

NRC 2001-0019

10 CFR 50.90

April 10, 2001

Document Control Desk  
U.S. Nuclear Regulatory Commission  
Mail Stop P1-137  
Washington, DC 20555

Ladies and Gentlemen:

DOCKETS 50-266 AND 50-301  
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION;  
SUPPLEMENT 2 TO TECHNICAL SPECIFICATIONS CHANGE REQUEST 219  
ADOPTION OF PRESSURE AND TEMPERATURE LIMITS REPORT  
AND REVISED P-T AND LTOP LIMITS  
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. MA8459 AND MA8460)

By submittal dated March 10, 2000, Wisconsin Electric Power Company (then Licensee) requested amendments to Facility Operating Licenses DPR-24 and DPR-27 for Point Beach Nuclear Power Plant, Units 1 and 2, respectively, to incorporate changes to the plant Operating Licenses and Technical Specifications. The purpose of the proposed amendments was to implement a Pressure and Temperature Limits Report (PTLR) concurrent with implementation of Improved Standard Technical Specifications at the Point Beach Nuclear Plant (PBNP).

An application to convert the PBNP custom Technical Specifications to Standard Technical Specifications based on NUREG-1431, "Standard Technical Specifications - Westinghouse Plants," Revision 1, was submitted on November 15, 1999. That submittal was based on the incorporation of a PTLR and appropriate references in the proposed Standard Technical Specifications. By letters dated June 21 and September 21, 2000, the staff requested additional information related to the PTLR. Our responses were provided by letters dated July 28 and November 20, 2000 respectively.

By letter dated March 12, 2001, the staff requested followup information related to the PTLR concerning the analysis methods and supporting calculations. Attachment 1 contains our response to the staff's followup request. Attachment 2 contains Revision 2 of the proposed PTLR for Point Beach. Revision 2 of the proposed PTLR replaces the originally submitted PTLR in its entirety. The Pressure-Temperature limits and Low Temperature Overpressure Protection system limits contained within this submittal are based on the analyses provided in support of this amendment request. The changes contained in Revision 2 of the proposed PTLR consist primarily of formatting changes and corrections to references.

ADD

April 10, 2001

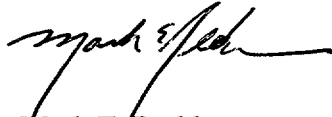
Page 2

Our evaluation concluded that the revisions proposed in this supplement are bounded by the no significant hazards consideration that was provided in our original PTLR submittal dated March 10, 2000.

We have determined that this additional information for the proposed amendments does not involve a significant hazards consideration, authorize a significant change in the types or total amounts of effluent release, or result in any significant increase in individual or cumulative occupational radiation exposure. Therefore, we conclude that the proposed amendments meet the categorical exclusion requirements of 10 CFR 51.22(c)(9) and that an environmental impact appraisal need not be prepared.

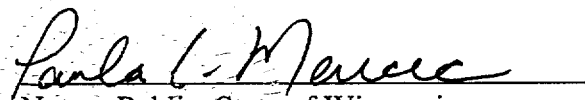
We request that these amendments be reviewed and approved such that the PTLR may be implemented with the improved TS at PBNP.

Sincerely,



Mark E. Reddemann  
Site Vice President

Subscribed and sworn before me on  
this 10th day of April, 2001.

  
Notary Public, State of Wisconsin

My commission expires October 21, 2001

JG/jlk

Attachments

cc: NRC Regional Administrator  
NRC Resident Inspector

NRC Project Manager  
PSCW

bcc:

R. G. Mende

A. J. Cayia

M. E. Reddemann

R. P. Pulec

J. Gadzala

M. D. Wadley

T. J. Webb

D. F. Johnson

R. R. Grigg

B. J. Onesti (OSRC)

J. L. Kudick (3)

D. Weaver

File

DOCKETS 50-266 AND 50-301

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION;

SUPPLEMENT 2 TO TECHNICAL SPECIFICATIONS CHANGE REQUEST 219

ADOPTION OF PRESSURE TEMPERATURE LIMITS REPORT

AND REVISED P-T AND LTOP LIMITS

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

An application to convert the PBNP Technical Specifications, to the improved Standard Technical Specifications based on NUREG-1431, "Standard Technical Specifications - Westinghouse Plants," Revision 1, was submitted November 15, 1999. That submittal is based on the incorporation of a PTLR and associated references in the proposed Technical Specifications. The proposed amendments to implement a PTLR were submitted March 10, 2000.

By letter dated March 12, 2001, the staff requested followup information related to the PTLR concerning the analysis methods and supporting calculations. The following information is provided in response to the NRC staff's request.

The staff's questions are restated below with our response following.

1. Since Point Beach Nuclear Plant (PBNP) relies on the pressurizer power-operated relief valve (PORV) for low-temperature overpressure protection (LTOP), clarify the statement (on p. 19 of Calc. No. 2000-0001, Rev. 0 (Attachment 5 of the licensee's submittal dated March 10, 2000)) that a setpoint of 500 psig is evaluated, which is the setpoint of the RHR high capacity relief valve, RH-861C.

