NEXTera ENERGY
POINT BEACH

January 9, 2020

NRC 2020-0001
TS 5.6.5

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

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

Pressure Temperature Limits Report (PTLR)

In accordance with the requirements of Point Beach Nuclear Plant (PBNP) Technical Specification 5.6.5, enclosed is TRM 2.2, Pressure Temperature Limits Report (PTLR).

The Enclosure to this letter contains the PBNP PTLR that was issued on December 17, 2019.

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

Sincerely,

NextEra Energy Point Beach, LLC

Eric Schultz
Licensing Manager

Enclosure

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

ENCLOSURE

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

PRESSURE TEMPERATURE LIMITS REPORT

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 (Ref 5.1), July 23, 2001 (Ref 5.2), October 18, 2007 (Ref 5.3), and June 30, 2014 (Ref 5.4).

The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto (Ref 5.16). Based upon fluence values in Westinghouse WCAP-16669-NP (Ref 5.5), this PTLR is effective for 50 EFPY (approximately through the end of year 2029).

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. (Ref 5.5)

b. A maximum cooldown rate of 100°F in any one hour. (Ref 5.5)

c. An average temperature change of ≤10°F per hour during inservice leak and hydrostatic testing operations. (Ref 5.5)

2.1.2 The RCS P-T limits for heatup and cooldown are specified by Figures 1 and 2, respectively. (Ref 5.5)

2.1.3 Using the approved methodology of ASME XI Appendix G, the minimum temperature for pressurization of the RCS is 60°F based on the most limiting of the two Point Beach reactor vessels' beltline welds.

The requirements of 10 CFR 50 Appendix G also limit operation with the reactor vessel head tensioned ("bolt-up") to ≥60°F based on the most limiting of the two Point Beach reactor vessels' flange region materials.

Correcting for possible instrument uncertainties, 60°F actual corresponds to a minimum indicated temperature of 78°F when reading the RCS loop wide range cold leg meter, or 70° when reading the A RHR heat exchanger outlet temperature (TE-627) on PPCS. Alternatively, the minimum acceptable reactor vessel flange temperature (not the reactor head flange temperature) for tensioned or pressurized operation is 70°F as indicated by hand-held digital pyrometer reading. (Ref 5.5)

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 5.6)

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

Pressurizer Power-Operated Relief Valve Lift Setting Limits

The limiting trip setpoint (Ref 5.6) 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 5.6)

2.3.2 Charging Pumps - Limit the number of operating charging pumps to two when LTOP is in service. (Ref 5.6)

2.4 <u>Criticality and Hydrostatic Leak Test Limits</u>

2.4.1 Criticality and hydrostatic leak test limits are shown on the RCS Pressure Temperature Limits for heatup, Figure 1. (Ref 5.5)

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 5.13 and 5.14)

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. 5.8)(Ref 5.15)

4.0 SUPPLEMENTAL DATA INFORMATION

The limiting RT_{PTS} values for the PBNP limiting beltline materials at 50 EFPY are:

- Unit 1 - Intermediate to Lower Shell Circ Weld = 245.3 °F; Intermediate Shell Long. Weld = 236.0 °F (Ref. 5.11 Table 1)
- Unit 2 - Intermediate to Lower Shell Circ Weld = 280.6 °F; Intermediate Shell Forging = 155.4 °F (Ref. 5.11 Table 1)

5.0 REFERENCES

5.1 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

5.2 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

5.3 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

5.4 NRC Letter, "Point Beach Nuclear Plant, Units 1 and 2 - Issuance of Amendment (Nos. 250 and 254) Regarding Change to Technical Specification 5.6.5, Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR) (TAC Nos. MF0532 and MF0533)," and NRC Letter, "Point Beach Nuclear Plant (Point Beach), Units 1 and 2 – Exemption From the Requirements of 10 CFR Section 50.61 and Appendix G to 10 CFR Part 50 (TAC Nos. MF0534 and MF0535)," dated June 30, 2014

