May 19, 2011

NRC 2011-0055
TS 5.6.4.d
TS 5.6.5.c

U.S. Nuclear Regulatory Commission
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Point Beach Nuclear Plant Units 1 and 2
Docket Nos. 50-266 and 50-301
Renewed License Nos. DPR-24 and DPR-27

Transmittal of Core Operating Limits and Pressure and Temperature Limits Reports

Enclosed please find the following documents:

TRM 2.1 (Unit 2), Unit 2 Cycle 32 Core Operating Limits Report (COLR), Revision 14

TRM 2.2 (Units 1 and 2), Pressure and Temperature Limits Report, Revision 7

NextEra Energy Point Beach, LLC is submitting these documents in accordance with Technical Specification (TS) 5.6.4.d, "Core Operating Limits Report (COLR)," and Technical Specification 5.6.5.c, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," respectively.

This letter contains no new regulatory commitments and no revisions to existing commitments.

Very truly yours,

NextEra Energy Point Beach, LLC

James Costedio
Licensing Manager

Enclosures

cc: Administrator, Region III, USNRC
Project Manager, Point Beach Nuclear Plant, USNRC
Resident Inspector, Point Beach Nuclear Plant, USNRC
PSCW

ENCLOSURE 1

NEXTERA ENERGY POINT BEACH, LLC
POINT BEACH NUCLEAR PLANT, UNIT 2

CORE OPERATING LIMITS REPORT (COLR)
REVISION 14

18 pages follow

TRM 2.1

CORE OPERATING LIMITS REPORT (COLR)

UNIT 2 CYCLE 32

REVISION 14

1.0 CORE OPERATING LIMITS REPORT

This Core Operating Limits Report (COLR) for Point Beach Nuclear Plant has been prepared in accordance with the requirements of Technical Specification (TS) 5.6.4.

A cross-reference between the COLR sections and the PBNP Technical Specifications affected by this report is given below:

COLR Section	PBNP TS	Description
2.1	2.1.1	Reactor Core Safety Limits
2.2	3.1.1	Shutdown Margin
	3.1.4	Rod Group Alignment Limits
	3.1.5	Shutdown Bank Insertion Limits
	3.1.6	Control Bank Insertion Limits
	3.1.8	Physics Test Exceptions
2.3	3.1.3	Moderator Temperature Coefficient
2.4	3.1.5	Shutdown Bank Insertion Limit
2.5	3.1.6	Control Bank Insertion Limits
2.6	3.2.1	Nuclear Heat Flux Hot Channel Factor ($F_Q(Z)$)
2.7	3.2.2	Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)
2.8	3.2.3	Axial Flux Difference (AFD)
2.9	3.3.1	Overtemperature ΔT Setpoint
2.10	3.3.1	Overpower ΔT Setpoint
2.11	3.4.1	RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits
2.12	3.9.1	Refueling Boron Concentration
Figure 1	2.1.1	Reactor Core Safety Limits Curve
Figure 2	3.1.1	Required Shutdown Margin
Figure 3	3.1.6	Control Bank Insertion Limits
Figure 4	3.2.1	Hot Channel Factor Normalized Operating Envelope (K(Z)) for 422V+ Fuel
Figure 5	3.2.1	Summary of W(Z) as a Function of Core Height
Figure 5A	3.2.1	BOC Part-Power Summary of W(Z) as a Function of Core Height
Figure 6	3.2.3	Flux Difference Operating Envelope

2.0 OPERATING LIMITS

The cycle-specific parameter limits for the specifications listed in Section 1.0 are presented in the following subsections. These limits have been developed using the NRC approved methodologies specified in Technical Specification 5.6.4.

2.1 Reactor Core Safety Limits (TS 2.1.1)

The combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in Figure 1.

Applicability: MODES 1 and 2

2.2 Shutdown Margin (TS 3.1.1 and referenced in TS 3.1.4, 3.1.5, 3.1.6, and 3.1.8)

2.2.1 SDM shall be ≥2.0% Δk/k (see Figure 2).

Applicability: MODES 1, 2, and 3

2.2.2 SDM shall be ≥1% Δk/k.

Applicability: MODES 4 and 5

2.3 Moderator Temperature Coefficient (TS 3.1.3)

2.3.1 The upper MTC limits shall be maintained within the limits.

2.3.2 The maximum upper MTC limits shall be:

≤5 pcm/°F for power levels ≤70% RTP
≤0 pcm/°F for power levels >70% RTP

Applicability: MODE 1 and MODE 2 with keff ≥1.0.

2.4 Shutdown Bank Insertion Limit (TS 3.1.5)

NOTE: This limit is not applicable while performing SR 3.1.4.2.

2.4.1 Each shutdown bank shall be fully withdrawn.

2.4.2 Fully withdrawn is defined as ≥225 steps.

Applicability: MODES 1 and 2

2.5 Control Bank Insertion Limits (TS 3.1.6)

NOTE: This limit is not applicable while performing SR 3.1.4.2.

The control banks shall be within the insertion, sequence and overlap limits specified in Figure 3.

Applicability: MODE 1 and MODE 2 with $k_{eff} \geq 1.0$

2.6 Nuclear Heat Flux Hot Channel Factor ($F_Q(Z)$) (TS 3.2.1)

The Heat Flux Hot Channel Factor shall be within the following limits:

FQ (Z) ≤ CFQ * K(Z) / P for P > 0.5

FQ (Z) ≤ CFQ * K(Z) / 0.5 for P ≤ 0.5

Where P is the fraction of Rated Power at which the core is operating.

FQ (Z) is both:

- Steady State $F_Q^C(Z) = F_Q(Z) * 1.08$
- Transient $F_Q^W(Z) = F_Q^C(Z) * W(Z) / P$ for P > 0.5

  $F_Q^W(Z) = F_Q^C(Z) * W(Z) / 0.5$ for P ≤ 0.5

$CF_Q = 2.60$

K(Z) is the function in Figure 4

W(Z) is the function in Figures 5 and 5A

The following FQ penalty factors are applicable to Cycle 32.

Cycle Burnup (MWD/MTU)	**$F^W_Q(Z)$ Penalty Factors**
150	1.0216
297	1.0221
443	1.0224
590	1.0224
737	1.0217
884 to End-of-Cycle	1.02

Applicability: MODE 1

2.7 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) (TS 3.2.2)

The Nuclear Enthalpy Rise Hot Channel Factor shall be within the following limit:

2.7.1 $F^N_{\Delta H} < 1.68 \times [1 + 0.3(1-P)]$

where: P is the fraction of Rated Power at which the core is operating.

Applicability: MODE 1

2.8 Axial Flux Difference (AFD) (TS 3.2.3)

The AFD target band is ±5%.

The AFD Acceptability Operation Limits are provided in Figure 6.

Applicability: MODE 1 with THERMAL POWER >15% RTP

2.9 Overtemperature ΔT Setpoint (TS 3.3.1, Table 3.3.1-1 note 1)

Overtemperature ΔT setpoint parameter values (for 1800 MWt core power):

ΔT_O	=	indicated ΔT at RTP, °F
T	=	indicated RCS average temperature, °F
T'	≤	576.0°F
P'	=	2235 psig
K_1	≤	1.175 (NTSP)[1]
K_1	≤	1.188 (AV)[2]
K_2	=	0.016
K_3	=	0.000811
τ_1	=	40 sec
τ_2	=	8 sec
τ_3	=	4 sec
τ_4	=	2 sec

f(ΔI) is an event function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of RTP, such that:

[1] Nominal Setpoint Value
[2] Allowable Value

2.9.1 For qt - qb within -12, +6 percent, $f(\Delta I) = 0$.

2.9.2 For each percent that the magnitude of qt - qb exceeds +6 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 2.00 percent of Rated Power.

2.9.3 For each percent that the magnitude of qt - qb exceeds -12 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 2.69 percent of Rated Power.

Applicability: MODES 1 and 2

2.10 Overpower ΔT Setpoint (TS 3.3.1, Table 3.3.1-1 note 2)

Overpower ΔT setpoint parameter values (for 1800 MWt core power):

ΔT_O	=	indicated ΔT at RTP, °F
T	=	indicated RCS average temperature, °F
T'	≤	576.0°F
K_4	≤	1.098 (NTSP)[1]
K_4	≤	1.111 (AV)[2]
K_5	=	0.0
K_6	=	0.00123 for $T \geq T'$
K_6	=	0.0 for $T < T'$
τ_5	=	0 sec
τ_3	=	4 sec
τ_4	=	2 sec

Applicability: MODES 1 and 2

[1] Nominal Setpoint Value
[2] Allowable Value

2.11 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits (TS 3.4.1)

2.11.1 Tavg shall be ≤577°F.

2.11.2 Pressurizer pressure shall be maintained ≥2205 psig during operation.

NOTE: Pressurizer pressure limit does not apply during:

1) THERMAL POWER ramp >5% RTP per minute; or

2) THERMAL POWER step >10% RTP.

2.11.3 Reactor Coolant System raw measured Total Flow Rate shall be maintained ≥186,000 gpm.

Applicability: MODE 1

2.12 Refueling Boron Concentration (TS 3.9.1)

Boron concentrations of the Reactor Coolant System, the refueling canal, and the refueling cavity shall be maintained ≥2300 ppm.

