.11	11.10	300	70	221
54	35.19	11.14	301	70	222
56	35.27	11.18	303	70	223
58	35.35	11.22	304	70	224
60	35.44	11.26	305	72	225
62	35.53	11.31	306	74	226
64	35.63	11.36	307	76	228
66	35.73	11.41	309	78	229
68	35.83	11.46	310	80	230
70	35.94	11.51	312	82	232
72	36.05	11.57	313	84	233
74	36.17	11.63	315	86	235
76	36.29	11.69	316	88	236
78	36.41	11.75	318	90	238
80	36.55	11.82	320	92	240
82	36.68	11.88	322	94	242
84	36.82	11.96	324	96	244
86	36.97	12.03	326	98	246
88	37.13	12.11	328	100	248
90	37.29	12.19	330	102	250
92	37.45	12.27	332	104	252
94	37.63	12.36	334	106	255
96	37.81	12.45	337	108	257
98	38.00	12.54	339	110	260
100	38.19	12.64	342	112	262
102	38.40	12.74	345	114	265
104	38.61	12.85	348	116	268
106	38.83	12.96	351	118	271
108	39.06	13.07	354	120	274
110	39.30	13.19	357	122	277
112	39.55	13.32	360	124	281
114	39.80	13.45	364	126	284
116	40.07	13.58	368	128	288
118	40.35	13.72	371	130	291
120	40.65	13.87	375	132	295
122	40.95	14.02	379	134	300
124	41.27	14.18	384	136	304
126	41.60	14.34	388	138	308
128	41.94	14.51	393	140	313
130	42.30	14.69	398	142	318
132	42.67	14.88	403	144	323
134	43.05	15.07	408	146	328
136	43.46	15.27	413	148	333
138	43.87	15.48	419	150	339

Table 2 (Continued)

Gauge Fluid Temperature (°F)	K_{Ic} (ksi·√in)	K_{Im} (ksi·√in)	Gauge Pressure (psig)	(1) Temperature for P-T Curve (°F)	(1) Adjusted Pressure for P-T Curve (psig)
140	44.31	15.70	425	152	345
142	44.76	15.92	431	154	351
144	45.23	16.16	437	156	358
146	45.73	16.41	444	158	364
148	46.24	16.66	451	160	371
150	46.77	16.93	458	162	378
152	47.32	17.20	466	164	386
154	47.90	17.49	473	166	394
156	48.50	17.79	482	168	402
158	49.12	18.10	490	170	410
160	49.77	18.43	499	172	419
162	50.45	18.77	508	174	428
164	51.15	19.12	518	176	438
166	51.89	19.49	527	178	448
168	52.65	19.87	538	180	458
170	53.44	20.26	548	182	469
172	54.27	20.68	560	184	480
174	55.13	21.11	571	186	491
176	56.02	21.55	583	188	504
178	56.95	22.02	596	190	516
180	57.92	22.51	609	192	529
182	58.93	23.01	623	194	543
184	59.98	23.53	637	196	557
186	61.08	24.08	652	198	572
188	62.21	24.65	667	200	587
190	63.40	25.24	683	202	603
192	64.63	25.86	700	204	620
194	65.91	26.50	717	206	637
196	67.25	27.17	735	208	655
198	68.64	27.86	754	210	674
200	70.08	28.59	774	212	694
202	71.59	29.34	794	214	714
204	73.16	30.12	815	216	735
206	74.79	30.94	837	218	757
208	76.48	31.78	860	220	780
210	78.25	32.67	884	222	804
212	80.09	33.59	909	224	829
214	82.00	34.54	935	226	855
216	83.99	35.54	962	228	882
218	86.07	36.58	990	230	910
220	88.22	37.66	1019	232	939
222	90.47	38.78	1050	234	970
224	92.81	39.95	1081	236	1001
226	95.24	41.16	1114	238	1034
228	97.77	42.43	1148	240	1069
230	100.41	43.75	1184	242	1104
232	103.15	45.12	1221	244	1141
234	106.00	46.55	1260	246	1180
236	108.98	48.03	1300	248	1220
238	112.07	49.58	1342	250	1262
240	115.29	51.19	1385	252	1306
242	118.64	52.86	1431	254	1351
244	122.12	54.60	1478	256	1398
246	125.75	56.42	1527	258	1447
248	129.53	58.31	1578	260	1498
250	133.46	60.27	1631	262	1551
252	137.55	62.32	1687	264	1607