5.5 WCAP-16669, "Point Beach Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," Revision 1, January 2009

5.6 Westinghouse calculation CN-SCS-06-68, "LTOPS Setpoint Analysis for Point Beach Units 1 and 2," 10/27/2011

5.7 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

5.8 PWR Owner Group Topical Report BAW-1543(NP), Revision 4, Supplement 7-A, "Supplement to the Master Integrated Reactor Vessel Surveillance Program" March 2018 (ML18184A520, includes NRC Final SER, ML18038A345)

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

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

5. 11 Areva Calculation 32 9019238 000, "RTPTS Values for Point Beach Unit 1 and Unit 2," Revision 0, May 2006

5. 12 Areva Calculation 32-9019240-000, "ART Values for Point Beach Unit 1 and Unit 2," Revision 0, June 2006

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

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

5.15 Deleted [NRC 2010-0004]

5.16 Point Beach Technical Specifications, LCO 5.6.5.c

5.17 WCAP-15856, Rev. 0, "Supplemental Reactor Vessel Surveillance Capsule "A" for the Point Beach Units 1 and 2 Reactor Vessel Installed in the Point Beach Unit 2 Reactor Vessel, "May 2002.

5.18 AR 2309642, "Clarification Required Regarding Implementation of LCO 3.4.3"

5.19 AR 2316053, "Revise TRM 2.2, PTLR Requirement 2.1.3"

Figure 1
RCS PRESSURE-TEMPERATURE LIMITS FOR HEATUP

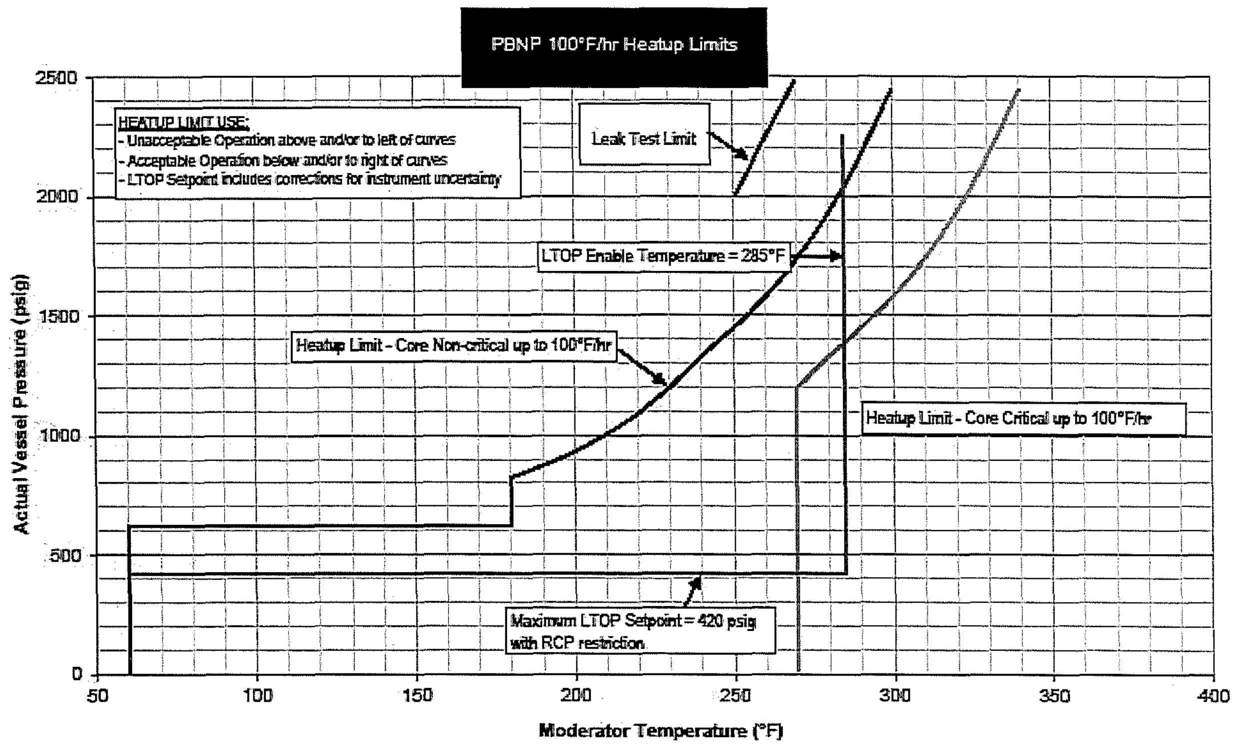

Figure 2

RCS PRESSURE-TEMPERATURE LIMITS FOR COOLDOWN

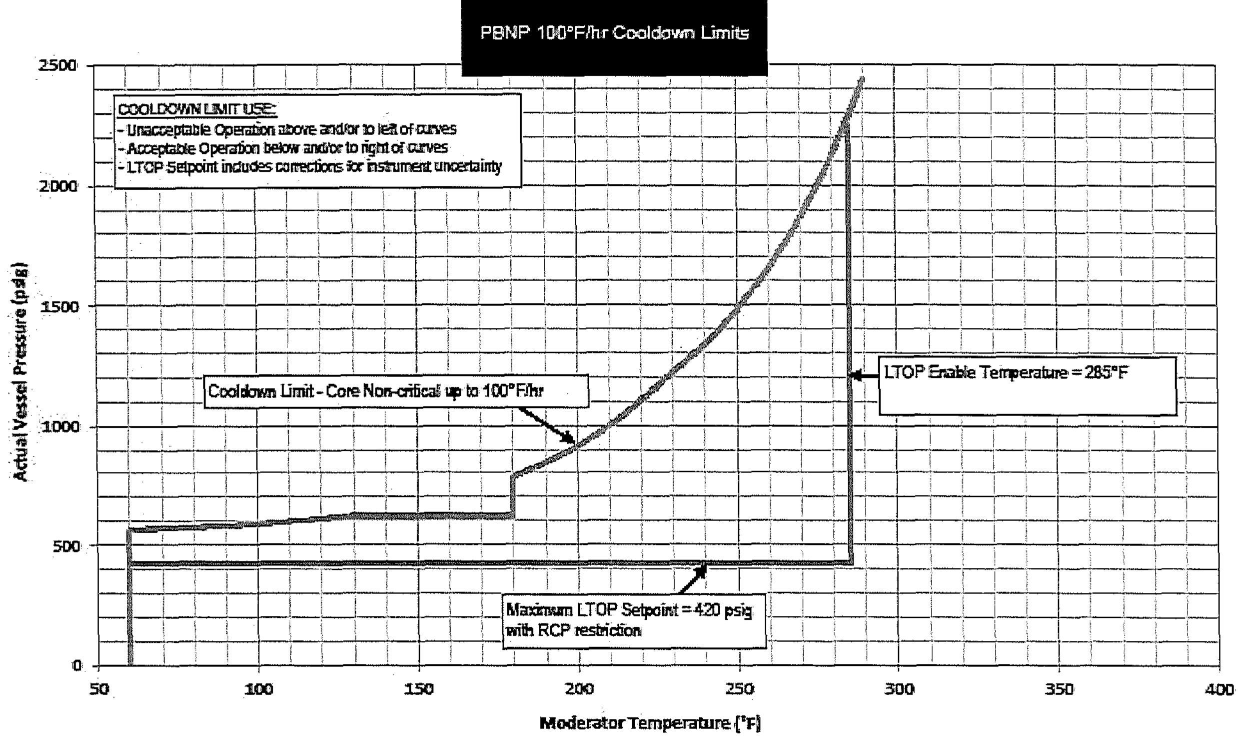

TABLE 1*
POINT BEACH NUCLEAR PLANT UNIT 1
REACTOR VESSEL SURVEILLANCE CAPSULE REMOVAL SCHEDULE

Capsule Identification Letter	Approximate Removal Date**
V	EOC-1 (Sept 1972)
S	EOC-3 (Dec 1975)
R	EOC-5 (Oct 1977)
T	EOC-11 (Mar 1984)
P	EOC-21 (Apr 1994 - Stored in SFP)
N	Standby