Applicability: MODE 6

FIGURE 1
REACTOR CORE SAFETY LIMITS CURVE
(Cores containing 422V+ fuel)

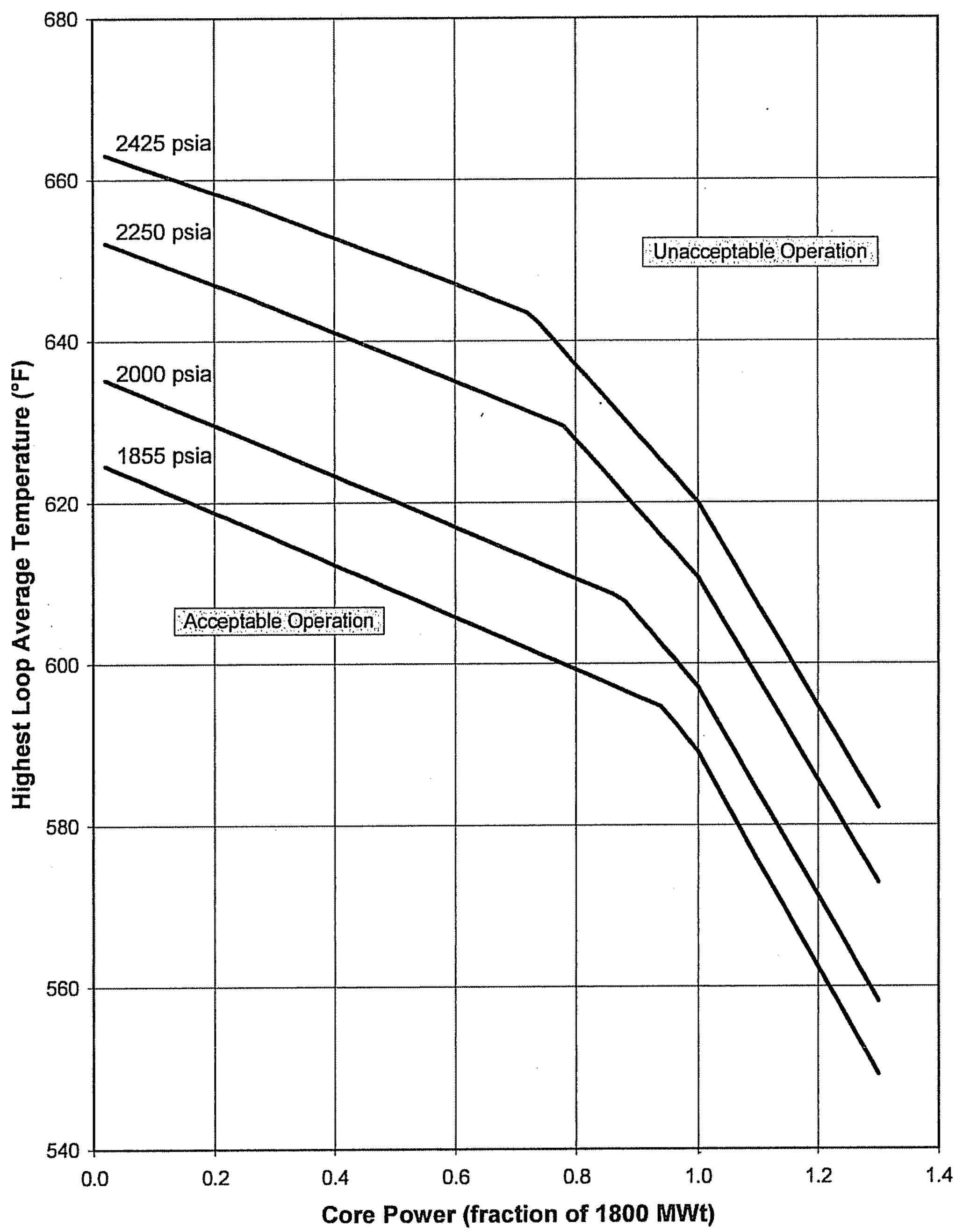


FIGURE 2
REQUIRED SHUTDOWN MARGIN

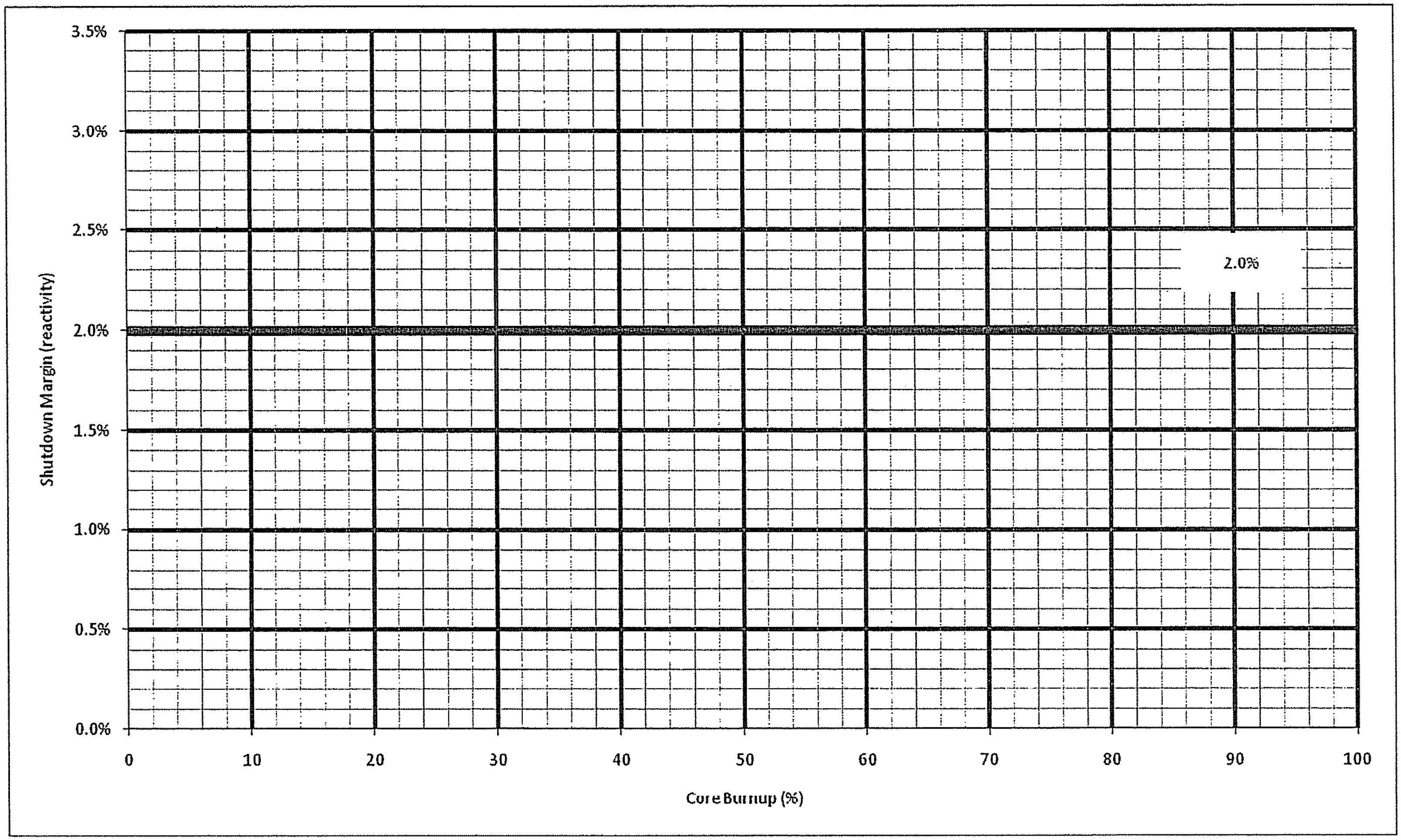
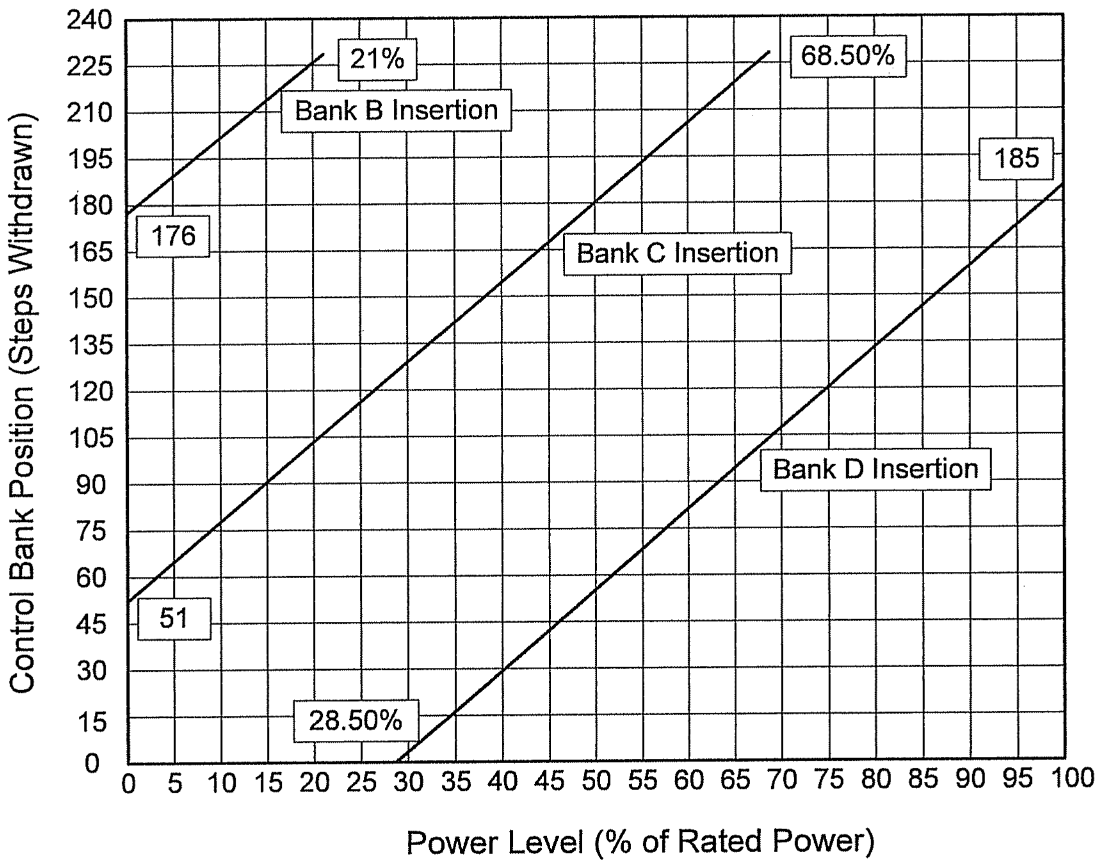


FIGURE 3
CONTROL BANK INSERTION LIMITS



NOTE: **The "fully withdrawn" parking position range ≥ 225 steps can be used without violating this Figure.**