(1) DATA IN THESE COLUMNS WERE USED TO PLOT P-T CURVES AND INCLUDE INSTRUMENT UNCERTAINTIES/STATIC HEAD CORRECTION.

NMP1 Pressure and Temperature Limits Report

Table 3: NMP1 Normal Operations – Core Critical (Curve C), EOC 25

Plant =	**NMP-1**	
Curve A Leak Test Temperature =	**189.1**	°F
Curve A leak Test Pressure =	**1,055**	psig
Unit Pressure	**1,875**	psig (hydrostatic pressure)
Flange RT_{NDT} =	**40**	°F

Adjusted P-T Curve Temperature (°F)	Adjusted P-T Curve Pressure (psig)
100	0
100	113
102	117
104	122
106	127
108	131
110	136
112	141
114	147
116	153
118	159
120	165
122	172
124	179
126	186
128	194
130	202
132	**210**
134	219
136	228
138	238
140	248
142	250
144	252
146	255
148	257
150	260
152	262
154	265
156	268
158	271
160	274
162	277
164	281
166	284
168	288
170	291
172	295
174	300
176	304
178	308
180	313
182	318
184	323
186	328
188	333
190	339
192	345
194	351
196	358
198	364
200	375
200	371
202	378

Table 3 (Continued)

Adjusted P-T Curve Temperature (°F)	Adjusted P-T Curve Pressure (psig)
206	394
208	402
210	410
212	419
214	428
216	438
218	448
220	458
222	469
224	480
226	491
228	504
230	516
232	529
234	543
236	557
238	572
240	587
242	603
244	620
246	637
248	655
250	674
252	694
254	714
256	735
258	757
260	780
262	804
264	829
266	855
268	882
270	910
272	939
274	970
276	1001
278	1034
280	1069
282	1104
284	1141
286	1180
288	1220
290	1262
292	1306
294	1351
296	1398
298	1447
300	1498
302	1551
304	1607
306	1664
308	1724
310	1787
312	1852

Table 4: NMP1 ART Calculations for 46 EFPY

	Description	Code No.	Heat No.	Flux Lot No.	Initial RT_{NDT} (°F)	Chemistry		Chemistry	Adjustments for 1/4T			
									ΔRT_{NDT}	Margin Terms		ART_{NDT}
						Cu (wt%)	Ni(wt%)	Factor(°F)	(°F)	σ_I(°F)	σ_Δ(°F)	(°F)
Plates	Upper Shell Plate	G-307-3	P2074	-	28	0.2	0.48	134.6	64.3	0	17	132.9
	Upper Shell Plate	G-307-4	P2076	-	40	0.27	0.53	173.85	83.0	0	17	165.5
	Upper Shell Plate	G-307-10	P2091	-	20	0.22	0.51	148.85	71.1	0	17	132.4
	Lower Shell Plate	G-8-1	P2112	-	36	0.236	0.503	228.35	91.6	0	17	171.2
	Lower Shell Plate	G-8-3/4	P2130A	-	-3	0.176	0.586	146.8	58.9	0	8.5	79.0
	Description	Code No.	Heat No.	Flux Lot No.	Initial RT_{NDT} (°F)	Chemistry		Chemistry	Adjustments for 1/4T			
									ΔRT_{NDT}	Margin Terms		ART_{NDT}
						Cu (wt%)	Ni(wt%)	Factor(°F)	(°F)	σ_I(°F)	σ_Δ(°F)	(°F)
Welds	Upper Shel Axial Welds	2-564A/C	86054B	4E5F	-50	0.214	0.046	97.59	50.9	0	25.4	51.7
	Lower Shell Axial Welds	2-564D/F	86054B	4E5F	-50	0.214	0.046	97.59	43.2	0	21.6	36.5
	Circumferential Weld Seam	3-564	1248	4M2F	-50	0.214	0.076	99.9	44.3	0	22.1	38.5

