FPL Energy Point Beach, LLC, 6610 Nuclear Road, Two Rivers, WI 54241

FPL Energy

Point Beach Nuclear Plant

May 30, 2008

NRC 2008-0029
TS 5.6.5

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

Point Beach Nuclear Plant, Units 1 and 2
Dockets 50-266 and 50-301
Renewed License Nos. DPR-24 and DPR-27

Pressure and Temperature Limits Report

In accordance with Technical Specification 5.6.5, enclosed are Revision 4 and Revision 5 of the Pressure and Temperature Limits Report for Point Beach Nuclear Plant, Units 1 and 2.

This letter contains no new commitments.

Very truly yours,

FPL Energy Point Beach, LLC

James H. McCarthy
Site Vice President

Enclosures

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

ENCLOSURE 1

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

PRESSURE TEMPERATURE LIMITS REPORT

REVISION 4, ISSUED APRIL 28, 2008

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. Based upon fluence values in Westinghouse report LTR-REA-04-64 (Ref 5.15), this PTLR is effective for 36.9 EFPY (approximately 2015). (Ref 5.19)

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."

EFPY values listed in this procedure are estimates based on the following past and assumed future reactor power and fuel management strategy for Unit 1 (limiting vessel):

Reactor Fuel Management Strategy	Reactor Power (MWt)
Low Leakage Without Hafnium Rods (Hafnium rods inserted from 5/89 (BOC 17) to 10/08)	1518.5 - startup to 02/03/2003 1540.0 - 02/03/2003 to 10/2008 1678.0 - 10/2008 to EOLE

Power operation outside of these future assumptions is acceptable. However, the effect on EFPY values would need to be evaluated.

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.

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 <u>Low Temperature Overpressure Protection System Enable Temperature (LCO 3.4.6, 3.4.7, 3.4.10 and 3.4.12)</u>

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 <u>Low Temperature Overpressure Protection System Setpoints (LCO 3.4.12)</u>

Pressurizer Power-Operated Relief Valve Lift Setting Limits

The lift setting 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)

3.0 <u>REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM</u>

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.

4.0 SUPPLEMENTAL DATA INFORMATION

The RT_{PTS} values for the PBNP limiting beltline materials at 36.9 EFPY is:

- Unit 1 - Intermediate to Lower Shell Circ Weld = 277°F; Lower Shell Axial Weld = 234°F (Ref. 5.8, Table 13)
- Unit 2 - Intermediate to Lower Shell Circ Weld = 292°F; Intermediate Shell Forging = 149°F (Ref. 5.8, Table 21)

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 Deleted

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 WCAP-16274-NP, "Evaluation of Pressurized Thermal Shock for Point Beach Units 1 and 2," Revision 0, June 2004

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 dated October 18, 2007 issuing 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)

5.15 Westinghouse Report LTR-REA-04-64, "Pressure Vessel Neutron Exposure Evaluation Point Beach Units 1 and 2," dated June 2004 (Westinghouse Letter WEP-04-107)