FIGURE 4
HOT CHANNEL FACTOR NORMALIZED OPERATING ENVELOPE (K(Z))
FOR 422V+ FUEL

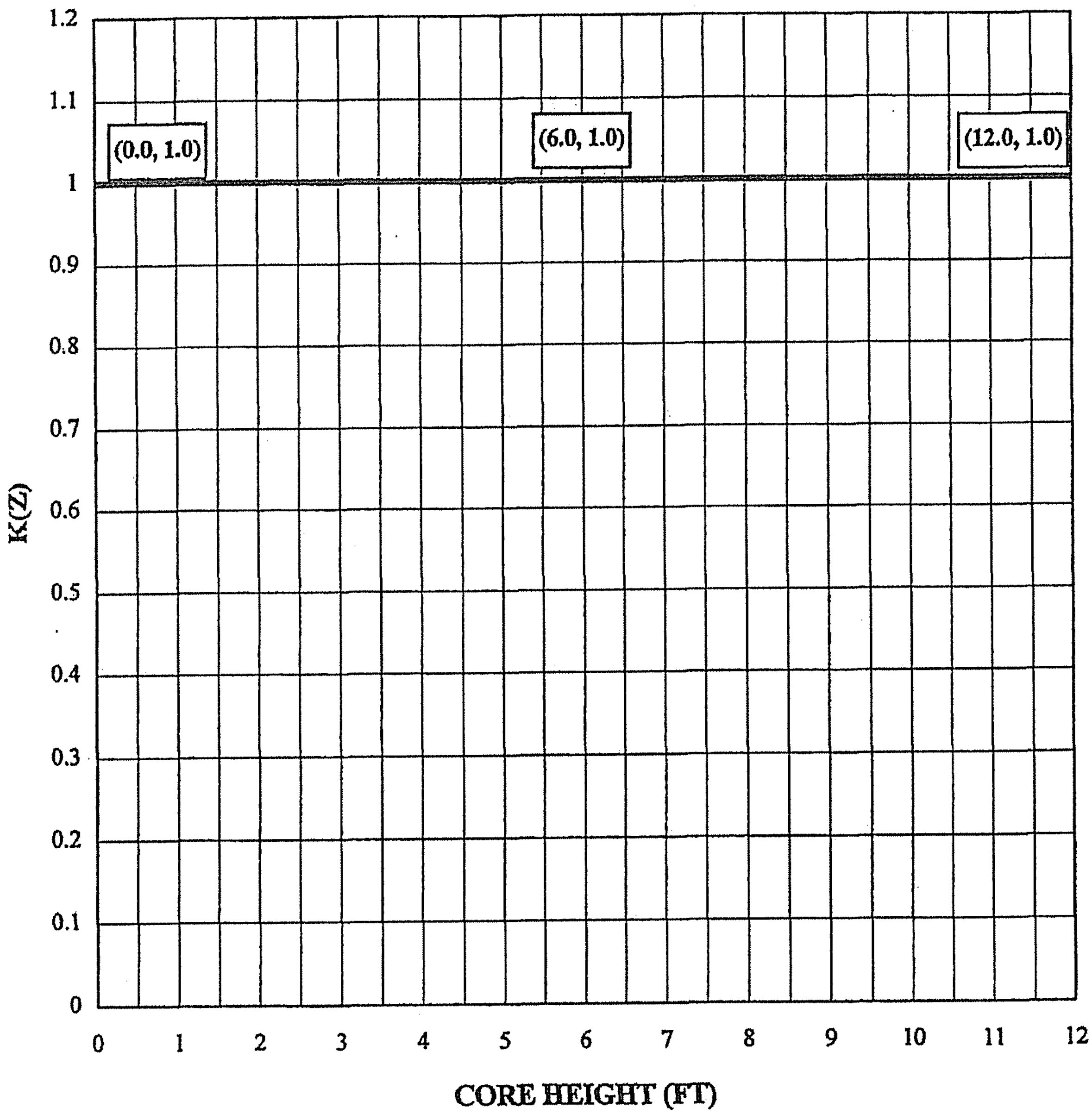


FIGURE 5
Summary of W(Z) as a Function of Core Height
(Top 15% and Bottom 12% Excluded)

Height (feet)	W(Z)				
	150 MWD/MTU	2000 MWD/MTU	8000 MWD/MTU	12000 MWD/MTU	16000 MWD/MTU
0.0	1.0000	1.0000	1.0000	1.0000	1.0000
0.2	1.0000	1.0000	1.0000	1.0000	1.0000
0.4	1.0000	1.0000	1.0000	1.0000	1.0000
0.6	1.0000	1.0000	1.0000	1.0000	1.0000
0.8	1.0000	1.0000	1.0000	1.0000	1.0000
1.0	1.0000	1.0000	1.0000	1.0000	1.0000
1.2	1.0000	1.0000	1.0000	1.0000	1.0000
1.4	1.0000	1.0000	1.0000	1.0000	1.0000
1.6	1.0980	1.0977	1.1091	1.1164	1.1225
1.8	1.0978	1.0977	1.1081	1.1145	1.1196
2.0	1.0974	1.0973	1.1066	1.1122	1.1162
2.2	1.0967	1.0966	1.1047	1.1093	1.1122
2.4	1.0957	1.0957	1.1024	1.1060	1.1076
2.6	1.0944	1.0945	1.0998	1.1022	1.1026
2.8	1.0929	1.0931	1.0968	1.0981	1.0973
3.0	1.0911	1.0913	1.0935	1.0935	1.0913
3.2	1.0892	1.0894	1.0896	1.0895	1.0870
3.4	1.0875	1.0876	1.0874	1.0869	1.0847
3.6	1.0864	1.0863	1.0855	1.0844	1.0827
3.8	1.0855	1.0855	1.0835	1.0841	1.0829
4.0	1.0845	1.0847	1.0825	1.0845	1.0839
4.2	1.0834	1.0836	1.0821	1.0852	1.0848
4.4	1.0823	1.0824	1.0816	1.0856	1.0855
4.6	1.0812	1.0809	1.0810	1.0858	1.0861
4.8	1.0800	1.0796	1.0803	1.0858	1.0864
5.0	1.0786	1.0782	1.0793	1.0856	1.0865
5.2	1.0769	1.0766	1.0783	1.0850	1.0862
5.4	1.0751	1.0747	1.0775	1.0841	1.0855
5.6	1.0730	1.0725	1.0764	1.0828	1.0845
5.8	1.0706	1.0702	1.0751	1.0812	1.0830
6.0	1.0684	1.0674	1.0734	1.0790	1.0811

FIGURE 5 (con't)
Summary of W(Z) as a Function of Core Height
(Top 15% and Bottom 12% Excluded)

Height (feet)	W(Z)				
	150 MWD/MTU	**2000 MWD/MTU**	**8000 MWD/MTU**	**12000 MWD/MTU**	**16000 MWD/MTU**
6.2	1.0663	1.0649	1.0714	1.0765	1.0786
6.4	1.0638	1.0625	1.0701	1.0735	1.0757
6.6	1.0618	1.0613	1.0689	1.0703	1.0730
6.8	1.0616	1.0614	1.0664	1.0704	1.0755
7.0	1.0611	1.0609	1.0674	1.0726	1.0795
7.2	1.0610	1.0612	1.0719	1.0765	1.0823
7.4	1.0636	1.0647	1.0759	1.0797	1.0849
7.6	1.0691	1.0702	1.0797	1.0827	1.0871
7.8	1.0746	1.0756	1.0831	1.0853	1.0890
8.0	1.0798	1.0808	1.0863	1.0879	1.0905
8.2	1.0849	1.0858	1.0893	1.0904	1.0916
8.4	1.0897	1.0905	1.0920	1.0924	1.0924
8.6	1.0943	1.0949	1.0943	1.0942	1.0929
8.8	1.0985	1.0991	1.0962	1.0955	1.0931
9.0	1.1024	1.1029	1.0977	1.0965	1.0931
9.2	1.1058	1.1062	1.0989	1.0971	1.0929
9.4	1.1089	1.1092	1.0996	1.0973	1.0927
9.6	1.1122	1.1124	1.1006	1.0972	1.0925
9.8	1.1152	1.1153	1.1014	1.0971	1.0922
10.0	1.1174	1.1175	1.1016	1.0970	1.0914
10.2	1.1196	1.1196	1.1028	1.0975	1.0910
10.4	1.0000	1.0000	1.0000	1.0000	1.0000
10.6	1.0000	1.0000	1.0000	1.0000	1.0000
10.8	1.0000	1.0000	1.0000	1.0000	1.0000
11.0	1.0000	1.0000	1.0000	1.0000	1.0000
11.2	1.0000	1.0000	1.0000	1.0000	1.0000
11.4	1.0000	1.0000	1.0000	1.0000	1.0000
11.6	1.0000	1.0000	1.0000	1.0000	1.0000
11.8	1.0000	1.0000	1.0000	1.0000	1.0000
12.0 (Top)	1.0000	1.0000	1.0000	1.0000	1.0000