Fluence Data

	Location		Wall thickness		Fluence at ID	Attenuation	Fluence at 1/4 T,f	Fluence Factor, FF
			Full	1/4T	(n/cm^2)	1/4 = e-0.24x	(n/cm^2)	f(0.28-0.10log f)
Plates	Upper Shell Plate	G-307-3	7.125	1.781	2.55E+18	0.652	1.66E+18	0.526
	Upper Shell Plate	G-307-4	7.125	1.781	2.55E+18	0.652	1.66E+18	0.526
	Upper Shell Plate	G-307-10	7.125	1.781	2.55E+18	0.652	1.66E+18	0.526
	Lower Shell Plate	G-8-1	7.125	1.781	1.74E+18	0.652	1.14E+18	0.443
	Lower Shell Plate	G-8-3/4	7.125	1.781	1.74E+18	0.652	1.14E+18	0.443
Welds	Upper Shell Axial Welds	2-564A/C	7.125	1.781	2.49E+18	0.652	1.63E+18	0.521
	Lower Shell Axial Welds	2-564D/F	7.125	1.781	1.74E+18	0.652	1.14E+18	0.443
	Circumferential Weld Seam	3-564	7.125	1.781	1.74E+18	0.652	1.14E+18	0.443

Table 5: Heat Transfer Coefficients for NMP1 Feedwater Nozzle

0% Flow Case		
Region	Temperature (°F)	Heat Transfer Coefficient (Btu/hr-ft^2-°F)
1	550.0	205.1
2	550.0	205.1
3	550.0	205.1
4	550.0	205.1

100% Flow Case		
Region	Temperature (°F)	Heat Transfer Coefficient (Btu/hr-ft^2-°F)
1	100.0	2108.8
2	325.0	673.9
3	325.0	191.8
4	550.0	1000.0

Table 6: Feedwater Nozzle Material Properties

Material Properties

All Steels:	*Poisson's Ratio*	0.3
	Density	0.283

Reactor Vessel Plate (SA 302 Gr.B) [5, Material Group D]

T	*α*	*E*	*Thermal Conductivity, K*	*Thermal Diffusivity*	*Specific Heat, Cp*
F	*in/in*F*	*psi*	*BTU/hr*ft*F*	*ft²/hr*	*BTU/lb*F*
300	7.74E-06	2.80E+07	24.7	0.42	0.12
350	7.88E-06		24.7	0.409	0.123
400	8.01E-06	2.74E+07	24.6	0.398	0.126
325	**7.81E-06**	**2.79E+07**	**24.7**	**0.4145**	**0.1215**

Nozzle Forging (SA 336 with Code Case 1236-1) [5, Material Group A]

T	*α*	*E*	*Thermal Conductivity, K*	*Thermal Diffusivity*	*Specific Heat, Cp*
F	*in/in*F*	*psi*	*BTU/hr*ft*F*	*ft 2/hr*	*BTU/lb*F*
300	7.30E-06	2.85E+07	23.9	0.406	0.120
350	7.49E-06		23.7	0.396	0.122
400	7.66E-06	2.79E+07	23.6	0.385	0.125
325	**7.395E-06**	**2.84E+07**	**23.8**	**0.401**	**0.121**

Safe End (CS-I SA-105 Gr. II) [5, Material Group B]

