NMC
Committed to Nuclear Excellence

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NRC 2001-075

10 CFR 50.90

November 5, 2001

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

Ladies/Gentlemen:

DOCKETS 50-266 AND 50-301
SUBMITTAL OF REVISION TO
PRESSURE TEMPERATURE LIMITS REPORT
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

In accordance with Point Beach Improved Technical Specification (ITS) 5.6.5.c, Nuclear Management Company, LLC (NMC) is hereby submitting a revision of the Pressure Temperature Limits Report (PTLR) for Point Beach Nuclear Plant, Units 1 and 2. The changes reflected in the enclosed PTLR were made in accordance with the requirements of 10 CFR 50.59. Attachment I to this letter provides a description of the changes in the enclosed PTLR.

The NRC approved the PTLR methodology on July 23, 2001, and approved the Point Beach ITS and PTLR in Amendments 201 and 206 on August 8, 2001. Point Beach is planning to implement the ITS, coincident with implementation of the PTLR, on November 20, 2001.

Please contact us if you have any questions or require additional information.

Sincerely,

A J Cayia

A. J. Cayia
Plant Manager

JG/tjv

A001

Rec'd at NRC/DCD 12/12/01

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Attachment: I Description of Changes

Enclosure: Point Beach Pressure Temperature Limits Report

cc: NRC Regional Administrator
NRC Resident Inspector
NRC Project Manager
PSCW

DESCRIPTION OF CHANGES
PRESSURE TEMPERATURE LIMITS REPORT
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

1.0 INTRODUCTION

In accordance with Point Beach Improved Technical Specification (ITS) 5.6.5.c, Nuclear Management Company, LLC (NMC) is submitting the following revisions to the Pressure Temperature Limits Report (PTLR) for Point Beach Nuclear Plant, Units 1 and 2. The PTLR enclosed with this letter will be the initial version implemented with Improved Technical Specifications and therefore designated as Revision 0.

2.0 DESCRIPTION

Changes to the previously submitted sample PTLR are shown and described below. Additions to the report are shown double-underlined. Deletions are shown with strikethrough.

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. Revisions to the PTLR shall be provided to the NRC ~~after~~ upon issuance.

This grammatical change makes the wording of the PTLR consistent with the wording of Improved Technical Specification 5.6.5.c.

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

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

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

The change in section 2.1.1 from degrees per hour to degrees in any one hour makes the PTLR heatup and cooldown allowance consistent with the wording in Point Beach Current Technical Specification 15.3.1-B.1.

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

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.

PTLR section 2.1.2 is clarified to specify that instrument uncertainty is included in the value specified for the limit. Section 2.1.3 is changed to reflect that the minimum temperature for pressurization must be met prior to bolt up of the vessel head, since bolt up stresses the vessel and pressurization is possible once the head is bolted to the vessel.

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

2.2.1 The enable temperature for the Low Temperature Overpressure Protection System is 270°F (includes instrument uncertainty for RCS T_c wide range).

Additional applicable LCOs are added to the title of PTLR section 2.2. Further, this section is clarified to specify that instrument uncertainty for RCS T_c wide range is included in the value specified for the enable temperature.

Finally, references 5.11 and 5.12 were added to the report.

ENCLOSURE

TRM 2.2

PRESSURE TEMPERATURE LIMITS REPORT

UNIT 1 AND UNIT 2

REVISION 0

(Effective through 25.59 EFPY for Unit 1)
(Effective through 30.51 EFPY for Unit 2)

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. Revisions to the PTLR shall be provided to the NRC upon issuance.

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. All 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 all applicable limits of the safety analysis are met. All 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 (includes instrument uncertainty).

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 270°F (includes instrument uncertainty for RCS T_c wide range).

2.3 Low Temperature Overpressure Protection System Setpoints (LCO 3.4.12)

Pressurizer Power Operated Relief Valve Lift Setting Limits

The lift setting for the pressurizer Power Operated Relief Valves (PORVs) is ≤500 psig (includes instrument uncertainty).

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.

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 Point Beach 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 USNRC Regulatory Guide 1.99, Rev. 2 for Point Beach Nuclear Plant, Units 1 and 2.

4.0 SUPPLEMENTAL DATA INFORMATION AND DATA TABLES

4.1 The RT_{PTS} values for the Point Beach Nuclear Plant limiting beltline materials is 278°F for Unit 1 and 291°F for Unit 2 at 32 EFPY.