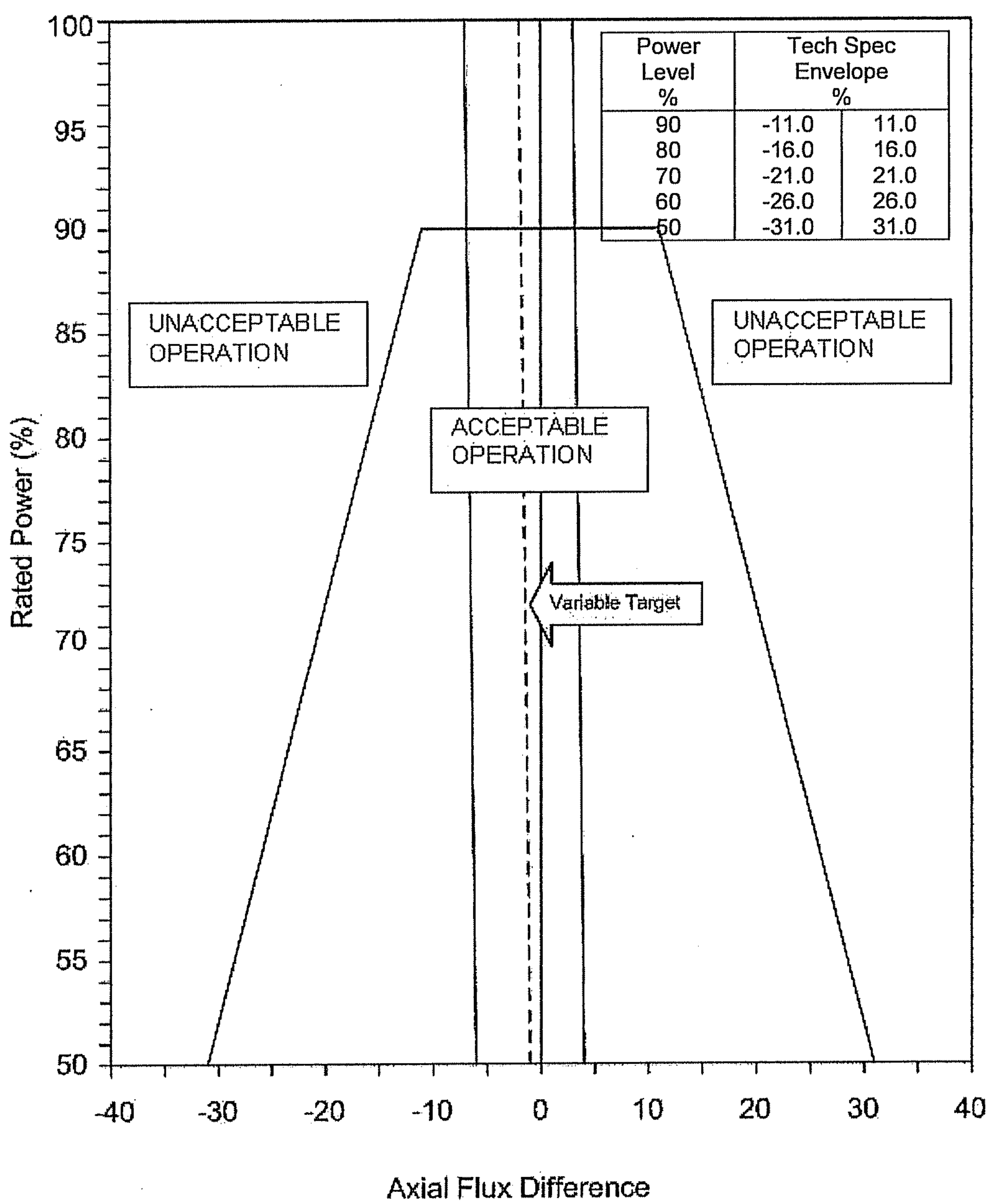
FIGURE 5A
BOC Part-Power Summary of W(Z) as a Function of Core Height
(Top 15% and Bottom 12% Excluded)

		Power W(Z) with Correction Factor (excluding power ratio)						
Height (feet)	Hot Full Power	30% Power	40% Power	50% Power	60% Power	70% Power	75% Power	80% Power
0.0 (Bottom)	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000
0.2	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000
0.4	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000
0.6	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000
0.8	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000
1.0	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000
1.2	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000
1.4	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000
1.6	1.0980	1.3417	1.2954	1.2573	1.2231	1.1897	1.1717	1.1551
1.8	1.0978	1.3239	1.2818	1.2470	1.2152	1.1838	1.1667	1.1510
2.0	1.0974	1.3071	1.2689	1.2370	1.2076	1.1780	1.1620	1.1471
2.2	1.0967	1.2911	1.2565	1.2274	1.1999	1.1724	1.1575	1.1435
2.4	1.0957	1.2751	1.2438	1.2175	1.1918	1.1664	1.1526	1.1395
2.6	1.0944	1.2587	1.2308	1.2070	1.1834	1.1599	1.1473	1.1349
2.8	1.0929	1.2413	1.2173	1.1954	1.1743	1.1527	1.1413	1.1295
3.0	1.0911	1.2243	1.2034	1.1839	1.1651	1.1455	1.1350	1.1240
3.2	1.0892	1.2079	1.1891	1.1726	1.1556	1.1384	1.1286	1.1187
3.4	1.0875	1.1912	1.1750	1.1612	1.1464	1.1309	1.1221	1.1132
3.6	1.0864	1.1758	1.1625	1.1509	1.1383	1.1241	1.1165	1.1084
3.8	1.0855	1.1640	1.1537	1.1434	1.1320	1.1196	1.1125	1.1051
4.0	1.0845	1.1520	1.1444	1.1354	1.1253	1.1148	1.1082	1.1015
4.2	1.0834	1.1394	1.1339	1.1266	1.1183	1.1093	1.1034	1.0976
4.4	1.0823	1.1261	1.1222	1.1170	1.1108	1.1032	1.0983	1.0934
4.6	1.0812	1.1126	1.1106	1.1073	1.1029	1.0969	1.0930	1.0890
4.8	1.0800	1.0986	1.0987	1.0974	1.0945	1.0903	1.0872	1.0842
5.0	1.0786	1.0843	1.0867	1.0872	1.0857	1.0833	1.0811	1.0791
5.2	1.0769	1.0700	1.0745	1.0767	1.0766	1.0760	1.0748	1.0736
5.4	1.0751	1.0567	1.0628	1.0666	1.0679	1.0685	1.0688	1.0684
5.6	1.0730	1.0423	1.0501	1.0555	1.0585	1.0606	1.0621	1.0626
5.8	1.0706	1.0266	1.0363	1.0436	1.0489	1.0529	1.0551	1.0565
6.0	1.0684	1.0103	1.0221	1.0313	1.0392	1.0453	1.0482	1.0504

FIGURE 5A (con't)
BOC Part-Power Summary of W(Z) as a Function of Core Height
(Top 15% and Bottom 12% Excluded)

Height (feet)	Hot Full Power	Power W(Z) with Correction Factor (excluding power ratio) 30% Power	40% Power	50% Power	60% Power	70% Power	75% Power	80% Power
6.2	1.0663	0.9939	1.0071	1.0182	1.0278	1.0360	1.0400	1.0434
6.4	1.0638	0.9767	0.9911	1.0039	1.0145	1.0246	1.0302	1.0348
6.6	1.0618	0.9628	0.9790	0.9928	1.0046	1.0165	1.0228	1.0284
6.8	1.0616	0.9517	0.9694	0.9842	0.9974	1.0110	1.0180	1.0245
7.0	1.0611	0.9413	0.9598	0.9759	0.9905	1.0056	1.0136	1.0208
7.2	1.0610	0.9314	0.9509	0.9682	0.9841	1.0006	1.0095	1.0175
7.4	1.0636	0.9241	0.9446	0.9630	0.9804	0.9981	1.0079	1.0168
7.6	1.0691	0.9195	0.9411	0.9605	0.9794	0.9983	1.0089	1.0190
7.8	1.0746	0.9152	0.9378	0.9582	0.9785	0.9987	1.0101	1.0212
8.0	1.0798	0.9106	0.9341	0.9558	0.9771	0.9990	1.0114	1.0235
8.2	1.0849	0.9064	0.9307	0.9537	0.9760	0.9995	1.0128	1.0259
8.4	1.0897	0.9026	0.9276	0.9516	0.9750	1.0001	1.0142	1.0282
8.6	1.0943	0.8990	0.9248	0.9497	0.9741	1.0007	1.0156	1.0304
8.8	1.0985	0.8957	0.9222	0.9479	0.9734	1.0014	1.0171	1.0326
9.0	1.1024	0.8924	0.9194	0.9460	0.9727	1.0019	1.0183	1.0349
9.2	1.1058	0.8895	0.9170	0.9445	0.9723	1.0025	1.0197	1.0373
9.4	1.1089	0.8878	0.9156	0.9438	0.9726	1.0039	1.0217	1.0402
9.6	1.1122	0.8868	0.9148	0.9434	0.9731	1.0059	1.0245	1.0440
9.8	1.1152	0.8880	0.9162	0.9450	0.9756	1.0094	1.0285	1.0488
10.0	1.1174	0.8916	0.9201	0.9494	0.9806	1.0147	1.0340	1.0547
10.2	1.1196	0.8982	0.9270	0.9568	0.9885	1.0223	1.0416	1.0622
10.4	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000
10.6	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000
10.8	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000
11.0	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000
11.2	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000
11.4	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000
11.6	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000
11.8	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000
12.0 (Top)	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000

FIGURE 6
FLUX DIFFERENCE OPERATING ENVELOPE



Power Level %	Tech Spec Envelope %	
90	-11.0	11.0
80	-16.0	16.0
70	-21.0	21.0
60	-26.0	26.0
50	-31.0	31.0

NOTE:

1. **This figure represents the Constant Axial Offset Control (CAOC) band used in safety analyses, it may be administratively tightened depending on in-core flux map results. Refer to Figure 2 of ROD 1.2 for the administrative limit.**
2. **The AFD may deviate outside the target band with THERMAL POWER < 50% RTP.**
3. **The AFD target band is +/-5% for THERMAL POWER ≥ 15% RTP.**

TABLE 1
NRC APPROVED METHODOLOGIES FOR COLR PARAMETERS

COLR Section	Parameter	NRC Approved Methodology
All	Reactor Thermal Output	Caldon, Inc., Engineering Report-80P, "TOPICAL REPORT: Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM√™ System," Revision 0, Mar 1997. Caldon, Inc., Engineering Report-160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM√™ System," Revision 0, May 2000.
2.1	Reactor Core Safety Limits	WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985
2.2	Shutdown Margin	WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985
2.3	Moderator Temperature Coefficient	WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985
2.4	Shutdown Bank Insertion Limit	WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985
2.5	Control Bank Insertion Limits	WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985
2.6	Nuclear Heat Flux Hot Channel Factor ($F_Q(Z)$)	WCAP-8403 (non-proprietary), "Power Distribution Control and Load Following Procedures," Westinghouse Electric Corporation, September 1974. WCAP-14449-P-A, "Application of Best Estimate Large Break LOCA Methodology to Westinghouse PWRs with Upper Plenum Injection," Revision 1, October 1999 (cores containing 422V + fuel) WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," January 2005. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985 WCAP-10054-P-A, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," Addendum 2, Revision 1, July 1997

TABLE 1
NRC APPROVED METHODOLOGIES FOR COLR PARAMETERS

COLR Section	Parameter	NRC Approved Methodology
2.7	Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)	WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 WCAP-16259-P-A, "Westinghouse Methodology for Applications of 3-D Transient Neutronics to Non-LOCA Accident Analysis," August 2006.
2.8	Axial Flux Difference (AFD)	WCAP-8403 (nonproprietary), "Power Distribution Control and Load Following Procedures," Westinghouse Electric Corporation, September 1974. NS-TMA-2198, Westinghouse to NRC Letter Attachment: "Operation and Safety Analysis Aspects of an Improved Load Follow Package," January 31, 1980. NS-CE-687, "Westinghouse to NRC Letter, " July 16, 1975.
2.9	Overtemperature ΔT Setpoint	WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986
2.10	Overpower ΔT Setpoint	WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986
2.11	RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits	WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989, for those events analyzed using RTDP WCAP-14787-P, Rev. 3, "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for Wisconsin Electric Power Company Point Beach Units 1 & 2 Power Uprate (1775 MWt Core Power with Feedwater Venturis, or 1800 MWt Core Power with LEFM on Feedwater Header)", February 2009. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 for those events not utilizing RTDP
2.12	Refueling Boron Concentration	WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985

ENCLOSURE 2

NEXTERA ENERGY POINT BEACH, LLC
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)
REVISION 7

14 pages follow

Note: Applicability limits for pressure temperature limits are discussed in Section 2.0, "Operating Limits."