T	*α*	*E*	*Thermal Conductivity, K*	*Thermal Diffusivity*	*Specific Heat, Cp*
F	*in/in*F*	*psi*	*BTU/hr*ft*F*	*ft²/hr*	*BTU/lb*F*
300	7.18E-06	2.81E+07	28.4	0.481	0.1207
350	7.47E-06		28.0	0.464	0.1234
400		2.75E+07			
325	**7.325E-06**	**2.80E+07**	**28.2**	**0.4725**	**0.1221**

APPENDIX A

NMP1 REACTOR VESSEL MATERIALS SURVEILLANCE PROGRAM

NMP1 has replaced the original materials surveillance program with the BWRVIP Integrated Surveillance Program (ISP). This program meets the requirements of 10 CFR 50, Appendix H, for integrated surveillance programs, and has been approved by the NRC (see NMP1 License Amendment No. 184, Reference 6.18). The representative plate material from the ISP is not the same heat number as the target plate in the NMP1 vessel. Also, the representative weld material is not the same heat number as the target weld in the NMP1 vessel. However, there is one matching plate heat number (heat number P2130-2) in the Supplemental Surveillance Program (SSP). Irradiated data is available from SSP capsules A, B, D, G, E, and I (Reference 6.7). Under the ISP, there is one weld heat that is scheduled to be tested in 2017. Representative surveillance capsule materials for the NMP1 weld are contained in the Hatch Unit 2 surveillance capsule program. Under the Supplemental Surveillance Program (SSP), there are no additional representative capsule materials to be tested.

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Nine Mile Point Unit 1

Activity/Document Number: ECP-16-000510 **Revision Number:** 00

Title: Revise NMP1 Design Documents Impacted by Reload 24 FCP/ECP

NOTE: For 50.59 Evaluations, information on this form will provide the basis for preparing the biennial summary report submitted to the NRC in accordance with the requirements of 10 CFR 50.59(d)(2).

Description of Activity:
(Provide a brief, concise description of what the proposed activity involves.)

The UFSAR Section X.H.1.0 Spent Fuel Storage Pool Filtering and Cooling System Design Bases states: "For a normal (full core offload or core shuffle) refueling, the offload time to the spent fuel pool and the RBCLC temperatures shall be verified to be consistent with a bulk pool temperature not to exceed 140°F with one cooling train operating." This activity performs the required UFSAR verification for N1R24.

The UFSAR Section V.C.4.0 and Appendix C require P-T limit curves to be periodically revised to account for changes in fracture toughness of the RPV components due to anticipated neutron embrittlement effects for higher accumulated fluences. The proposed activity involves the revision of the P-T limits report to include revised P-T curves beyond the end-of-cycle 22.

Reason for Activity:
(Discuss why the proposed activity is being performed.)

The UFSAR Section X.H.2.0 System Design notes the limiting design basis offload rates assuming maximum RBCLC temperatures of 95°F and a full core offload. The system design also states that "A more expedited offload may be performed if the plant conditions exist to maintain the pool water temperature at or below 140°F with one SFC train operating." This activity evaluates the specific plant conditions required to control the offload time and the RBCLC temperatures to ensure one loop of SFC train maintains the bulk pool temperature below 140°F.

ECP-16-000510 is the reload ECP for N1R24/Cycle 23. This reload core design will include an increase in enrichment for the upper and lower sections of the fuel bundles. This reload ECP also includes new TRACG LOCA analysis. These changes affect the core loading design. Hence, this reload ECP can affect the peak vessel fluence previously used to derive the 46 EFPY P-T curves. Calculation change notice (CCN) ECP-16-000510-CN-001 0800297.301-01.00 provides the technical basis to ensure previously developed 46 EFPY curves derived under previous ECPs remain conservative given the changes to neutron fluence. The 46 EFPY P-T curves are being issued in Revision 3 to PTLR-1.

Effect of Activity:
(Discuss how the activity impacts plant operations, design bases, or safety analyses described in the UFSAR.)