4.2 Tables

Table Number	Table Description
Table 1	Point Beach Nuclear Plant, Unit 1 Reactor Vessel Surveillance Capsule Removal Schedule
Table 2	Point Beach Nuclear Plant, Unit 2 Reactor Vessel Surveillance Capsule Removal Schedule
Table 3	Point Beach Unit 1 RPV Beltline 25.59 EFPY Fluence Values
Table 4	Point Beach Unit 2 RPV Beltline 30.51 EFPY Fluence Values
Table 5	Point Beach Unit 1 RPV 1/4t Beltline Material Adjusted Reference Temperatures at 25.59 EFPY
Table 6	Point Beach Unit 2 RPV 1/4t Beltline Material Adjusted Reference Temperatures at 30.51 EFPY
Table 7	Point Beach Unit 1 RPV 3/4t Beltline Material Adjusted Reference Temperatures at 25.59 EFPY
Table 8	Point Beach Unit 2 RPV 3/4t Beltline Material Adjusted Reference Temperatures at 30.51 EFPY

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-12794, "Reactor Cavity Neutron Measurement Program for Point Beach Unit 1," Rev. 4, February 2000

5.3 WCAP-12795, "Reactor Cavity Neutron Measurement Program for Point Beach Unit 2," Rev. 3, August 1995

5.4 EPRI TR-107450, "P-T Calculator for Windows, Version 3.0," Revision 0, December 1998

5.5 Westinghouse Report, "Pressure Mitigating Systems Transient Analysis Results," July 1977

5.6 Westinghouse Report, "Supplement to the July 1977 Report, Pressure Mitigating Systems Transient Analysis Results," September 1977

5.7 Wisconsin Electric Calculation 2000-0001, Revision 0, RCS P-T Limits and LTOP Setpoints Applicable through 32.2 EFPY - Unit 1 and 34.0 EFPY - Unit 2