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

Capsule Identification Letter	Approximate Removal Date**
V	EOC-1 (Nov 1974)
T	EOC-3 (Mar 1977)
R	EOC-5 (Apr 1979)
S	EOC-16 (Oct 1990)
P	EOC-22 (Jun 1997- Stored in SFP)
N	Standby
Supplemental Capsule “A” ***	43 EFPY (~Fall 2024)

* 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. 5.8)

** For capsules that have not been withdrawn yet, the actual dates will be adjusted to coincide with the closest scheduled plant refueling outage.

*** Supplemental Capsule “A” (also identified as Supplemental Capsule “W” in some documents) was installed in Cycle 25 and is described in WCAP-15856. (Ref 5.17) The removal date is provided in effective full power years (EFPY) as agreed between the NRC and PWROG. (Ref. 5.8)

TABLE 3
POINT BEACH UNIT 1 RPV BELTLINE 50 EFPY VALUES(E)

Based on Westinghouse Report, WCAP-16669, Revision 1, "Point Beach Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," (Ref 5.5).

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

Component Description	Heat or Heat/Lot	Inside Surface Fluence (E19 n/cm^2)	1/4T Fluence (E19 n/cm^2) (B)	1/4T Fluence Factor (C)	3/4T Fluence (E19 n/cm^2) (B)	3/4T Fluence Factor (C)
Nozzle Belt Forging	122P237	0.36	0.24	0.62	0.11	0.44
Intermediate Shell Plate	A9811-1	4.90	3.32	1.31	1.52	1.12
Lower Shell Plate	C1423-1	4.55	3.09	1.30	1.41	1.10
Nozzle Belt to Intermed. Shell Circ Weld (100%)	8T1762 (SA-1426)	0.36	0.24	0.62	0.11	0.44
Intermediate Shell Long (A) Weld (ID 27%)	1P0815 (SA-812)	3.19	2.16	1.21	N/A	N/A
Intermediate Shell Long (A) Weld (OD 73%)	1P0661 (SA-775)	3.19	N/A	N/A	0.99	1.00
Intermed. to Lower Shell (A) Circ. Weld (100%)	71249 (SA-1101)	4.43	3.00	1.29	1.38	1.09
Lower Shell Long Weld (100%)	61782 (SA-847)	3.05	2.07	1.20	0.95	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.

(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.

(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.

TABLE 4
POINT BEACH UNIT 2 RPV BELTLINE 50 EFPY VALUES(E)

Based on Westinghouse Report WCAP-16669, Revision 1, "Point Beach Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," (Ref 5.5).

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

Component Description	Heat or Heat/Lot	Inside Surface Fluence (E19 n/cm^2)	1/4T Fluence (E19 n/cm^2) (B)	1/4T Fluence Factor (C)	3/4T Fluence (E19 n/cm^2) (B)	3/4T Fluence Factor (C)
Nozzle Belt Forging	123V352	0.50	0.34	0.70	0.16	0.51
Intermediate Shell Forging (A)	123V500	5.05	3.42	1.32	1.57	1.12
Lower Shell Forging	122W195	4.90	3.32	1.31	1.52	1.12
Nozzle Belt to Intermed. Shell Circ Weld (100%)	21935	0.50	0.34	0.70	0.16	0.51
Intermed. to Lower Shell Circ Weld (100%) (A)	72442 (SA-1484)	4.65	3.15	1.30	1.44	1.10

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.

(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.

(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.