The activity ensures the UFSAR requirement to verify and control the offload times and that RBCLC temperatures are sufficient to maintain the SFC bulk pool temperature below the 140°F design basis maximum assuming one SFC cooling loop operating.

This activity ensures previously developed PT curves referenced in plant procedures remain conservative for operation beyond cycle 22 up to the end-of-cycle 25.

Summary of Conclusion for the Activity's 50.59 Review:
(Provide justification for the conclusion, including sufficient detail to recognize and understand the essential arguments leading to the conclusion. Provide more than a simple statement that a 50.59 Screening, 50.59 Evaluation, or a License Amendment Request, as applicable, is not required.)

A 50.59 Screening was performed resulting in all five questions answered "no". The N1R24 offload restrictions and RBCLC restrictions are consistent with the UFSAR design basis requirements and do not required a 50.59 Evaluation, change to the Technical Specifications, or Facility Operating License. The PTLR revision is also consistent with UFSAR design basis requirements.

Attachments:
Attach all 50.59 Review forms completed, as appropriate.

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Nine Mile Point Unit 1

Activity/Document Number: ECP-16-000510 **Revision Number:** 00

Title: Revise NMP1 Design Documents Impacted by Reload 24 FCP/ECP

Forms Attached: (Check all that apply.)

☐	**Applicability Review**				
☒	**50.59 Screening**	**50.59 Screening No.**	**5059-2017-120**	**Rev.**	**00**
☐	**50.59 Evaluation**	**50.59 Evaluation No.**		**Rev.**	

See LS-AA-104, Section 5, Documentation, for record retention requirements for this and all other 50.59 forms associated with the Activity.

50.59 SCREENING FORM

50.59 Screening No. 5059-2017-120 **Rev. No.** 00.00

Activity/Document Number: ECP-16-000510 **Revision Number:** 00.00

Title: Revise NMP1 Design Documents Impacted by Reload 24 FCP/ECP

I. 50.59 Screening Questions (Check correct response and provide separate written response providing the basis for the answer to each question)(See Section 5 of the Resource Manual (RM) for additional guidance):

1. Does the proposed Activity involve a change to an SSC that adversely affects an UFSAR described design function? (See Section 5.2.2.1 of the RM) ☐ YES ☒ NO

Fuel Bundle Offload Times:

The UFSAR Section X.H.1.0 Spent Fuel Storage Pool Filtering and Cooling System Design Basis states:

For a normal (full core offload or core shuffle) refueling, the offload time to the spent fuel pool and the RBCLC temperatures shall be verified to be consistent with a bulk pool temperature not to exceed 140°F with one cooling train operating.

The UFSAR Section X.H.2.0 System Design notes the limiting design basis offload rates assuming maximum RBCLC temperature of 95°F and a full core offload. The system design also states that *"A more expedited offload may be performed if the plant conditions exist to maintain the pool water temperature at or below 140°F with one SFC train operating."*

The change notice ECP-16-000510-CN-002 to the design basis engineering calculation S14-54HX018 evaluates the N1R24 specific conditions to demonstrate that the core offload time and RBCLC temperatures are verified to ensure the bulk pool temperature does not exceed the UFSAR design basis maximum of 140°F.

The conclusion is that the activity is consistent with the UFSAR provisions to allow for crediting actual refueling offload conditions to verify that the design basis is maintained during the offload and is not a change to an SSC that adversely affects the UFSAR.

Revised Pressure Temperature Limits Report (PTLR-1):

UFSAR Section V.C.4.1 through V.C.4.6 describes the design basis and the design function of the pressure temperature curves and the fact that the P-T curves are maintained in the PTLR. The design function of the P-T curve limits as described in UFSAR is to specify maximum allowable pressure as a function of reactor coolant temperature during heat-up, cooldown and core operation to ensure the reactor pressure boundary is operated with fracture toughness limits specified by ASME XI and 10CFR50 Appendix G.