1.0 RCS PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

This RCS Pressure and Temperature Limits Report (PTLR) for Point Beach Nuclear Plant Units 1 and 2 has been prepared in accordance with the requirements of Technical Specification 5.6.5.

The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC; specifically those described in NRC Safety Evaluations dated October 6, 2000, July 23, 2001, and October 18, 2007.

The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto (Ref 5.19). Based upon fluence values in Westinghouse report LTR-REA-08-144 (Ref 5.15), this PTLR is effective for 35.9 EFPY (approximately June 2014). (Ref 5.8)

The Technical Specifications addressed in this report are listed below:

1.1 3.4.3 Pressure/Temperature (P-T) Limits

1.2 3.4.12 Low Temperature Overpressure Protection (LTOP) System

2.0 OPERATING LIMITS

The cycle-specific parameter limits for the specifications listed in Section 1.0 are presented in the following subsections. Changes to these limits must be developed using the NRC approved methodologies specified in Technical Specification 5.6.5. These limits have been determined such that applicable limits of the safety analysis are met. Items that appear in capitalized type are defined in Technical Specification 1.1, "Definitions."

2.1 RCS Pressure and Temperature Limits (LCO 3.4.3)

2.1.1 The RCS temperature rate-of-change limits are:

a. A maximum heatup rate of 100°F in any one hour.

b. A maximum cooldown rate of 100°F in any one hour.

c. An average temperature change of ≤10°F per hour during inservice leak and hydrostatic testing operations.

2.1.2 The RCS P-T limits for heatup and cooldown are specified by Figures 1 and 2, respectively. (Ref 5.2)

2.1.3 The minimum temperature for pressurization or bolt up, using the methodology, is 60°F, which when corrected for possible instrument uncertainties is a minimum indicated RCS temperature of 78°F (as read on the RCS cold leg meter) or 70°F using the hand-held, digital pyrometer.

2.2 Low Temperature Overpressure Protection System Enable Temperature (LCO 3.4.6, 3.4.7, 3.4.10 and 3.4.12)

The enable temperature for the Low Temperature Overpressure Protection System is 285°F (includes instrument uncertainty for RCS T_c wide range). (Ref 5.4)

2.3 Low Temperature Overpressure Protection System Setpoints (LCO 3.4.12)

Pressurizer Power-Operated Relief Valve Lift Setting Limits

The limiting trip setpoint (Ref 5.26) for the pressurizer power-operated relief valves (PORVs) is ≤420 psig (includes instrument uncertainty).

The following operating restrictions ensure continued operability of the LTOP system:

2.3.1 RCP Operating Restriction - No more than one RCP in operation for RCS temperature <180°F. (Ref 5.20 to 5.24)

2.3.2 Charging Pumps - Limit the number of operating charging pumps to two when LTOP is in service. (Ref 5.20 to 5.24)

2.4 Criticality and Hydrostatic Leak Test Limits

2.4.1 Criticality and hydrostatic leak test limits are shown on the RCS Pressure Temperature Limits for heatup, Figure 1. (Ref 5.2)

3.0 REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM

The reactor vessel material irradiation surveillance specimens shall be removed and examined to determine changes in material properties. The removal schedules for Units 1 and 2 are provided in Tables 1 and 2, respectively.

For the period of the renewed facility operating license, all capsules in the reactor vessel that are removed and tested shall meet the test procedures and reporting requirements of ASTM E 185-82. Any changes to the capsule withdrawal schedule, including spare capsules, shall be approved by the NRC prior to implementation. (Ref 5.16 and 5.17)

The pressure vessel surveillance program is in compliance with Appendix H to 10 CFR 50, entitled, "Reactor Vessel Radiation Surveillance Program." The material test requirements and the acceptance standard utilize the nil-ductility temperature, RT_{NDT}, which is determined in accordance with ASTM E208. 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.

Surveillance specimens for the limiting materials for the PBNP reactor vessels are not included in the plant specific surveillance program. Therefore, the results of the examinations of these specimens do not meet the credibility criteria of Regulatory Guide 1.99, Revision 2, for PBNP Units 1 and 2.

During the period of extended operation, reactor vessel surveillance capsules will be removed and tested in accordance with the schedule contained in the most recently NRC-approved Pressurized Water Reactor Owners Group (PWROG) Master Integrated Reactor Vessel Surveillance Program (MIRVSP) Document. (Ref. 5.5)(Ref 5.25)

4.0 SUPPLEMENTAL DATA INFORMATION

The limiting RT_{PTS} values for the PBNP limiting beltline materials at 35.9 EFPY are:

- Unit 1 - Intermediate to Lower Shell Circ Weld = 281.0°F; Lower Shell Axial Weld = 250.3°F (Ref. 5.8, Attachment A)
- Unit 2 - Intermediate to Lower Shell Circ Weld = 295.1°F; Intermediate Shell Forging = 150.2°F (Ref. 5.8, Attachment A)

5.0 REFERENCES

5.1 WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," Revision 2, January 1996

5.2 WCAP-15976, "Point Beach Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," Revision 1, March 2008

5.3 WEPCO Calculation Addendum No. 98-0156-00-A, Revision 0, "Evaluation of New Surveillance Data on Chemistry Factor for Weld Wire Heat 61782, Point Beach Unit 1," 9/22/1999

5.4 Westinghouse Letter WEP-08-25, "Transmittal of LTOPS Setpoint Evaluation," dated March 14, 2008

5.5 PWR Owner Group Topical Report BAW-1543(NP), Revision 4, Supplement 6-A, "Supplement to the Master Integrated Reactor Vessel Surveillance Program" (TAC No. MC9608), June 2007

5.6 BAW-2325, "Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity," May 1998

5.7 CEOG Report "Best Estimate Copper and Nickel Values in CE Fabricated Reactor Vessel Welds," CE NPSD-1039, Revision 2, Final Report, June 1997

5.8 Westinghouse Letter LTR-PCAM-08-57, "Point Beach Units 1 and 2 EPU P-T Limit Curve Applicability Determination and Related Calculations," dated December 2008

5.9 ASME B&PVC Code Case N-641, "Alternative Pressure-Temperature Relationship and Low Temperature Overpressure Protection System Requirements, Section XI, Division 1"

5.10 NRC Letter, "Point Beach Nuclear Plant, Units 1 and 2 – Exemption from the Requirements of 10CFR50.60 (TAC NOS. MA9680 and MA9681)," dated October 6, 2000

5. 11 NRC Letter, "Point Beach Nuclear Plant, Units 1 and 2 – Acceptance of Methodology for Referencing Pressure Temperature Limits Report (TAC Nos. MA8459 and MA8460)," dated July 23, 2001

5. 12 NRC Letter, "Point Beach Nuclear Plant, Units 1 and 2 – Issuance of Amendments RE: The Conversion to Improved Technical Specifications (TAC Nos. MA7186 and MA7187)," dated August 8, 2001

5.13 Deleted

5.14 NRC SE "Amendment Nos. 229/234 to Facility Operating Licenses DPR-24 and DPR-27, (approving use of FERRET Code as approved methodology for determining RCS pressure and temperature limits)," dated October 18, 2007

5.15 Westinghouse Letter LTR-REA-08-144, "Summary of Neutron Fluence Evaluations for the Point Beach Units 1 and 2 Extended Power Uprate," dated January 2009

5.16 Renewed Facility Operating License DPR-24, Point Beach Nuclear Plant Unit 1

5.17 Renewed Facility Operating License DPR-27, Point Beach Nuclear Plant Unit 2

5.18 Deleted

5.19 Root Cause Evaluation 01092944, "Apparent Non-compliance with TS 5.6.5.c," Corrective Action to Prevent Recurrence (CATPR) 2 Root Cause (RC)2.

5.20 CL 4C, Low Temperature Overpressurization Protection Unit 1

5.21 CL 4C, Low Temperature Overpressurization Protection Unit 2

5.22 OP 3C, Hot Standby to Cold Shutdown

5.23 OP 4B, Reactor Coolant Pump Operation

5.24 OP 1A, Cold Shutdown to Hot Standby

5.25 NextEra Point Beach Letter, "Reactor Vessel Surveillance Program Request to Change Reactor Vessel Surveillance Specimen Withdrawal Schedule," dated January 19, 2010