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

Figure 1
RCS PRESSURE-TEMPERATURE LIMITS FOR HEATUP

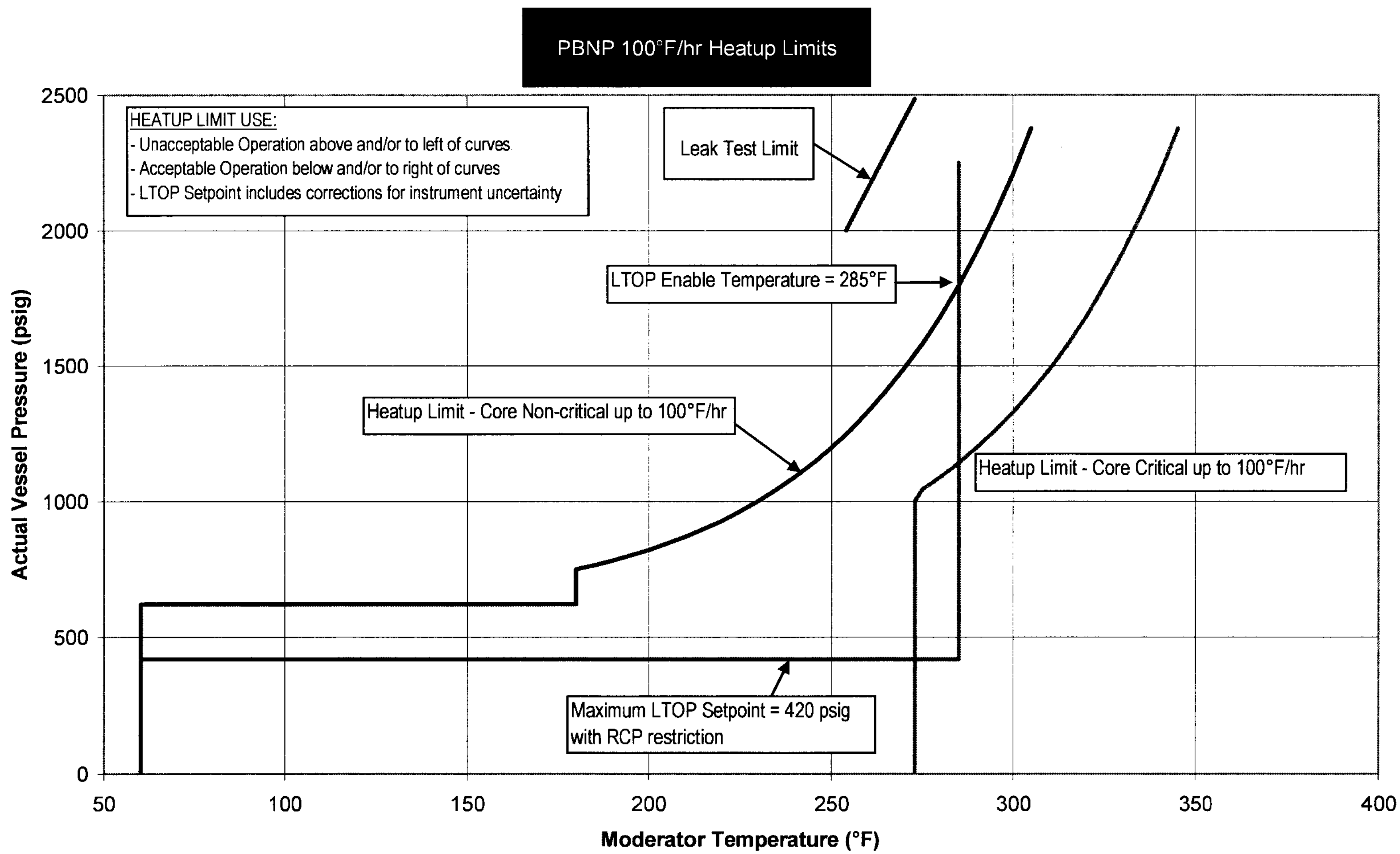

Figure 2
RCS PRESSURE-TEMPERATURE LIMITS FOR COOLDOWN

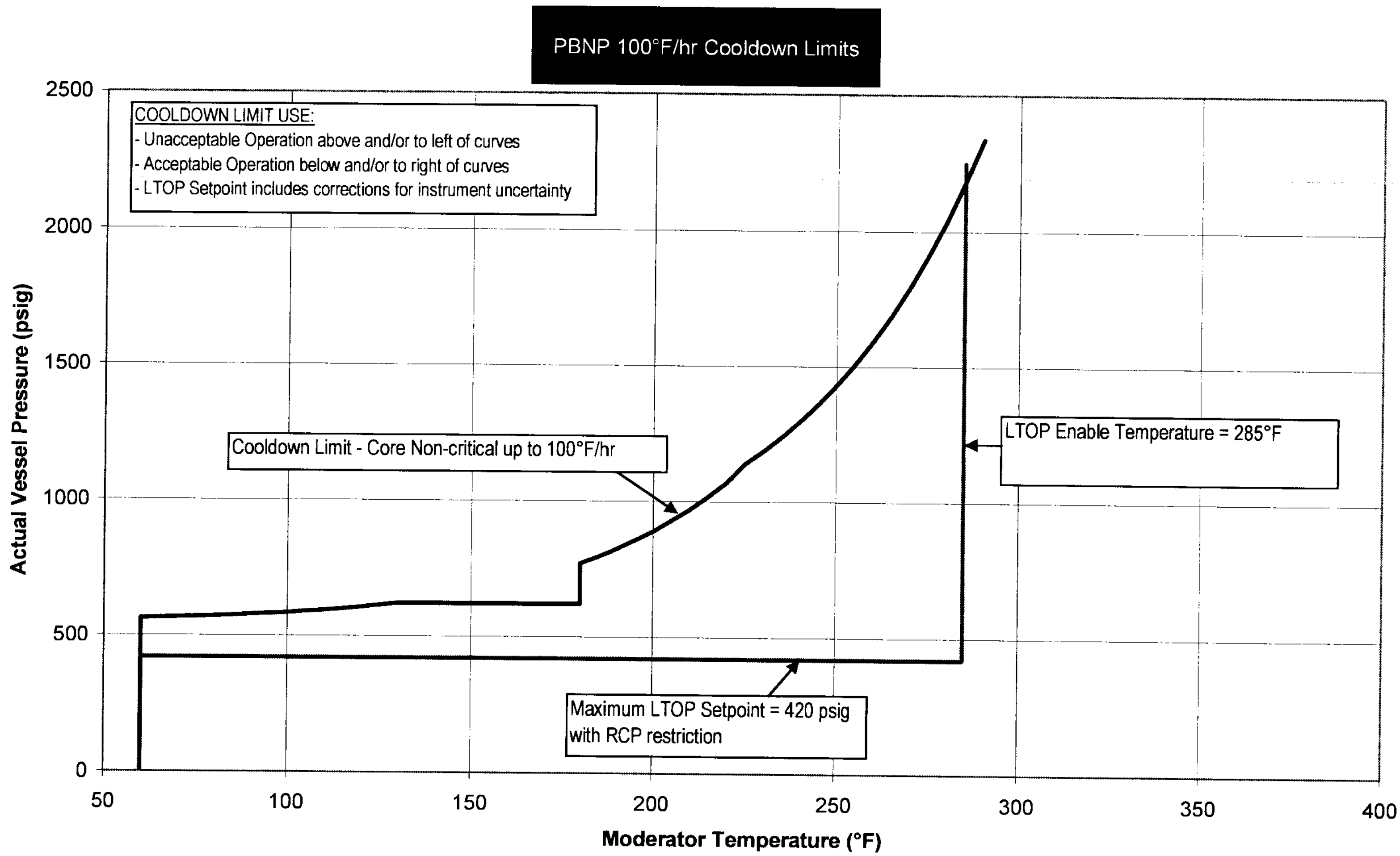

TABLE 1
POINT BEACH NUCLEAR PLANT UNIT 1
REACTOR VESSEL SURVEILLANCE CAPSULE REMOVAL SCHEDULE

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
A	April 2022**

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

** The actual removal date will be adjusted depending on the implementation of a power uprate and operating history of Unit 2. (NRC SEdated 12/2005, NUREG 1839)

TABLE 3
POINT BEACH UNIT 1 RPV BELTLINE 36.9 EFPY VALUES[(E)]

Based on Westinghouse Report LTR-REA-04-64, "Pressure Vessel Neutron Exposure Evaluations Point Beach Units 1 and 2," June 2004 (Ref 5.15).. 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.

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

Component Description	Heat or Heat/Lot	36.9 EFPY Inside Surface Fluence (E19 n/cm^2)	36.9 EFPY 1/4T Fluence (E19 n/cm^2) [(B)]	36.9 EFPY 1/4T Fluence Factor [(C)]	36.9 EFPY 3/4T Fluence (E19 n/cm^2) [(B)]	36.9 EFPY 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	0.68	N/A
Intermediate Shell Long Seam (OD 73%) [(A)]	1P0661 (SA-775)	2.19	1.48	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 36.9 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 36.9 EFPY 1/4T fluence factor for nozzle belt forging, heat no. 122P237, $ff = 0.17^{(0.28 - 0.10 \log 0.3724)} = 0.53$.