UFSAR Section C.2.1.2 states that the P-T limit curves will be periodically revised to account for changes in fracture toughness of the RPV components due to anticipated neutron embrittlement effects for higher accumulated fluences. This section also states that the calculation of P-T limit curves using the projected fluence at the end of the period of extended operation would result in unnecessarily restrictive operating curves. However, projection of the adjusted reference temperature (ART), which is used in development of the curves, to the end of the period of extended operation provides assurance that development of P-T limit curves will be feasible up to the maximum predicted effective full power year (EFPY).

ECP-16-000510 is the reload ECP for N1R24/Cycle 23. This reload core design will include an increase in enrichment for the upper and lower sections of the fuel bundles. This reload ECP also includes new TRACG LOCA analysis. These changes affect the core loading design. Hence, this reload ECP can affect the peak vessel fluence previously used to derive the 46 EFPY P-T curves. Calculation change notice (CCN) ECP-16-000510-CN-001 0800297.301-01.00 provides the technical basis to ensure previously developed 46 EFPY curves derived under previous ECPs

50.59 Screening No. 5059-2017-120 **Rev. No.** 00.00

Activity/Document Number: ECP-16-000510 **Revision Number:** 00.00

Title: Revise NMP1 Design Documents Impacted by Reload 24 FCP/ECP

remain conservative given the changes to neutron fluence. The 46 EFPY P-T curves are being issued in Revision 3 to PTLR-1.

The CCN determined that the projected fluence used to develop the 46 EFPY curves in PTLR-1 Revision 3 is bounded until the end-of-cycle 25 despite the potential impacts on fluence created by the reload ECP core design. Therefore, the function of the P-T curves as described in the UFSAR is not adversely impacted by the new curves incorporated into PTLR-1 Revision 3.

2. Does the proposed Activity involve a change to a procedure that adversely affects how UFSAR described SSC design functions are performed or controlled? (See Section 5.2.2.2 of the RM) ☐ YES ☒ NO

Fuel Bundle Offload Times:

The UFSAR Sections X.H.1.0 and 2.0 provision to control the offload time and RBCLC temperature are implemented through operating procedures that provide the specific offload and RBCLC restrictions based on the cycle specific N1R24 conditions. The restrictions on the offload time and RBCLC temperature are defined in the calculation and through the ECP-15-000510-CN-002 procedure changes are implemented to NI-OP-4 and NI-OP-34.

The conclusion is that the changes to the procedures are consistent with the UFSAR design basis as noted in question 1 and therefore the change does not adversely affect how the UFSAR described SSC design functions are performed or controlled.

Revised Pressure Temperature Limits Report (PTLR-1):

The PTLR curves are directly referenced and/or pressure/temperature values from the curves are directly incorporated into plant operating, maintenance and surveillance procedures N1-OP-34, N1-OP-43A, N1-OP-43C, N1-ST-R30 and N1-MMP-GEN-901.

The above listed procedures are not adversely affected by the 46 EFPY curves added to the PTLR provided the curves are limited for use until the end-of-cycle 25 as evaluated in ECP-16-000510-CN-001 0800297.301-01.00 for the potential changes to fluence resulting from the reload ECP. N1-OP-34, N1-OP-43A and N1-OP-43C provide direction to operate, cooldown and heat-up the reactor within the limits of the PTLR curves. The new PT curves do not adversely restrict operation of the station as there is significant operating margin to the right of the curves. The latest revision of N1-ST-R30 (Revision 13) used pressure/temperature values based on a previously derived and approved PT Pressure Test Curve A for 46 EFPY in ECP-10-000337-CN-007 0800297.301. ECP-16-000510-CN-001 0800297.301-01.00 has determined that the pressure/temperature values in the existing revision of N1-ST-R30 remain valid until the end-of-operating cycle 25. N1-MMP-GEN-901 incorporates allowable bolt-up temperatures described in the USFAR. The revised PTLR does not change the allowable bolt-up temperature.