5.8 Wisconsin Electric Calculation 2000-0001-00-A, Revision 0, Evaluation of P-T Limit and LTOP Applicability Date

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

FIGURE 1
RCS PRESSURE-TEMPERATURE LIMITS FOR HEATUP

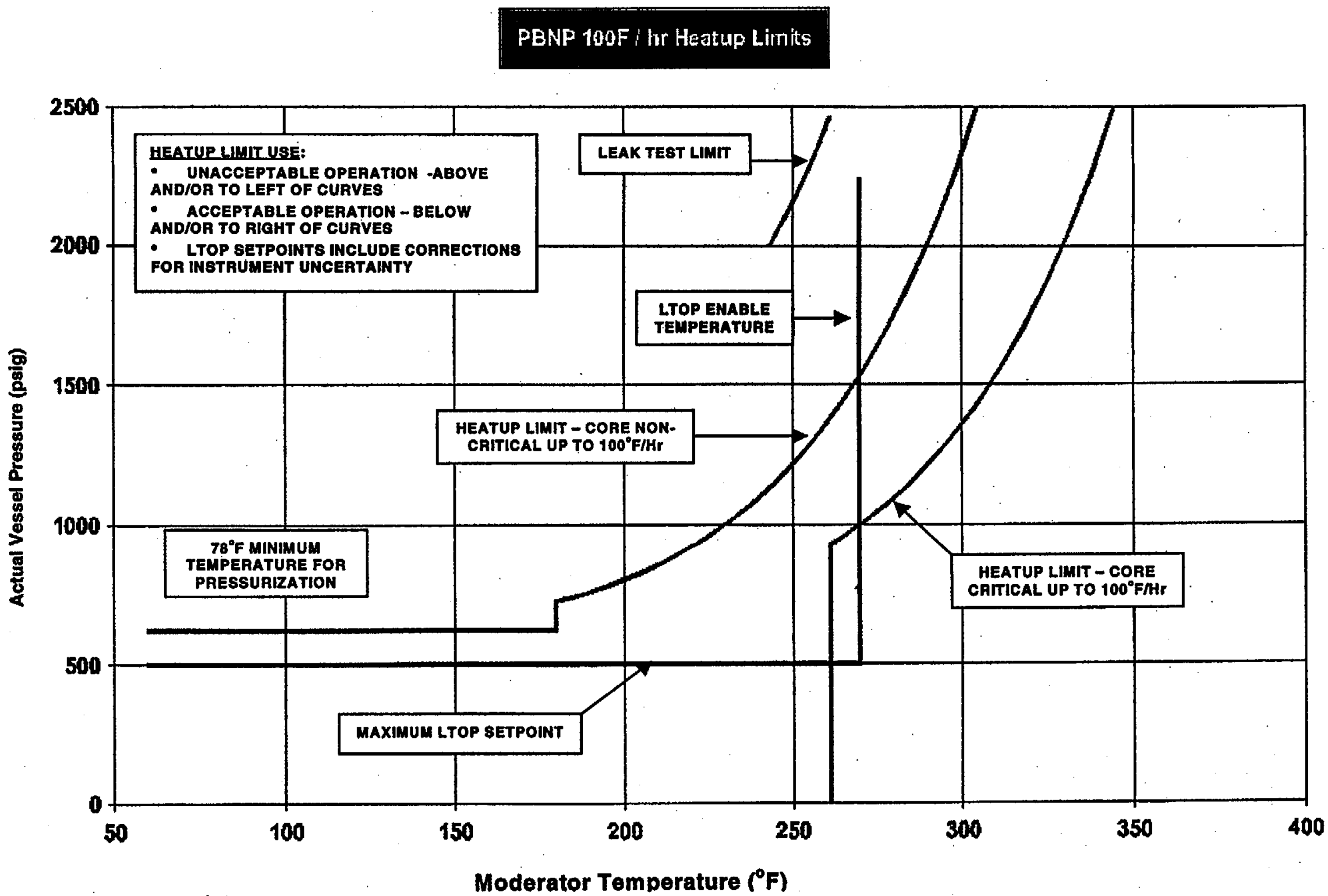

FIGURE 2
RCS PRESSURE-TEMPERATURE LIMITS FOR COOLDOWN

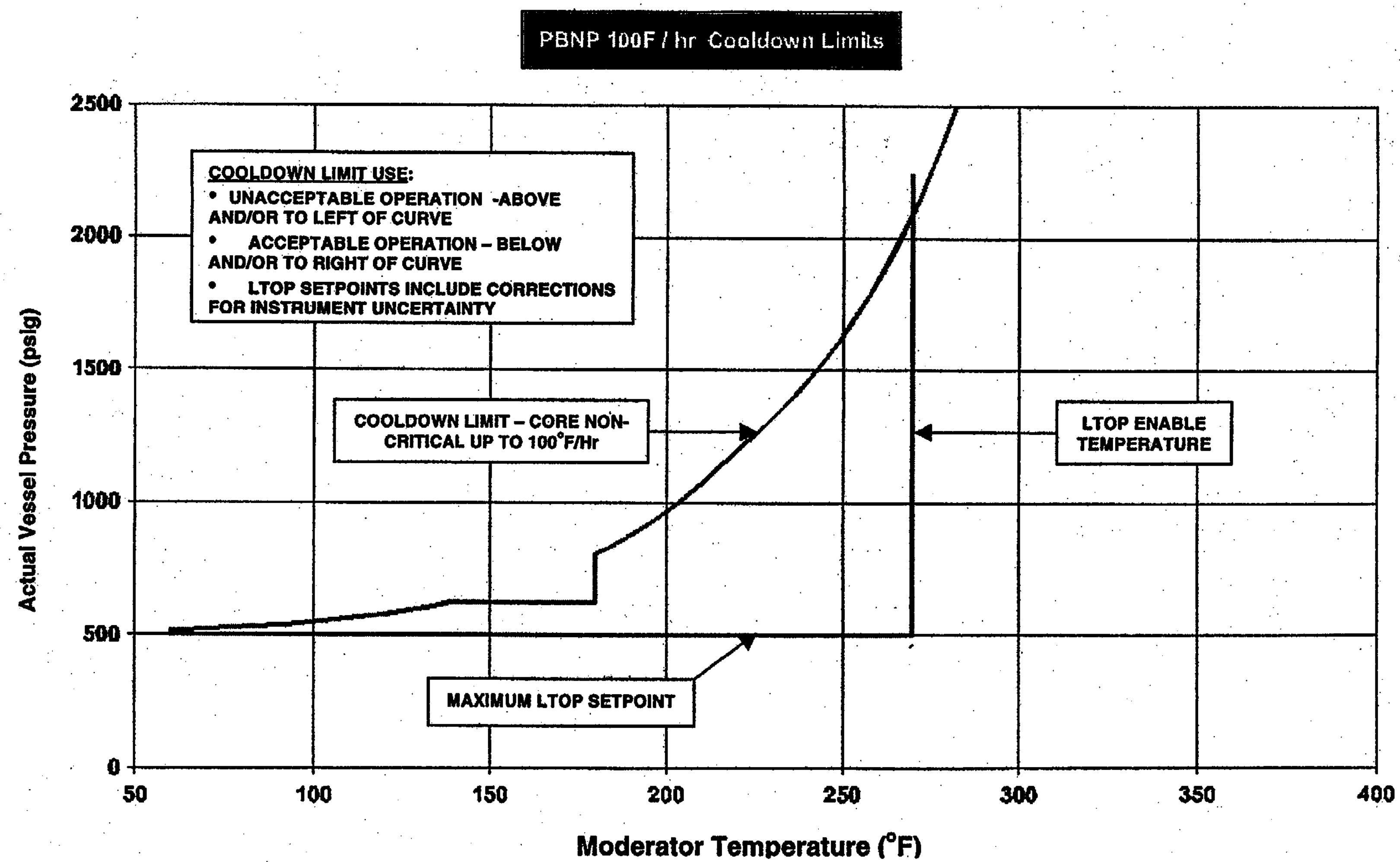

TABLE 1
POINT BEACH NUCLEAR PLANT UNIT 1
REACTOR VESSEL SURVEILLANCE CAPSULE REMOVAL SCHEDULE

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.

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

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

TABLE 3
POINT BEACH UNIT 1 RPV BELTLINE 25.59 EFPY ϕ_{Calc} (32.2 EFPY $\phi_{Best.Est.}$) VALUES[E]

Based on WCAP-12794, "Reactor Cavity Neutron Measurement Program for Wisconsin Electric Power Company Point Beach Unit 1," Rev. 4, February 2000. Note that the estimated fluence at a specific point in time is not linearly interpolated between zero and the estimated fluence at 32 EFPY, due to changes in core design at certain points in the operating history of the unit. As intermediate input to further calculations, these values are not rounded in accordance with ASTM E29 (Ref. 11).

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

Component Description	Heat or Heat/Lot	32 EFPY $\phi_{Best.Est.}$ Inside Surface Fluence (E19 n/cm^2)	32.2 EFPY $\phi_{Best.Est.}$ Inside Surface Fluence (E19 n/cm^2) [A]	32.2 EFPY $\phi_{Best.Est.}$ 1/4T Fluence (E19 n/cm^2) [B]	32.2 EFPY $\phi_{Best.Est.}$ 1/4T Fluence Factor [C]	32.2 EFPY $\phi_{Best.Est.}$ 3/4T Fluence (E19 n/cm^2) [B]	32.2 EFPY $\phi_{Best.Est.}$ 3/4T Fluence Factor [C]