5.26 Point Beach Nuclear Plan Design Guide DG-I01, Instrument Setpoint Methodology

Figure 1
RCS PRESSURE-TEMPERATURE LIMITS FOR HEATUP

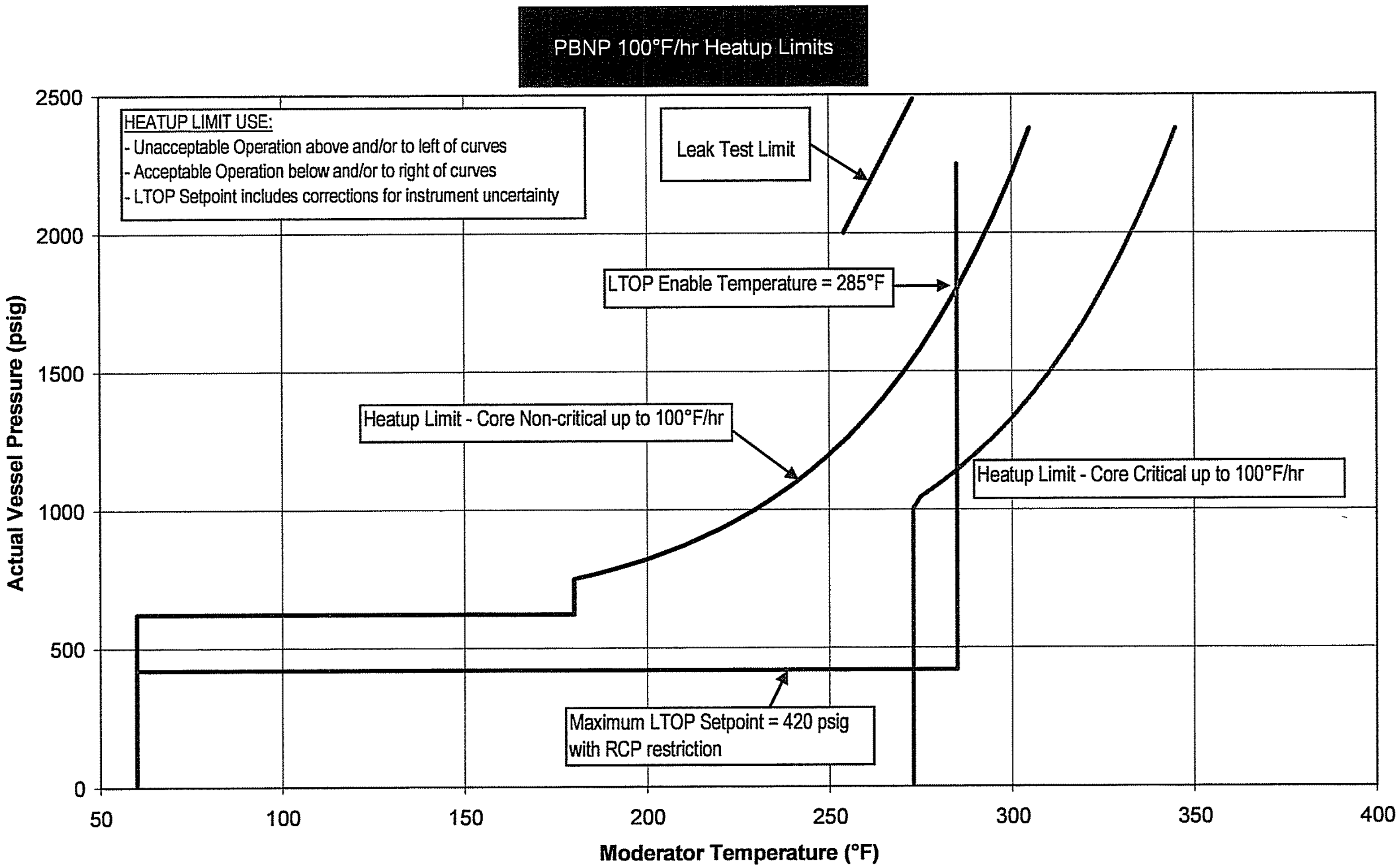


Figure 2
RCS PRESSURE-TEMPERATURE LIMITS FOR COOLDOWN

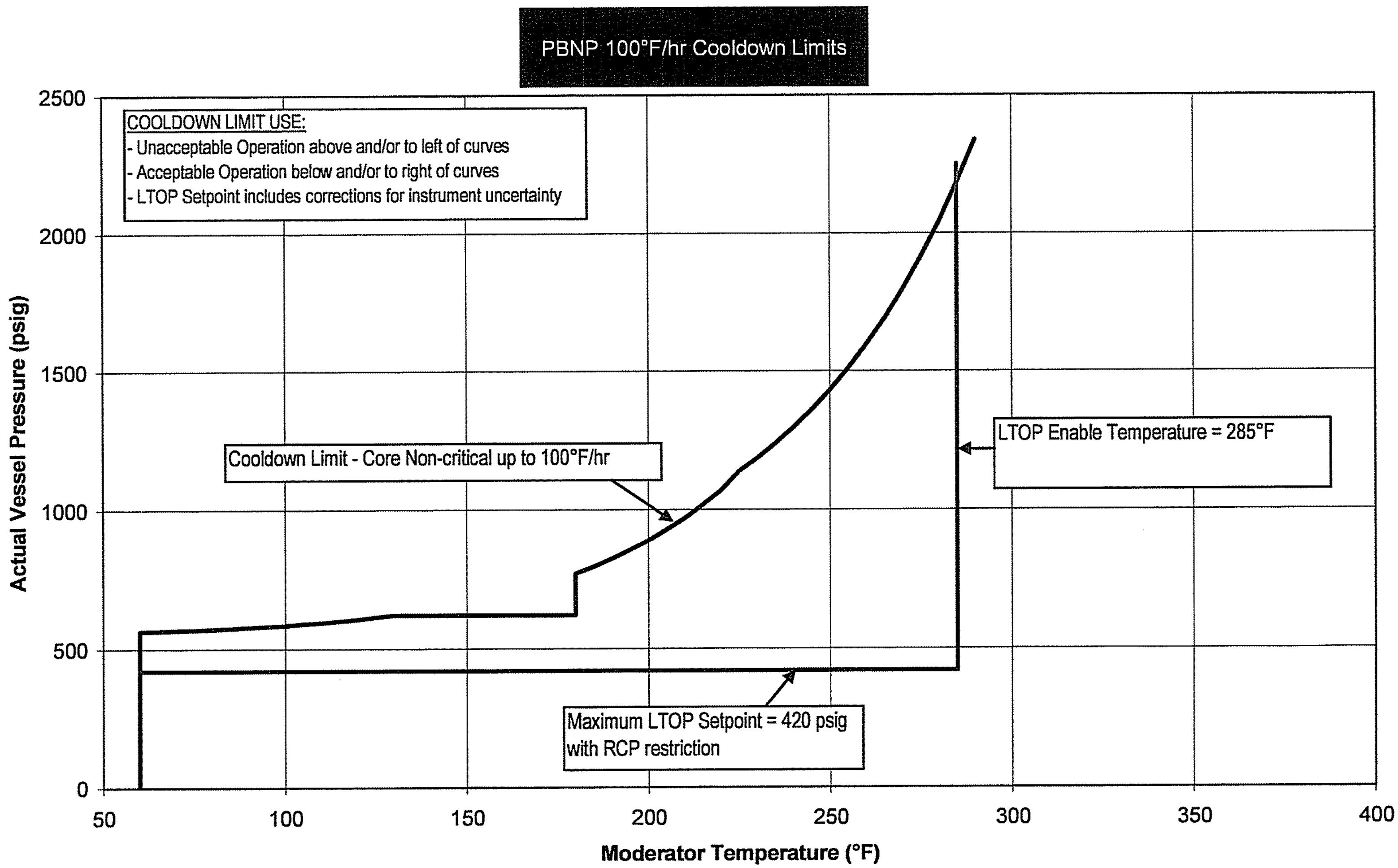


TABLE 1 (**)
POINT BEACH NUCLEAR PLANT UNIT 1
REACTOR VESSEL SURVEILLANCE CAPSULE REMOVAL SCHEDULE

Capsule Identification Letter	Approximate Removal Date*
V	September 1972 (actual)
S	December 1975 (actual)
R	October 1977 (actual)
T	March 1984 (actual)
P	April 1994 (actual)
N	Standby

* The actual removal dates will be adjusted to coincide with the closest scheduled plant refueling outage or major reactor plant shutdown.

** During the period of extended operation, reactor vessel surveillance capsules will be removed and tested in accordance with the schedule contained in the most recently NRC-approved Pressurized Water Reactor Owners Group (PWROG) Master Integrated Reactor Vessel Surveillance Program (MIRVSP) Document. (Ref. 5.5)(Ref 5.25)

TABLE 2 (**)
POINT BEACH NUCLEAR PLANT UNIT 2
REACTOR VESSEL SURVEILLANCE CAPSULE REMOVAL SCHEDULE

Capsule Identification Letter	Approximate Removal Date*
V	November 1974 (actual)
T	March 1977 (actual)
R	April 1979 (actual)
S	October 1990 (actual)
P	June 1997 (actual)
N	Standby
A	April 2022

* The actual removal dates will be adjusted to coincide with the closest scheduled plant refueling outage or major reactor plant shutdown.

** During the period of extended operation, reactor vessel surveillance capsules will be removed and tested in accordance with the schedule contained in the most recently NRC-approved Pressurized Water Reactor Owners Group (PWROG) Master Integrated Reactor Vessel Surveillance Program (MIRVSP) Document. (Ref. 5.5)(Ref 5.25).

TABLE 3
POINT BEACH UNIT 1 RPV BELTLINE 35.9 EFPY VALUES(E)

Based on Westinghouse Report WCAP-15976, Revision 1, "Point Beach Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," (Ref 5.2). Note that the estimated fluence at a specific point in time is not linearly interpolated between zero and the estimated fluence at 36.9 EFPY, due to changes in core design at certain points in the operating history of the unit. Although the analysis in WCAP-15976 is based on 36.9 EFPY, the applicability of the analysis is now 35.9 EFPY per LTR-PCAM-08-57, "Point Beach Units 1 and 2 EPU P-T Limit Curve Applicability Determination and Related Calculations" (Ref 5.8).

Vessel Manufacturer:	Babcock & Wilcox
Plate and Weld Thickness (without cladding):	6.5", without clad (D)

Component Description	Heat or Heat/Lot	35.9 EFPY (E) Inside Surface Fluence (E19 n/cm^2)	35.9 EFPY (E) 1/4T Fluence (E19 n/cm^2) (B)	35.9 EFPY (E) 1/4T Fluence Factor (C)	35.9 EFPY (E) 3/4T Fluence (E19 n/cm^2) (B)	35.9 EFPY (E) 3/4T Fluence Factor (C)
Nozzle Belt Forging	122P237	0.25	0.17	0.53	0.08	0.37
Intermediate Shell Plate	A9811-1	3.38	2.29	1.22	1.05	1.01
Lower Shell Plate	C1423-1	3.04	2.06	1.20	0.94	0.98
Nozzle Belt to Intermed. Shell Circ Weld (100%)	8T1762 (SA-1426)	0.25	0.17	0.53	0.08	0.37
Intermediate Shell Long Seam (ID 27%)	1P0815 (SA-812)	2.19	1.48	1.11	N/A	N/A
Intermediate Shell Long (A) Seam (OD 73%)	1P0661 (SA-775)	2.19	N/A	N/A	0.68	0.89
Intermed. to Lower Shell Circ. Weld (100%)	71249 (SA-1101)	3.05	2.07	1.20	0.95	0.99
Lower Shell Long Seam (A) (100%)	61782 (SA-847)	2.08	1.41	1.10	0.65	0.88

Footnotes:

(A) Limiting material

(B) From an inside surface fluence value (not including cladding), fluence is attenuated to a desired thickness using equation (3) of Regulatory Guide 1.99, Revision 2: $f = f_{surf} \times e^{-0.24x}$, where f_{surf} is expressed in units of E19 n/cm^2, E>1 MeV, and x is the desired depth in inches into the vessel wall. For example, for the nozzle belt forging, heat no. 122P237, at 35.9 EFPY, at a depth of 1/4 of the 6.5" vessel wall (1.625"), $f = 0.25 \times e^{-0.24(1.625)} = 0.17$ E19 n/cm^2.