3. Does the proposed Activity involve an adverse change to an element of a UFSAR described evaluation methodology, or use of an alternative evaluation methodology, that is used in establishing the design bases or used in the safety analyses? (See Section 5.2.2.3 of the RM) ☐ YES ☒ NO

Fuel Bundle Offload Times:

The UFSAR does not provide a specific methodology for performing the verification of maximum spent fuel temperature for defining the offload times to the spent fuel pool.

The NMP1 GNF2 Cycle Independent Analyses document provides the cycle-independent decay heat safety analysis for GNF2 fuel at Nine Mile Point Unit 1. Per table 3.3, Decay Heat Inputs for NMP-1 GNF2 Equilibrium cycle, ANSI/ANS-51.1-1979 (including GE SIL No. 636) was used as the standard to calculate the decay heat values. The decay heat values for N1R24 were evaluated under ANSI/ANS-5.1-1994, which also included GE SIL No. 636. Analysis of both cases was performed under Attachment A in ECP-16-000510 and shows consistency between the GNF2 safety analysis and the cycle-specific decay heat values within 2σ and does not constitute an adverse change.

The specific evaluation of the core shuffle conditions where the spent fuel pool is considered the combined spent fuel pool and vessel cavity with the spent fuel pool gates open is included in the S14-54HX018 GOTHIC model. The conditions where the reactor shutdown cooling system is secured and the single spent fuel pool cooling loop is evaluated to maintain the combined cavity and pool below 140°F is evaluated consistent with the base calculation. To ensure a conservative interpretation of the UFSAR design requirement the pool average conditions include the average core exit bulk cavity temperature remains below the 140 degree limit. This condition defines the restrictions on RBCLC temperature and core offload time restrictions. The offload time restrictions and RBCLC restrictions are defined in the calculation and implemented conservatively into the operating procedures.

Revised Pressure Temperature Limits Report (PTLR-1):

UFSAR Section V.C.4.1 describes the PT limit curves as being developed using the methodology specified in Licensing Topical Report SIR-05-044-A and ASME Code Case N-640, as well as 10CFR50 Appendix G, and the 1989 Edition of ASME Section XI, Appendix G.

The PTLR-1 Revision 3 PT limit curves were developed under a previous ECP using the same methodology described in UFSAR Section V.C.4.1.

UFSAR Section V.C.4.3 describes the PT limit curve for the in-service system pressure tests being based on a calculated adjustment to the RT_{NDT}, based on Revision 2 of RG 1.99 to account for the effect of fast neutrons. Similarly section V.C.4.5 describes the use of RG 1.99 methodology to adjust the nil-ductility reference temperature.

The PTLR-1 Revision 3 PT limit curves for the in-service system pressure tests and core critical and not critical heat-up and cooldown used the methodology of RG 1.99 Revision 2 to adjust the RT_{NDT}.

UFSAR Section V.C.4.6 describes that reactor vessel neutron fluence has been evaluated using a method in accordance with the recommendations of RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," dated March 2001. Future evaluations of reactor vessel fluence will be completed using a method in accordance with the recommendations of RG 1.190.

CCN ECP-16-000510-CN-001 0800297.301-01.00 used a previous fluence calculation spreadsheet to extrapolate and project fluence at the end-of-cycle 25. The previous fluence calculation spreadsheet was developed using the methodology described in UFSAR Section V.C.4.6.

Based on the above review the proposed change to the PT limits report does not involve a change in PT limit curve calculation methodology described in the UFSAR.

4. Does the proposed Activity involve a test or experiment not described in the UFSAR, where an SSC is utilized or controlled in a manner that is outside the reference bounds of the design for that ☐ YES ☒ NO

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SSC or is inconsistent with analyses or descriptions in the UFSAR? (See Section 5.2.2.4 of the RM)

Fuel Bundle Offload Times:

The restrictions established on RBCLC and offload parameters are based on the engineering calculation for the N1R24 offload and are cycle specific. These restrictions control the same parameters on the offload and RBCLC in previous cycles to ensure the spent fuel pool remains below 140 degrees with one SFC loop in service.