TABLE 5
POINT BEACH UNIT 1 RPV 1/4T BELTLINE MATERIAL ADJUSTED REFERENCE TEMPERATURES AT 50 EFPY.(H)

Unless otherwise noted, all ART input data obtained from WCAP-16669, Revision 1, "Point Beach Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," (Ref 5.5).

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 Fluence Factor(A)	ΔRT_{NDT} (°F)	σ_I(G)	σ_Δ(G)	Margin (°F)	ART (°F)(E)
Nozzle Belt Forging	122P237	+50	0.11	0.82	77.0	Table	0.62	47.4	0	17.0	34.0	131.4
Intermediate Shell Plate	A9811-1	+1	0.20	0.06	79.3	Surv. Data(B)	1.31	104.1	26.9	8.5	56.4	161.5
Lower Shell Plate	C1423-1	+1	0.12	0.07	35.8	Surv. Data(B)	1.30	46.4	26.9	8.5	56.4	103.8
Nozzle Belt to Intermed. Shell Circ Weld (100%)	8T1762 (SA-1426)	-47.6	0.19	0.57	167.0	Table	0.62	102.9	17.2	28	65.7	121.0
Intermediate Shell Long Weld (ID 27%)	1P0815 (SA-812)	-47.6	0.17	0.52	167.0	Table	1.21	201.9	17.2	28	65.7	220.0
Intermediate Shell Long Weld (OD 73%)	1P0661 (SA-775)	-47.6	0.17	0.64	167.0	Table	N/A	N/A	17.2	28	N/A	N/A
Intermed. to Lower Shell Circ. Weld (100%)	71249 (SA-1101)	-47.4	0.23	0.59	167.6	Table(C)	1.29	216.4	12.9	28	61.7	230.7
Lower Shell Long Weld (100%)	61782 (SA-847)	-47.6	0.23	0.52	167.0	Table	1.20	199.9	17.2	28	65.7	218.1

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.5.7) 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}.
(F) Instruction Manual, 132-Inch I.D. Reactor Pressure Vessel, Babcock & Wilcox, September 1969.
(G) σ_I and σ_Δ values obtained from Areva Calculation 32-9019240-000, Table 41 (Ref. 5.12)
(H) EFPY value listed here is based on various reactor fuel management strategies and reactor power levels.

TABLE 6
POINT BEACH UNIT 2 RPV 1/4T BELTLINE MATERIAL ADJUSTED REFERENCE TEMPERATURES AT 50 EFPY.[(I)]

Unless otherwise noted, all ART input data obtained from WCAP-16669, Revision 1, "Point Beach Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," (Ref 5.5).

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-Fluence Factor[(A)]	ΔRT_{NDT} (°F)	σ_I[(G)]	σ_Δ[(G)]	Margin (°F)	ART (°F)[(E)]
Nozzle Belt Forging	123V352	+40	0.11	0.73	76.0	Table	0.70	53.5	0	17.0	34.0	127.5
Intermediate Shell Forging	123V500	+40	0.09	0.70	58.0	Table[(B)]	1.32	76.6	0	17.0	34.0	150.6
Lower Shell Forging	122W195	+40	0.05	0.72	42.8	Surv. Data[(C)]	1.31	56.2	0	8.5	17.0	113.2
Nozzle Belt to Intermed. Shell Circ Weld (100%)	21935	-56	0.18	0.70	170.5	Table[(H)]	0.70	120.0	17.0	28	65.5	129.5
Intermed. to Lower Shell Circ. Weld (100%)	72442 (SA-1484)	-30	0.26	0.60	180.0	Table[(D)]	1.30	234.4	11.9	28	60.8	265.2

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}.
(F) Instruction Manual, Reactor Vessel, Point Beach Nuclear Plant Unit 2, Combustion Engineering, CE Book #4869, October 1970.
(G) σ_I and σ_Δ values obtained from Areva Calculation 32-9019240-000, Table 41 (Ref. 5.12)
(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.5.10).
(I) EFPY value listed here is based on various reactor fuel management strategies and reactor power levels.