(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 Section 2.0, "Operating Limits," for discussion of EFPY values.

TABLE 4
POINT BEACH UNIT 2 RPV BELTLINE 36.9 EFPY VALUES[(E)]

Based on Westinghouse Report LTR-REA-04-64, "Pressure Vessel Neutron Exposure Evaluations Point Beach Units 1 and 2," June 2004(Ref 5.15). 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.

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

Component Description	Heat or Heat/Lot	36.9 EFPY Inside Surface Fluence (E19 n/cm^2)	36.9 EFPY 1/4T Fluence (E19 n/cm^2) [(B)]	36.9 EFPY 1/4T Fluence Factor [(C)]	36.9 EFPY 3/4T Fluence (E19 n/cm^2) [(B)]	36.9 EFPY 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 36.9 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 36.9 EFPY 1/4T fluence factor for nozzle belt forging, heat no. 123V352, $ff = 0.23^{(0.28 - 0.10 \log 0.3910)} = 0.60$.

(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 Section 2.0, "Operating Limits," for discussion of EFPY values.

TABLE 5
POINT BEACH UNIT 1 RPV 1/4T BELTLINE MATERIAL ADJUSTED REFERENCE TEMPERATURES AT 36.9 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)

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 36.9 EFPY 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}. For example, for nozzle belt forging, heat no.122P237, ART = 50 + (77 × 0.53) + 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) Deleted.
(H) EFPY value listed here is based on various reactor fuel management strategies and reactor power levels. See Section 2.0, "Operating Limits," for discussion of EFPY values.

TABLE 6
POINT BEACH UNIT 2 RPV 1/4T BELTLINE MATERIAL ADJUSTED REFERENCE TEMPERATURES AT 36.9 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).

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 36.9 EFPY 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	0.60	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 $\times$ 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 $\times$ 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 Section 2.0, "Operating Limits," for discussion of EFPY values.

TABLE 7
POINT BEACH UNIT 1 RPV 3/4T BELTLINE MATERIAL ADJUSTED REFERENCE TEMPERATURES AT 36.9 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)

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 36.9 EFPY 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 Section 2.0, "Operating Limits," for discussion of EFPY values.

TABLE 8
POINT BEACH UNIT 2 RPV 3/4T BELTLINE MATERIAL ADJUSTED REFERENCE TEMPERATURES AT 36.9 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)

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 36.9 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	0.44	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}. For example, for nozzle belt forging, heat no. 123V352, ART = 40 + (76 × 0.44) + 34 = 107°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) 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 Section 2.0, "Operating Limits," for discussion of applicability dates.

ENCLOSURE 2

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

PRESSURE TEMPERATURE LIMITS REPORT

REVISION 5, ISSUED MAY 30, 2008

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-04-64 (Ref 5.15), this PTLR is effective for 36.9 EFPY (approximately 2015). (Ref 5.2)

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 lift setting 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.

4.0 SUPPLEMENTAL DATA INFORMATION

The RT_{PTS} values for the PBNP limiting beltline materials at 36.9 EFPY is:

- Unit 1 - Intermediate to Lower Shell Circ Weld = 277°F; Lower Shell Axial Weld = 234°F (Ref. 5.8, Table 13)
- Unit 2 - Intermediate to Lower Shell Circ Weld = 292°F; Intermediate Shell Forging = 149°F (Ref. 5.8, Table 21)

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 WCAP-16274-NP, "Evaluation of Pressurized Thermal Shock for Point Beach Units 1 and 2," Revision 0, June 2004

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 dated October 18, 2007 issuing 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)

5.15 Westinghouse Report LTR-REA-04-64, "Pressure Vessel Neutron Exposure Evaluation Point Beach Units 1 and 2," dated June 2004 (Westinghouse Letter WEP-04-107)