Nozzle Belt Forging	122P237	0.547	0.550	0.3724	0.7269	0.1707	0.5322
Intermediate Shell Plate	A9811-1	2.64	2.65	1.794	1.160	0.8225	0.9452
Lower Shell Plate	C1423-1	2.24	2.25	1.523	1.116	0.6983	0.8993
Nozzle Belt to Intermed. Shell Circ Weld (100%)	8T1762 (SA-1426)	0.547	0.550	0.3724	0.7269	0.1707	0.5322
Intermediate Shell Long Seam (ID 27%)	1P0815 (SA-812)	1.74	1.75	1.185	1.047	N/A	N/A
Intermediate Shell Long Seam (OD 73%)	1P0661 (SA-775)	1.74	1.75	N/A	N/A	0.5431	0.8293
Intermed. to Lower Shell Circ. Weld (100%)	71249 (SA-1101)	2.24	2.25	1.523	1.116	0.6983	0.8993
Lower Shell Long Seam (100%)	61782 (SA-847)	1.54	1.55	1.049	1.013	0.4811	0.7960

Footnotes:

(A) Interpolation of neutron exposure (in units of E19 n/cm^2, E>1 MeV) to a particular value of effective full power years (EFPY) is performed based on WCAP-12794, Revision 4. For example, for the nozzle belt forging, heat no. 122P237,

$$\text{fluence} = 0.547 + \frac{(0.796 - 0.547)}{(8\text{ EFPY} - 32\text{ EFPY})} \times (32.2\text{ EFPY} - 32.0\text{ EFPY}) = 0.550\text{ E19 n/cm}^2$$

(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 32.2 EFPY, at a depth of 1/4 of the 6.5" vessel wall (1.625"), $f = 0.550 \times e^{-0.24(1.625)} = 0.3724$ 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 32.2 EFPY 1/4T fluence factor for nozzle belt forging, heat no. 122P237, $ff = 0.3724^{(0.28 - 0.10 \log 0.3724)} = 0.7269$.