TABLE 7
POINT BEACH UNIT 1 RPV 3/4T BELTLINE MATERIAL ADJUSTED REFERENCE TEMPERATURES AT 50 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. 5.9) and WCAP-16669, Revision 1, "Point Beach Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," (Ref 5.5).

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-Fluence Factor(A)	ΔRT_{NDT} (°F)	σ_I(G)	σ_Δ(G)	Margin (°F)	ART (°F)(E)
Nozzle Belt Forging	122P237	+50	0.11	0.82	77.0	Table	0.44	33.7	0	17.0	33.7	117.4
Intermediate Shell Plate	A9811-1	+1	0.20	0.06	79.3	Surv. Data(B)	1.12	88.5	26.9	8.5	56.4	145.9
Lower Shell Plate	C1423-1	+1	0.12	0.07	35.8	Surv. Data(B)	1.10	39.2	26.9	8.5	56.4	96.6
Nozzle Belt to Intermed. Shell Circ Weld (100%)	8T1762 (SA-1426)	-47.6	0.19	0.57	167.0	Table	0.44	73.2	17.2	28	65.7	91.3
Intermediate Shell Long Weld (ID 27%)	1P0815 (SA-812)	-47.6	0.17	0.52	167.0	Table	N/A	N/A	17.2	28	N/A	N/A
Intermediate Shell Long Weld (OD 73%)	1P0661 (SA-775)	-47.6	0.17	0.64	167.0	Table	1.00	166.5	17.2	28	65.7	184.6
Intermed. To Lower Shell Circ. Weld (100%)	71249 (SA-1101)	-47.4	0.23	0.59	167.6	Table(C)	1.09	182.3	12.9	28	61.7	196.6
Lower Shell Long Weld (100%)	61782 (SA-847)	-47.6	0.23	0.52	167.0	Table	0.99	164.5	17.2	28	65.7	182.7

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}.
(F) Instruction Manual, 132-Inch I.D. Reactor Pressure Vessel, Babcock & Wilcox, September 1969.
(G) σ_I and σ_Δ values obtained from Areva Calculation 32-9019240-000, Table 41 (Ref. 5.12)
(H) EFPY value listed here is based on various reactor fuel management strategies and reactor power levels.

TABLE 8
POINT BEACH UNIT 2 RPV 3/4T BELTLINE MATERIAL ADJUSTED REFERENCE TEMPERATURES AT 50 EFPY *(I)*

Unless otherwise noted, all ART input data obtained from WCAP-16669, Revision 1, "Point Beach Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," (Ref 5.5).

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*(I)* Fluence Factor*(A)*	ΔRT_{NDT} (°F)	σ_I*(G)*	σ_Δ*(G)*	Margin (°F)	ART (°F)*(E)*
Nozzle Belt Forging	123V352	+40	0.11	0.73	76.0	Table	0.51	38.9	0	17.0	34.0	112.9
Intermediate Shell Forging	123V500	+40	0.09	0.70	58.0	Table*(B)*	1.12	65.2	0	17.0	34.0	139.2
Lower Shell Forging	122W195	+40	0.05	0.72	42.8	Surv. Data*(C)*	1.12	47.8	0	8.5	17.0	104.8
Nozzle Belt to Intermed. Shell Circ Weld (100%)	21935	-56	0.18	0.70	170.5	Table*(H)*	0.51	87.3	17.0	28	65.5	96.8
Intermed. to Lower Shell Circ. Weld (100%)	72442 (SA-1484)	-30	0.26	0.60	180.0	Table *(D)*	1.10	198.4	11.9	28	60.8	229.2

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}.
(F) Instruction Manual, Reactor Vessel, Point Beach Nuclear Plant No. 2, Combustion Engineering, CE Book #4869, October 1970.
(G) σ_I and σ_Δ values obtained from Areva Calculation 32-9019240-000, Table 41 (Ref. 5.12)
(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.