Therefore the restrictions established and implemented in the OP changes are consistent with the UFSAR Section X.H provisions and this activity does not involve a test or experiment not described in the UFSAR where an SCC is utilized or controlled in a manner that is outside the reference bounds.

Revised Pressure Temperature Limits Report (PTLR-1):

The revised PTLR incorporates new pressure temperature curves that have accounted for the potential effects of neutron fluence changes using methods previously described in the USFAR. Therefore, use of the revised PT curves is does not involve a test or experiment.

5. Does the proposed Activity require a change to the Technical Specifications or Facility Operating License? (See Section 5.2.2.5 of the RM) ☐ YES ☒ NO

No changes are required to either the Technical Specifications or the Facility Operating License.

Tech Spec Section 1.18 states that the PTLR limit curves shall be determined for each fluence period in accordance with Specification 6.6.7.

Tech Specs 3.2.1/3.2.2 requires reactor vessel heat-up, cooldown rates, low temperature operation, criticality and in-service leakage testing be maintained within the limits of the PTLR.

Tech Spec 6.6.7 describes the analytical methods used to determine pressure and temperature limits.

The proposed changes to the PTLR as evaluated in CCN ECP-16-000510-CN-001 0800297.301-01.00 do not require a change to the Technical Specification or Facility Operating License as the changes were made consistent with the Tech Spec requirements.

II. List the documents (e.g., UFSAR, Technical Specifications, other licensing basis, technical, commitments, etc.) reviewed, including sections numbers where relevant information was found (if not identified in the response to each question).

UFSAR (U1-UFSAR)
Section V.C.4.0 – Material Radiation Exposure
Section X - A. REACTOR SHUTDOWN COOLING SYSTEM
Section X - H. SPENT FUEL STORAGE POOL FILTERING AND COOLING SYSTEM
UFSAR Appendix C – License Renewal
NMP1 Technical Specifications
3.2.7 Reactor Coolant System Isolation Valves
4.2.7 Reactor Coolant System Isolation Valves
3.2.1 Reactor Vessel Heat-up and Cooldown
3.2.2 Minimum Reactor Vessel Temperature for Pressurization
6.6.7 RCS Pressure and Temperature Limits Report (PTLR)

50.59 SCREENING FORM

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III. Select the appropriate conditions:

- ☒ **If** all questions are answered NO, **then** a 50.59 Evaluation is not required.
- ☐ **If** question 1, 2, 3, or 4 is answered YES for any portion of an Activity and question 5 is answered NO, **then** a 50.59 Evaluation shall be performed for the affected portion of the Activity.
- ☐ **If** question 5 is answered YES for any portion of an Activity and questions 1 through 4 are answered NO for the remaining portions of the Activity, **then** a License Amendment is required prior to implementation of the portion of the Activity that requires the amendment; however, a 50.59 Evaluation is **not** required for the remaining portions of the Activity.
- ☐ **If** question 5 is answered YES for any portion of an Activity and question 1, 2, 3, or 4 is answered YES for any of the remaining portions of the Activity, **then** a License Amendment is required prior to implementation of the portion of the Activity that requires the amendment **and** a 50.59 Evaluation is required for the remaining affected portions of the Activity.

IV. Screening Signoffs:

50.59 Screener: Rebecca Gazda (Offload Times)/R. Corieri (PTLR) Sign: [signature] /RLC Date: 3/14/2017
(Print name) (Signature)

50.59 Reviewer: G. Inch Sign: [signature] Date: 3/14/2017
(Print name) (Signature)

See LS-AA-104, Section 5, Documentation, for record retention requirements for this and all other 50.59 forms associated with the Activity.