(D) Instruction Manual, 132-Inch I.D. Reactor Pressure Vessel, Babcock & Wilcox, September 1969 (Ref. 12).

(E) Per Wisconsin Electric Calculation 2000-001-00-A the calculated fluence for the critical material (SA-847) occurs at 25.59 EFPY versus 32.2 EFPY $\phi_{Best.Est.}$ based upon K=ϕBεστ.Εστ./ϕCalc=0.838.

TABLE 4
POINT BEACH UNIT 2 RPV BELTLINE 30.51 EFPY ϕ_{Calc} (34.0 EFPY $\phi_{Best.Est.}$) VALUES[(E)]

Based on WCAP-12795, "Reactor Cavity Neutron Measurement Program for Wisconsin Electric Power Company Point Beach Unit 2," Rev. 3, August 1995. Note that the estimated fluence at a specific point in time is not linearly interpolated between zero and the estimated fluence at 32 EFPY, due to changes in core design at certain points in the operating history of the unit. As intermediate input to further calculations, these values are not rounded in accordance with ASTM E29 (Ref. 11).

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

Component Description	Heat or Heat/Lot	32 EFPY $\phi_{Best.Est.}$ Inside Surface Fluence (E19 n/cm^2)	34.0 EFPY $\phi_{Best.Est.}$ Inside Surface Fluence (E19 n/cm^2) [(A)]	34.0 EFPY $\phi_{Best.Est.}$ 1/4T Fluence (E19 n/cm^2) [(B)]	34.0 EFPY $\phi_{Best.Est.}$ 1/4T Fluence Factor [(C)]	34.0 EFPY $\phi_{Best.Est.}$ 3/4T Fluence (E19 n/cm^2) [(B)]	34.0 EFPY $\phi_{Best.Est.}$ 3/4T Fluence Factor [(C)]
Nozzle Belt Forging	123V352	0.548	0.5775	0.3910	0.7399	0.1792	0.5435
Intermediate Shell Forging	123V500	3.01	3.174	2.149	1.208	0.9851	0.9958
Lower Shell Forging	122W195	2.52	2.654	1.797	1.161	0.8237	0.9456
Nozzle Belt to Intermed. Shell Circ Weld (100%)	21935	0.548	0.5775	0.3910	0.7399	0.1792	0.5435
Intermed. to Lower Shell Circ Weld (100%)	72442 (SA-1484)	2.49	2.606	1.764	1.156	0.8088	0.9405

Footnotes:

(A) Interpolation of neutron exposure (in units of E19 n/cm^2, E>1 MeV) to a particular value of effective full power years (EFPY) is performed based on WCAP-12795, Revision 3. For example, for the nozzle belt forging, heat no. 123v352,

$$\text{fluence} = 0.548 + \left(\frac{0.784 - 0.548}{48\ \text{EFPY} - 32\ \text{EFPY}}\right) \times (34\ \text{EFPY} - 32\ \text{EFPY}) = 0.5775\ \text{E19 n/cm}^2$$

(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 34.0 EFPY, at a depth of 1/4 of the 6.5" vessel wall (1.625"), $f = 0.5775 \times e^{-0.24(1.625)} = 0.3910$ 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 34.0 EFPY 1/4T fluence factor for nozzle belt forging, heat no. 123V352, $ff = 0.3910^{(0.28 - 0.10 \log 0.3910)} = 0.7399$.

(D) Instruction Manual, Reactor Vessel, Point Beach Nuclear Plant No. 2, Combustion Engineering, CE Book #4869, October 1970 (Ref. 13).

(E) Per Wisconsin Electric Calculation 2000-001-00-A the calculated fluence for the critical material (SA-1484) occurs at 30.51 EFPY versus 34.0 EFPY $\phi_{Best.Est.}$ based upon K=ϕBεστ.Εστ./ϕCalc=0.921.