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

Figure 1
RCS PRESSURE-TEMPERATURE LIMITS FOR HEATUP

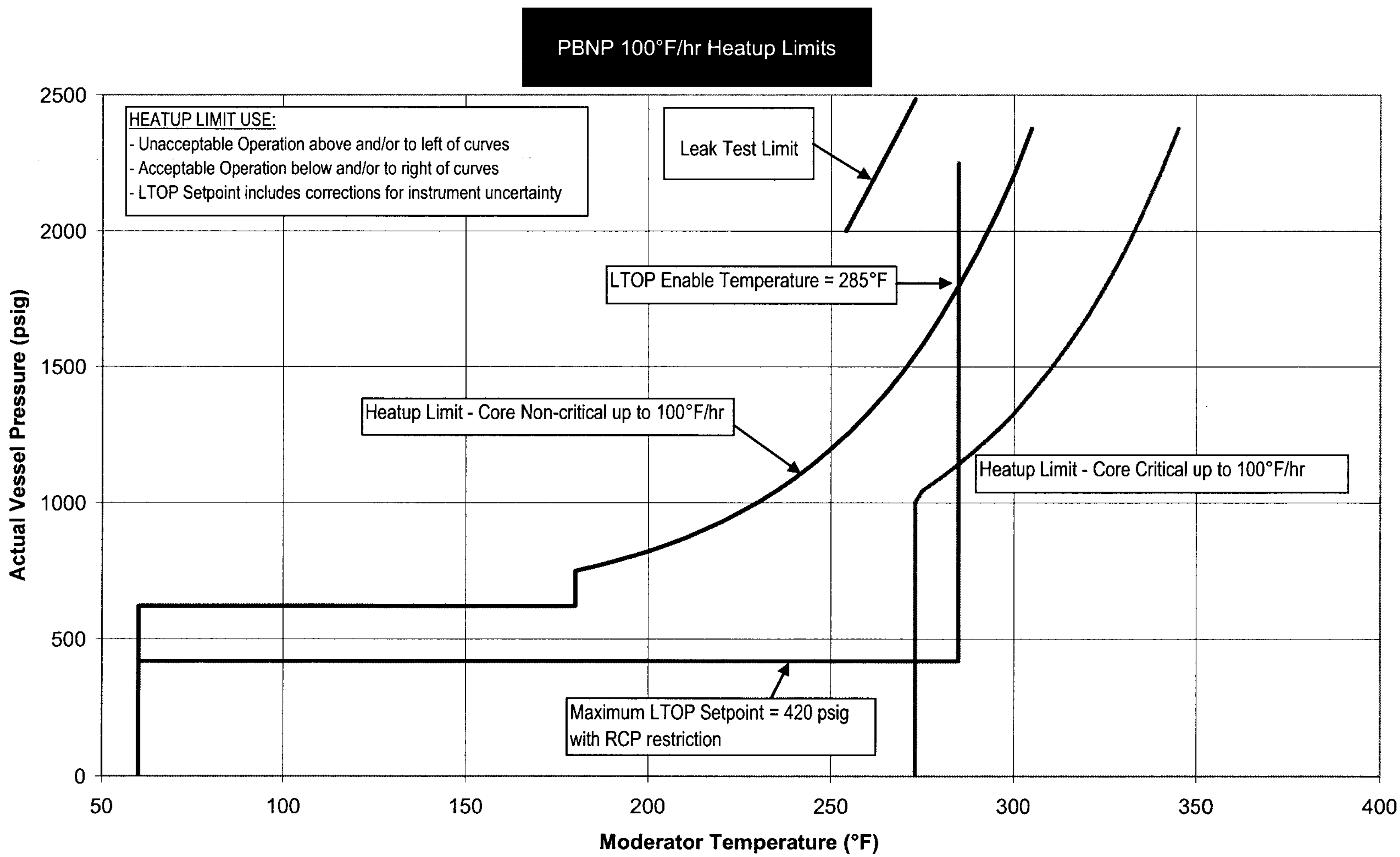

Figure 2
RCS PRESSURE-TEMPERATURE LIMITS FOR COOLDOWN

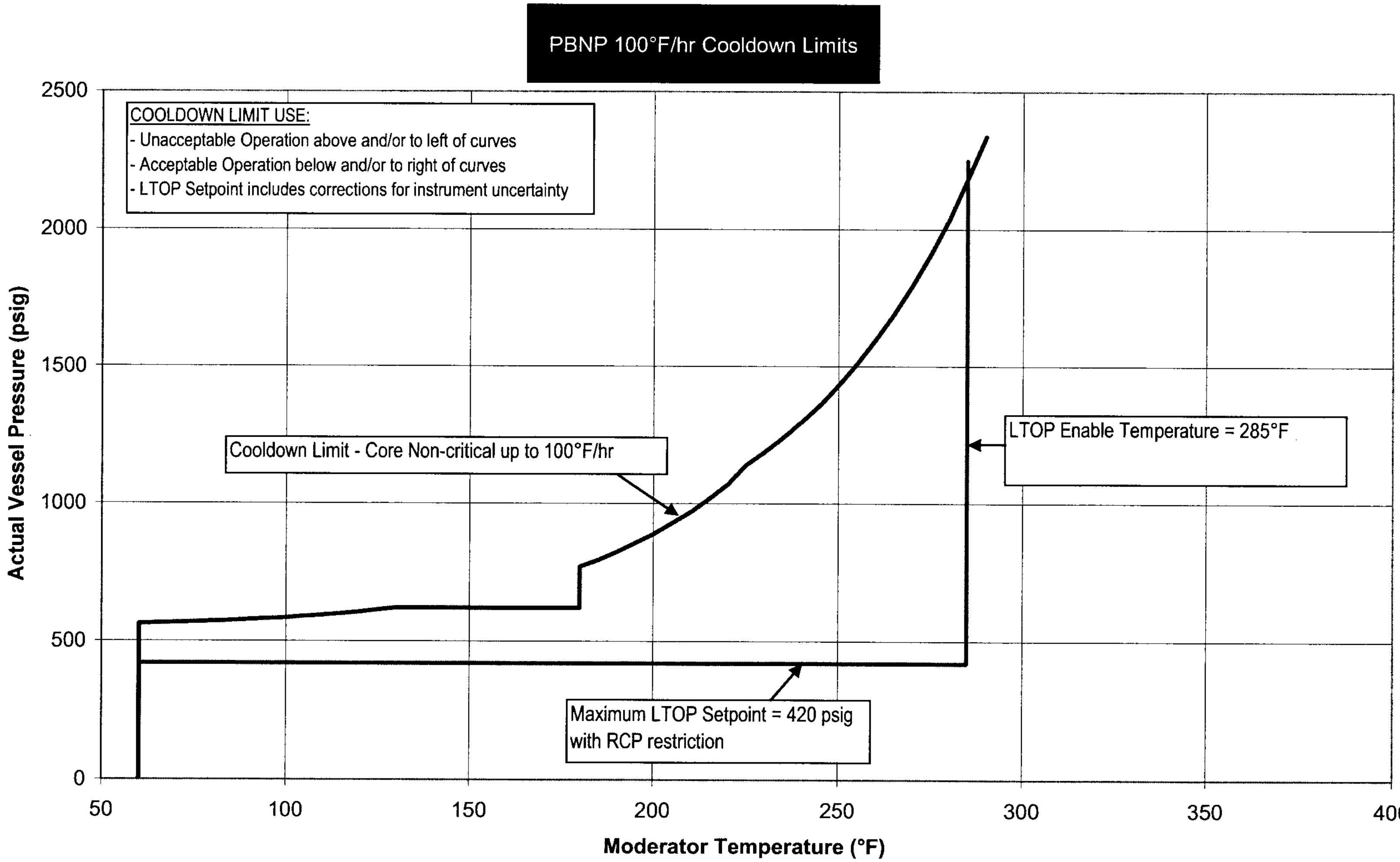

TABLE 1
POINT BEACH NUCLEAR PLANT UNIT 1
REACTOR VESSEL SURVEILLANCE CAPSULE REMOVAL SCHEDULE

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
A	April 2022**

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

** The withdraw schedule for Capsule A is shown in PWROG BAW-1543 (NP) (Ref. 5.5). The actual removal date will be adjusted depending on the implementation of a power uprate and operating history of Unit 2. (NRC SE dated 12/2005, NUREG 1839)