(C) The dimensionless fluence factor is calculated using the fluence factor formula from equation (2) of Regulatory Guide 1.99, Revision 2: $ff = f^{(0.28 - 0.10 \log f)}$, where f is the fluence in units of E19 n/cm^2. For example, the 35.9 EFPY 1/4T fluence factor for nozzle belt forging, heat no. 122P237, $ff = 0.17^{(0.28 - 0.10 \log 0.17)} = 0.53$.

(D) Instruction Manual, 132-Inch I.D. Reactor Pressure Vessel, Babcock & Wilcox, September 1969 .

(E) EFPY value listed here is based on various reactor fuel management strategies and reactor power levels. See WCAP-15976, Revision 1 (Ref 5.2) for discussion of EFPY values. The 36.9 EFPY values listed in WCAP-15976, Revision 1, are now applicable to 35.9 EFPY.

TABLE 4
POINT BEACH UNIT 2 RPV BELTLINE 35.9 EFPY VALUES(E)

Based on Westinghouse Report WCAP-15976, Revision 1, "Point Beach Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," (Ref 5.2). Note that the estimated fluence at a specific point in time is not linearly interpolated between zero and the estimated fluence at 36.9 EFPY, due to changes in core design at certain points in the operating history of the unit. Although the analysis in WCAP-15976 is based on 36.9 EFPY, the applicability of the analysis is now 35.9 EFPY per LTR-PCAM-08-57, "Point Beach Units 1 and 2 EPU P-T Limit Curve Applicability Determination and Related Calculations" (Ref 5.8).

Vessel Manufacturer:	Babcock & Wilcox and Combustion Engineering
Plate and Weld Thickness (without cladding):	6.5", without clad (D)

Component Description	Heat or Heat/Lot	35.9 EFPY (E) Inside Surface Fluence (E19 n/cm^2)	35.9 EFPY (E) 1/4T Fluence (E19 n/cm^2) (B)	35.9 EFPY (E) 1/4T Fluence Factor (C)	35.9 EFPY (E) 3/4T Fluence (E19 n/cm^2) (B)	35.9 EFPY (E) 3/4T Fluence Factor (C)
Nozzle Belt Forging	123V352	0.34	0.23	0.60	0.11	0.44
Intermediate Shell Forging (A)	123V500	3.38	2.29	1.22	1.05	1.01
Lower Shell Forging	122W195	3.30	2.23	1.22	1.02	1.01
Nozzle Belt to Intermed. Shell Circ Weld (100%)	21935	0.34	0.23	0.60	0.11	0.44
Intermed. to Lower Shell Circ Weld (100%) (A)	72442 (SA-1484)	3.13	2.12	1.20	0.97	0.99

Footnotes:

(A) Limiting Material

(B) From an inside surface fluence value (not including cladding), fluence is attenuated to a desired thickness using equation (3) of Regulatory Guide 1.99, Revision 2: $f = f_{surf} \times e^{-0.24x}$, where f_{surf} is expressed in units of E19 n/cm^2, E>1 MeV, and x is the desired depth in inches into the vessel wall. For example, for the nozzle belt forging, heat no. 123V352, at 35.9 EFPY, at a depth of 1/4 of the 6.5" vessel wall (1.625"), $f = 0.34 \times e^{-0.24(1.625)} = 0.23$ E19 n/cm^2.

(C) The dimensionless fluence factor is calculated using the fluence factor formula from equation (2) of Regulatory Guide 1.99, Revision 2: $ff = f^{(0.28 - 0.10 \log f)}$, where f is the fluence in units of E19 n/cm^2. For example, the 35.9 EFPY 1/4T fluence factor for nozzle belt forging, heat no. 123V352, $ff = 0.23^{(0.28 - 0.10 \log 0.23)} = 0.60$.

(D) Instruction Manual, Reactor Vessel, Point Beach Nuclear Plant No. 2, Combustion Engineering, CE Book #4869, October 1970.

(E) EFPY value listed here is based on various reactor fuel management strategies and reactor power levels. See WCAP-15976, Revision 1 (Ref 5.2) for discussion of EFPY values. The 36.9 EFPY values listed in WCAP-15976, Revision 1, are now applicable to 35.9 EFPY.

TABLE 5
POINT BEACH UNIT 1 RPV 1/4T BELTLINE MATERIAL ADJUSTED REFERENCE TEMPERATURES AT 35.9 EFPY[(H)]

Unless otherwise noted, all ART input data obtained from BAW-2325, "Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity," May 1998. (Ref. 5.6) and WCAP-15976, Revision 1, "Point Beach Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," (Ref 5.2). Although the analysis in WCAP-15976 is based on 36.9 EFPY, the applicability of the analysis is now 35.9 EFPY per LTR-PCAM-08-57, "Point Beach Units 1 and 2 EPU P-T Limit Curve Applicability Determination and Related Calculations" (Ref 5.8).

Vessel Manufacturer:	Babcock & Wilcox
Plate and Weld Thickness (without cladding):	6.5", without clad[(F)]

Component Description	Heat or Heat/Lot	Initial RT_{NDT} (°F)	%Cu	%Ni	CF	CF Method	1/4T 35.9 EFPY[(H)] Fluence Factor[(A)]	ΔRT_{NDT} (°F)	σ_I	σ_Δ	Margin (°F)	ART (°F)[(E)]
Nozzle Belt Forging	122P237	+50	0.11	0.82	77	Table	0.53	40.8	0	17	34	125
Intermediate Shell Plate	A9811-1	+1	0.20	0.06	88	Table	1.22	107.4	26.9	17	63.64	172
"	"	"			79.3	Surv. Data[(B)]	"	96.7	"	8.5	56.42	154
Lower Shell Plate	C1423-1	+1	0.12	0.07	55.3	Table	1.20	66.4	26.9	17	63.64	131
"	"	"			35.8	Surv. Data[(B)]	"	43.0	"	8.5	56.42	100
Nozzle Belt to Intermed. Shell Circ Weld (100%)	8T1762 (SA-1426)	-5	0.19	0.57	152.4	Table	0.53	80.8	19.7	28	68.47	144
Intermediate Shell Long Seam (ID 27%)	1P0815 (SA-812)	-5	0.17	0.52	138.2	Table	1.11	153.4	19.7	28	68.47	217
Intermediate Shell Long Seam (OD 73%)	1P0661 (SA-775)	-5	0.17	0.64	157.6	Table	N/A	N/A	19.7	28	68.47	N/A
Intermed. to Lower Shell Circ. Weld (100%)	71249 (SA-1101)	+10	0.23	0.59	167.6	Table[(C)]	1.20	201.1	0	28	56	267
Lower Shell Long Seam (100%)	61782 (SA-847)	-5	0.23	0.52	157.4	Table	1.10	173.1	19.7	28	68.47	237
"	"	"			163.3	Surv. Data[(D)]	"	179.6	"	14	48.34	223

Footnotes:

(A) See Table 3
(B) Credible Surveillance Data; see BAW-2325 for evaluation.
(C) Non-credible surveillance data; see BAW-2325 for evaluation. Table CF conservative because difference between ratio-adjusted measure ΔRT_{NDT} and predicted ΔRT_{NDT} based on Table CF is less than 2σ (56°F).
(D) Credible Surveillance Data; see WE Calculation Addendum 98-0156-00-A, "Evaluation of New Surveillance Data on Chemistry Factor for Weld Wire Heat 61782, Point Beach Unit 1," (Ref.5.3) utilizing latest time-weighted temperature data for Point Beach Unit 1, which supersedes BAW-2325.
(E) Adjusted reference temperature (ART) calculated per Regulatory Guide 1.99, Rev. 2. ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin, where ΔRT_{NDT} = Chemistry Factor × Fluence Factor, and Margin = $2(\sigma_I^2 + \sigma_\Delta^2)^{0.5}$, with σ_I defined as the standard deviation of the Initial RT_{NDT} and σ_Δ defined as the standard deviation of ΔRT_{NDT}. Calculated ART values are rounded to the nearest °F in accordance with the rounding-off method of ASTM Practice E29.
(F) Instruction Manual, 132-Inch I.D. Reactor Pressure Vessel, Babcock & Wilcox, September 1969.
(G) Deleted.
(H) EFPY value listed here is based on various reactor fuel management strategies and reactor power levels. See WCAP-15976, Revision 1 (Ref 5.2) for discussion of EFPY values. The 36.9 EFPY values listed in WCAP-15976, Revision 1, are now applicable to 35.9 EFPY.

TABLE 6
POINT BEACH UNIT 2 RPV 1/4T BELTLINE MATERIAL ADJUSTED REFERENCE TEMPERATURES AT 35.9 EFPY [(I)]

Unless otherwise noted, all ART input data obtained from BAW-2325, "Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity," May 1998 (Ref. 5.6) and WCAP-15976, Revision 1, "Point Beach Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," (Ref 5.2). Although the analysis in WCAP-15976 is based on 36.9 EFPY, the applicability of the analysis is now 35.9 EFPY per LTR-PCAM-08-57, "Point Beach Units 1 and 2 EPU P-T Limit Curve Applicability Determination and Related Calculations" (Ref 5.8).