TABLE 5
POINT BEACH UNIT 1 RPV 1/4T BELTLINE MATERIAL ADJUSTED REFERENCE TEMPERATURES AT 25.59 EFPY ϕ_{Calc} (32.2 EFPY $\phi_{Best.Est.}$)[(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. 14), including the most recent best-estimate chemistry values for welds, applying current B&WOG mean-of-the-sources approach. All beltline materials are included for comparison.

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 32.2 EFPY $\phi_{Best.Est.}$ 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	0.7269	55.97	0	17	34	140
Intermediate Shell Plate	A9811-1	+1	0.20	0.06	88	Table	1.160	102.08	26.9	17	63.64	167
"	"	"			79.3	Surv. Data[(B)]	"	91.99	"	8.5	56.42	149
Lower Shell Plate	C1423-1	+1	0.12	0.07	55.3	Table	1.116	61.71	26.9	17	63.64	126
"	"	"			35.8	Surv. Data[(B)]	"	39.95	"	8.5	56.42	97
Nozzle Belt to Intermed. Shell Circ Weld (100%)	8T1762 (SA-1426)	-5	0.19	0.57	152.4	Table	0.7269	110.78	19.7	28	68.47	174
Intermediate Shell Long Seam (ID 27%)	1P0815 (SA-812)	-5	0.17	0.52	138.2	Table	1.047	144.70	19.7	28	68.47	208
Intermediate Shell Long Seam (OD 73%)	1P0661 (SA-775)	-5	0.17	0.64	157.6	Table	N/A	N/A	19.7	N/A	N/A	N/A
Intermed. to Lower Shell Circ. Weld (100%)	71249 (SA-1101)	+10	0.23	0.59	167.6	Table[(C)]	1.116	187.04	0	28	56	
Lower Shell Long Seam (100%)	61782 (SA-847)	-5	0.23	0.52	157.4	Table	1.013	159.45	19.7	28	68.47	223
"	"	"			163.3	Surv. Data[(D)]	"	165.42	"	14	48.34	209[(G)]

Footnotes:

(A) See Table 1.

(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. 15) 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 × 0.7269) + 34 = 140°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) By inspection, these are the limiting material properties.

(H) Per Wisconsin Electric Calculation 2000-001-00-A the calculated fluence for the critical material (SA-847) occurs at 25.59 EFPY versus 32.2 EFPY $\phi_{Best.Est.}$ based upon $K=\phi B\epsilon\sigma\tau.E\sigma\tau./\phi Calc=0.838$.

TABLE 6
POINT BEACH UNIT 2 RPV 1/4T BELTLINE MATERIAL ADJUSTED REFERENCE TEMPERATURES AT 30.51 EFPY ϕ_{Calc} (34.0 EFPY $\phi_{Best.Est.}$)[(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. 14), including the most recent best-estimate chemistry values for welds, applying current B&EWOG mean-of-the-sources approach. All beltline materials are included for comparison.

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 34.0 EFPY $\phi_{Best.Est.}$ Fluence Factor[(A)]	ΔRT_{NDT} (°F)	σ_I	σ_Δ	Margin (°F)	ART (°F)[(E)]
Nozzle Belt Forging	123V352	+40	.011	0.73	76	Table	0.7399	56.23	0	17	34	130
Intermediate Shell Forging	123V500	+40	0.09	0.70	58	Table[(B)]	1.208	70.06	0	17	34	144[(G)]
Lower Shell Forging	122W195	+40	0.05	0.72	31	Table	1.161	35.99	0	17	34	110
"	"	"			42.8	Surv. Data[(C)]	"	49.69	"	8.5	17	107
Nozzle Belt to Intermed. Shell Circ Weld (100%)	21935	-56	0.18	0.70	170	Table[(H)]	0.7399	125.78	17	28	65.51	135
Intermed. to Lower Shell Circ. Weld (100%)	72442 (SA-1484)	-5	0.26	0.60	180	Table[(D)]	1.156	208.08	19.7	28	68.47	272[(G)]

Footnotes:

(A) See Table 2.

(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 × 0.7399) + 34 = 130°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) By inspection, these are the limiting material properties.

(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. 6).

(I) Per Wisconsin Electric Calculation 2000-001-00-A the calculated fluence for the critical material (SA-1484) occurs at 30.51 EFPY versus 34.0 EFPY $\phi_{Best.Est.}$ based upon $K=\phi B\varepsilon\sigma\tau.E\sigma\tau/\phi Calc=0.921$.