TABLE 3
POINT BEACH UNIT 1 RPV BELTLINE 36.9 EFPY VALUES[E]

Based on Westinghouse Report LTR-REA-04-64, "Pressure Vessel Neutron Exposure Evaluations Point Beach Units 1 and 2," June 2004 (Ref 5.15).. 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.

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

Component Description	**Heat or Heat/Lot**	**36.9 EFPY Inside Surface Fluence (E19 n/cm^2)**	**36.9 EFPY 1/4T Fluence (E19 n/cm^2) [B]**	**36.9 EFPY 1/4T Fluence Factor [C]**	**36.9 EFPY 3/4T Fluence (E19 n/cm^2) [B]**	**36.9 EFPY 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	0.68	N/A
Intermediate Shell Long Seam (OD 73%) [A]	1P0661 (SA-775)	2.19	1.48	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 36.9 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 36.9 EFPY 1/4T fluence factor for nozzle belt forging, heat no. 122P237, $ff = 0.17^{(0.28 - 0.10 \log 0.3724)} = 0.53$.

(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 (Ref .52)" for discussion of EFPY values.

TABLE 4
POINT BEACH UNIT 2 RPV BELTLINE 36.9 EFPY VALUES[(E)]

Based on Westinghouse Report LTR-REA-04-64, "Pressure Vessel Neutron Exposure Evaluations Point Beach Units 1 and 2," June 2004 (Ref 5.15). 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.

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

Component Description	Heat or Heat/Lot	36.9 EFPY Inside Surface Fluence (E19 n/cm^2)	36.9 EFPY 1/4T Fluence (E19 n/cm^2) [(B)]	36.9 EFPY 1/4T Fluence Factor [(C)]	36.9 EFPY 3/4T Fluence (E19 n/cm^2) [(B)]	36.9 EFPY 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 36.9 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 36.9 EFPY 1/4T fluence factor for nozzle belt forging, heat no. 123V352, $ff = 0.23^{(0.28 - 0.10 \log 0.3910)} = 0.60$.

(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 (Ref 5.2) for discussion of EFPY values.

TABLE 5
POINT BEACH UNIT 1 RPV 1/4T BELTLINE MATERIAL ADJUSTED REFERENCE TEMPERATURES AT 36.9 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)

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 36.9 EFPY 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}. For example, for nozzle belt forging, heat no.122P237, ART = 50 + (77 × 0.53) + 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) Deleted.
(H) EFPY value listed here is based on various reactor fuel management strategies and reactor power levels. See WCAP-15976 (Ref 5.2) for discussion of EFPY values.

TABLE 6
POINT BEACH UNIT 2 RPV 1/4T BELTLINE MATERIAL ADJUSTED REFERENCE TEMPERATURES AT 36.9 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).

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 36.9 EFPY 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	0.60	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 × 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 (Ref 5.2) for discussion of EFPY values.

TABLE 7
POINT BEACH UNIT 1 RPV 3/4T BELTLINE MATERIAL ADJUSTED REFERENCE TEMPERATURES AT 36.9 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)

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 36.9 EFPY 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 (Ref 5.2) for discussion of EFPY values.

TABLE 8
POINT BEACH UNIT 2 RPV 3/4T BELTLINE MATERIAL ADJUSTED REFERENCE TEMPERATURES AT 36.9 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)

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 36.9 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	0.44	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}. For example, for nozzle belt forging, heat no. 123V352, ART = 40 + (76 × 0.44) + 34 = 107°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) 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 (Ref 5.2) for discussion of EFPY values.