Vessel Manufacturer:	Babcock & Wilcox and Combustion Engineering
Plate and Weld Thickness (without cladding):	6.5", without clad[(F)]

Component Description	Heat or Heat/Lot	Initial RT_{NDT} (°F)	%Cu	%Ni	CF	CF Method	1/4T 35.9 [(I)] EFPY Fluence Factor[(A)]	ΔRT_{NDT} (°F)	σ_I	σ_Δ	Margin (°F)	ART (°F)[(E)]
Nozzle Belt Forging	123V352	+40	0.11	0.73	76	Table	0.60	45.6	0	17	34	120
Intermediate Shell Forging	123V500	+40	0.09	0.70	58	Table[(B)]	1.22	70.8	0	17	34	145
Lower Shell Forging	122W195	+40	0.05	0.72	31	Table	1.22	37.8	0	17	34	112
"	"	"			42.8	Surv. Data[(C)]	"	52.5	"	8.5	17	110
Nozzle Belt to Intermed. Shell Circ Weld (100%)	21935	-56	0.18	0.70	170	Table[(H)]	0.60	102	17	28	65.51	112
Intermed. to Lower Shell Circ. Weld (100%)	72442 (SA-1484)	-5	0.26	0.60	180	Table[(D)]	1.20	216.0	19.7	28	68.47	280

Footnotes:
(A) See Table 4
(B) Non-credible surveillance data; see BAW-2325 for evaluation. Table CF conservative because difference between measured ΔRT_{NDT} and predicted ΔRT_{NDT} based on Table CF is less than 2σ (34°F)
(C) Credible surveillance data; see BAW-2325 for evaluation.
(D) Non-credible surveillance data; Table CF value based on best-estimate chemistry is higher than best fit calculated using surveillance data, and therefore, conservative.
(E) Adjusted reference temperature (ART) calculated per Regulatory Guide 1.99, Rev. 2. ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin, where ΔRT_{NDT} = Chemistry Factor × Fluence Factor, and Margin = $2(\sigma_I^2 + \sigma_\Delta^2)^{0.5}$, with σ_I defined as the standard deviation of the Initial RT_{NDT}, and σ_Δ defined as the standard deviation of ΔRT_{NDT}. For example, for nozzle belt forging, heat no. 123V352, ART = 40 + (76 x 0.60) + 34 = 120°F. Calculated ART values are rounded to the nearest °F in accordance with the rounding-off method of ASTM Practice E29.
(F) Instruction Manual, Reactor Vessel, Point Beach Nuclear Plant Unit 2, Combustion Engineering, CE Book #4869, October 1970.
(G) Deleted.
(H) Table CF value based on best-estimate chemistry data from CEOG Report "Best Estimate Copper and Nickel Values in CE Fabricated Reactor Vessel Welds," CE NPSD-1039, Revision 2, Final Report, June 1997 (Ref.5.7).
(I) EFPY value listed here is based on various reactor fuel management strategies and reactor power levels. See WCAP-15976, Revision 1 (Ref 5.2) for discussion of EFPY values. The 36.9 EFPY values listed in WCAP-15976, Revision 1, are now applicable to 35.9 EFPY.

TABLE 7
POINT BEACH UNIT 1 RPV 3/4T BELTLINE MATERIAL ADJUSTED REFERENCE TEMPERATURES AT 35.9 EFPY (H)

Unless otherwise noted, all ART input data obtained from BAW-2325, "Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity," May 1998 (Ref. 5.6) and WCAP-15976, Revision 1, "Point Beach Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," (Ref 5.2). Although the analysis in WCAP-15976 is based on 36.9 EFPY, the applicability of the analysis is now 35.9 EFPY per LTR-PCAM-08-57, "Point Beach Units 1 and 2 EPU P-T Limit Curve Applicability Determination and Related Calculations" (Ref 5.8).

Vessel Manufacturer:	Babcock & Wilcox
Plate and Weld Thickness (without cladding):	6.5", without clad(F)

Component Description	Heat or Heat/Lot	Initial RT_{NDT} (°F)	%Cu	%Ni	CF	CF Method	3/4T 35.9 EFPY(H) Fluence Factor(A)	ΔRT_{NDT} (°F)	σ_I	σ_Δ	Margin (°F)	ART (°F)(E)
Nozzle Belt Forging	122P237	+50	0.11	0.82	77	Table	0.37	28.5	0	17	34	113
Intermediate Shell Plate	A9811-1	+1	0.20	0.06	88	Table	1.01	88.9	26.9	17	63.64	154
"	"	"			79.3	Surv. Data(B)	"	80.1	"	8.5	56.42	138
Lower Shell Plate	C1423-1	+1	0.12	0.07	55.3	Table	0.98	54.2	26.9	17	63.64	119
"	"	"			35.8	Surv. Data(B)	"	35.1	"	8.5	56.42	93
Nozzle Belt to Intermed. Shell Circ Weld (100%)	8T1762 (SA-1426)	-5	0.19	0.57	152.4	Table	0.37	56.4	19.7	28	68.47	120
Intermediate Shell Long Seam (ID 27%)	1P0815 (SA-812)	-5	0.17	0.52	138.2	Table	N/A	N/A	19.7	28	68.47	N/A
Intermediate Shell Long Seam (OD 73%)	1P0661 (SA-775)	-5	0.17	0.64	157.6	Table	0.89	140.3	19.7	28	68.47	204
Intermed. To Lower Shell Circ. Weld (100%)	71249 (SA-1101)	+10	0.23	0.59	167.6	Table(C)	0.99	165.9	0	28	56	232
Lower Shell Long Seam (100%)	61782 (SA-847)	-5	0.23	0.52	157.4	Table	0.88	138.5	19.7	28	68.47	202
"	"	"			163.3	Surv. Data(D)	"	143.7	"	14	48.34	187

Footnotes:

(A) See Table 3.

(B) Credible Surveillance Data; see BAW-2325 for evaluation.

(C) Non-credible surveillance data; see BAW-2325 for evaluation. Table CF conservative because difference between ratio-adjusted measured ΔRT_{NDT} are predicted ΔRT_{NDT} based on Table CF is less than 2σ (56°F).

(D) Credible Surveillance Data; see WE Calculation Addendum 98-0156-00-A, "Evaluation of New Surveillance Data on Chemistry Factor for Weld Wire Heat 61782, Point Beach Unit 1," utilizing latest time-weighted temperature data for Point Beach Unit 1, which supersedes BAW-2325.

(E) Adjusted reference temperature (ART) calculated per Regulatory Guide 1.99, Rev. 2. ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin, where ΔRT_{NDT} = Chemistry Factor × Fluence Factor, and Margin = $2(\sigma_I^2 + \sigma_\Delta^2)^{0.5}$, with σ_I defined as the standard deviation of the Initial RT_{NDT}, and σ_Δ defined as the standard deviation of ΔRT_{NDT}. For example, for nozzle belt forging, heat no. 122P237, ART = 50 + (77 × 0.37) + 34 = 113°F. Calculated ART values are rounded to the nearest °F in accordance with the rounding-off method of ASTM Practice E29.

(F) Instruction Manual, 132-Inch I.D. Reactor Pressure Vessel, Babcock & Wilcox, September 1969.

(G) Deleted.

(H) EFPY value listed here is based on various reactor fuel management strategies and reactor power levels. See WCAP-15976, Revision 1 (Ref 5.2) for discussion of EFPY values. The 36.9 EFPY values listed in WCAP-15976, Revision 1, are now applicable to 35.9 EFPY.

TABLE 8
POINT BEACH UNIT 2 RPV 3/4T BELTLINE MATERIAL ADJUSTED REFERENCE TEMPERATURES AT 35.9 EFPY [(I)]

Unless otherwise noted, all ART input data obtained from BAW-2325, "Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity," May 1998 (Ref. 5.6) and WCAP-15976, Revision 1, "Point Beach Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," (Ref 5.2). Although the analysis in WCAP-15976 is based on 36.9 EFPY, the applicability of the analysis is now 35.9 EFPY per LTR-PCAM-08-57, "Point Beach Units 1 and 2 EPU P-T Limit Curve Applicability Determination and Related Calculations" (Ref 5.8).

Vessel Manufacturer:	Babcock & Wilcox and Combustion Engineering
Plate and Weld Thickness (without cladding):	6.5", without clad[(F)]

Component Description	Heat or Heat/Lot	Initial RT_{NDT} (°F)	%Cu	%Ni	CF	CF Method	3/4T 35.9 EFPY[(I)] Fluence Factor[(A)]	ΔRT_{NDT} (°F)	σ_I	σ_Δ	Margin (°F)	ART (°F)[(E)]
Nozzle Belt Forging	123V352	+40	0.11	0.73	76	Table	0.44	33.4	0	17	34	107
Intermediate Shell Forging	123V500	+40	0.09	0.70	58	Table[(B)]	1.01	58.6	0	17	34	133
Lower Shell Forging	122W195	+40	0.05	0.72	31	Table	1.01	31.3	0	17	34	105
'	"	"			42.8	Surv. Data[(C)]	"	43.4	"	8.5	17	100
Nozzle Belt to Intermed. Shell Circ Weld (100%)	21935	-56	0.18	0.70	170	Table[(H)]	0.44	74.8	17	28	65.51	84
Intermed. to Lower Shell Circ. Weld (100%)	72442 (SA-1484)	-5	0.26	0.60	180	Table [(D)]	0.99	178.2	19.7	28	68.47	242

Footnotes:

(A) See Table 4.

(B) Non-credible surveillance data; see BAW-2325 for evaluation. Table CF conservative because difference between measured ΔRT_{NDT} and predicted ΔRT_{NDT} based on Table CF is less than 2σ (56°F).

(C) Credible surveillance data; see BAW-2325 for evaluation.

(D) Non-credible surveillance data; Table CF value based on best-estimate chemistry is higher than best fit calculated using surveillance data, and therefore, conservative.

(E) Adjusted reference temperature (ART) calculated per Regulatory Guide 1.99, Rev. 2. ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin, where ΔRT_{NDT} = Chemistry Factor × Fluence Factor, and Margin = $2(\sigma_I^2 + \sigma_\Delta^2)^{0.5}$, with σ_I defined as the standard deviation of the Initial RT_{NDT}, and σ_Δ defined as the standard deviation of ΔRT_{NDT}. Calculated ART values are rounded to the nearest °F in accordance with the rounding-off method of ASTM Practice E29.

(F) *Instruction Manual, Reactor Vessel, Point Beach Nuclear Plant No. 2, Combustion Engineering, CE Book #4869, October 1970.*

(G) Deleted.

(H) Table CF value based on best-estimate chemistry data from CEOG Report "Best Estimate Copper and Nickel Values in CE Fabricated Reactor Vessel Welds," CE NPSD-1039, Revision 2, Final Report, June 1997

(I) EFPY value listed here is based on various reactor fuel management strategies and reactor power levels. See WCAP-15976, Revision 1 (Ref 5.2) for discussion of EFPY values. The 36.9 EFPY values listed in WCAP-15976, Revision 1, are now applicable to 35.9 EFPY.