TABLE 7
POINT BEACH UNIT 1 RPV 3/4T BELTLINE MATERIAL ADJUSTED REFERENCE TEMPERATURES AT 25.59 EFPY ϕ_{Calc} (32.2 EFPY $\phi_{Best.Est.}$)[(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, including the most recent best-estimate chemistry values for welds, applying current B&WOG mean-of-the-sources approach. All beltline materials are included for comparison.

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 32.2 EFPY $\phi_{Best.Est.}$ 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	0.5322	40.98	0	17	34	125
Intermediate Shell Plate	A9811-1	+1	0.20	0.06	88	Table	0.9452	83.18	26.9	17	63.64	148
"	"	"			79.3	Surv. Data[(B)]	"	74.95	"	8.5	56.42	132
Lower Shell Plate	C1423-1	+1	0.12	0.07	55.3	Table	0.8993	49.73	26.9	17	63.64	114
"	"	"			35.8	Surv. Data[(B)]	"	32.19	"	8.5	56.42	90
Nozzle Belt to Intermed. Shell Circ Weld (100%)	8T1762 (SA-1426)	-5	0.19	0.57	152.4	Table	0.5322	81.11	19.7	28	68.47	145
Intermediate Shell Long Seam (ID 27%)	1P0815 (SA-812)	-5	0.17	0.52	138.2	Table	N/A	N/A	19.7	N/A	N/A	N/A
Intermediate Shell Long Seam (OD 73%)	1P0661 (SA-775)	-5	0.17	0.64	157.6	Table	0.8293	130.70	19.7	28	68.47	194[(G)]
Intermed. To Lower Shell Circ. Weld (100%)	71249 (SA-1101)	+10	0.23	0.59	167.6	Table[(C)]	0.8993	150.72	0	28	56	217[(G)]
Lower Shell Long Seam (100%)	61782 (SA-847)	-5	0.23	0.52	157.4	Table	0.7960	125.29	19.7	28	68.47	189
"	"	"			163.3	Surv. Data[(D)]	"	129.99	"	14	48.34	173

Footnotes:

(A) See Table 1.

(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 × 0.5322) + 34 = 125°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) By inspection, these are the limiting material properties.

(H) Per Wisconsin Electric Calculation 2000-001-00-A the calculated fluence for the critical material (SA-847) occurs at 25.59 EFPY versus 32.2 EFPY $\phi_{Best.Est.}$ based upon K=φBεστ.Eστ./φCalc=0.838.

TABLE 8
POINT BEACH UNIT 2 RPV 3/4T BELTLINE MATERIAL ADJUSTED REFERENCE TEMPERATURES AT 30.51 EFPY ϕ_{Calc} (34.0 EFPY $\phi_{Best.Est.}$)[(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, including the most recent best-estimate chemistry values for welds, applying current B&WOG mean-of-the-sources approach. All beltline materials are included for comparison.

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 32.2 EFPY $\phi_{Best.Est.}$ 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	0.5435	41.31	0	17	34	115
Intermediate Shell Forging	123V500	+40	0.09	0.70	58	Table[(B)]	0.9958	57.76	0	17	34	132[(G)]
Lower Shell Forging	122W195	+40	0.05	0.72	31	Table	0.9456	29.31	0	17	34	103
	"	"			42.8	Surv. Data[(C)]	"	40.47	"	8.5	17	97
Nozzle Belt to Intermed. Shell Circ Weld (100%)	21935	-56	0.18	0.70	170	Table[(H)]	0.5435	92.40	17	28	65.51	102
Intermed. to Lower Shell Circ. Weld (100%)	72442 (SA-1484)	-5	0.26	0.60	180	Table [(D)]	0.9405	169.29	19.7	28	68.47	233[(G)]

Footnotes:

(A) See Table 2.

(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}. For example, for nozzle belt forging, heat no. 123V352, ART = 40 + (76 × 0.5435) + 34 = 115°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 No. 2, Combustion Engineering, CE Book #4869, October 1970.

(G) By inspection, these are the limiting material properties.

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

(I) Per Wisconsin Electric Calculation 2000-001-00-A the calculated fluence for the critical material (SA-1484) occurs at 30.51 EFPY versus 34.0 EFPY $\phi_{Best.Est.}$ based upon K=ϕBεστ.Εστ./ϕCalc=0.921.