Response – For certain overpressure transients, RHR relief valve RH-861C can provide backup overpressure protection for the reactor vessel. However, no credit is taken in the safety analysis for the relief capability of this valve. Although not credited for overpressure mitigation, WE correspondence to the NRC discusses the RHR high capacity relief valve (RH-861C) as being diverse and redundant to the PORVs in LTOP. Therefore, the 500 psig setpoint of RH-861C was evaluated in the PTLR for completeness. The PORVs provide the required overpressure protection and the analysis supports an allowable setpoint of up to 500 psig (as stated in the PTLR), independent of the RHR relief valve.

2. The equation (p. 20 of Calc 2000-0001, Rev. 0) used in the determination of the PORV setpoint overshoot includes an "EXP Ratio" term, which is calculated to be 0.74 (p. 21).

- (A) The equation is inconsistent with the simplified equation described in Section 4.2 of Westinghouse report, "Pressure Mitigating Systems Transient Analysis Results," July 1977. The Westinghouse simplified equation does not have the EXP Ratio term.

Provide the basis for including the EXP Ratio term in the equation, and describe the effects of including this term on other terms in the equation, as they were developed, without considering the metal expansion effect.

- (B) Figure 5.2 in the Westinghouse report dated July 1997 was developed to show the conservatism in the simplified equation for not considering the effect of metal expansion in the setpoint overshoot calculation. The pressure overshoot reduction calculation was based on the elastic effects on volume changes of the simple geometric shape of cylinders and hemispheres subject to a pressure change of 1000 psi (Table 5.2 in the July 1977 Westinghouse report). Figure 5.2 was based on a relief valve setpoint of 600 psig, which is higher than the PBNP LTOP PORV setpoint of 500 psig.

Provide your justification for the use of Figure 5.2 in the PBNP license amendment application.

Response – Wisconsin Electric letter to the NRC dated July 28, 1977, justified use of the 0.74 expansion term in the pressure overshoot calculation. This letter provided the final details of the proposed pressure mitigating system for PBNP. The NRC staff had previously found this method of analysis to be acceptable, as documented in the NRC Safety Evaluation for License Amendments 45 and 50 (for Units 1 and 2 respectively), dated May 20, 1980.

3. Though stated in the submittals and listed in the proposed PTLR as the method used for the P-T limits determination, WCAP-14040-NP-A is not referenced or mentioned in Calculation 2001-0001-00, "RCS P-T limits and LTOP Setpoint Applicable Thru 32.2 EFPY - Unit 1 and 34.0 - Unit 2."

Are the P-T limits and LTOP setpoint actually calculated based on the methods of WCAP-14040-NP-A? Are the calculations consistent with the WCAP-14040 methodology?

Response – As stated in WE letter to NRC, dated March 10, 2000 (NPL 2000-0123), the associated PBNP calculation accounts for all factors described in the approved methodology document, WCAP-14040.

WCAP-14040-NP-A does not contain the details of the methodologies that are required in the development of heatup and cooldown curves and LTOP setpoints. Rather, it is a reference compilation of the specific methodologies that are used to develop the PTLR. WE calculation 2000-0001 utilized (and referenced) these specific methodologies that are compiled within WCAP-14040-NP-A.

4. The Safety Evaluation (Attachment 2 to the March 10, 2000, amendment request) states that, for PBNP, Units 1 and 2, a single setpoint is established based on the most limiting reactor vessel beltline materials at the minimum allowable pressure for the RCS pressurization established pursuant to Appendix G to 10 CFR Part 50.

Clarify the above statement with “minimum” allowable pressure, which is inconsistent with the “maximum” allowable pressure in PBNP Calculation 2000-0001.

Response – The intent of this statement is to communicate the following information:

A single set point is established for both PBNP Unit 1 and 2. The setpoint is derived using the most limiting reactor vessel beltline material.

The value of maximum calculated pressure from an LTOP transient is smaller than the allowable value [i.e., lowest (or “minimum”) point] on the cooldown limit curve.

Appendix G to 10 CFR 50 is satisfied since the maximum calculated pressure from a LTOP transient is lower than the pressure limit on the limit curve.

In summary, the term “minimum”(minimum allowable pressure), as it was used in the March 10, 2000 safety evaluation, referred to the minimum point of the PBNP cooldown limits curve (where the curve asymptotically approaches 500 psig). Since this curve defines the maximum allowable pressure for operation, the intent of the statement is consistent with the “maximum” allowable pressure in PBNP Calculation 2000-0001.

NRC 2001-0019

Attachment 2

Page 1

DOCKETS 50-266 AND 50-301

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION;

SUPPLEMENT 2 TO TECHNICAL SPECIFICATIONS CHANGE REQUEST 219

ADOPTION OF PRESSURE TEMPERATURE LIMITS REPORT

AND REVISED P-T AND LTOP LIMITS

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

PROPOSED PRESSURE TEMPERATURE LIMITS REPORT  
REVISION 2

(EFFECTIVE THROUGH: 25.59 EFPY – UNIT 1; AND 30.51 EFPY – UNIT 2)

# TRM 2.2

## PRESSURE TEMPERATURE LIMITS REPORT

### UNIT 1 AND UNIT 2

#### REVISION 2 DRAFT

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

**Note:** Rev. 2 of this document incorporates all changes of PTLR Rev. 1 submitted to the NRC via NPL 2000-0510 and corrections to ITS references in section 1.0, 2.0, 1.2, 2.2, and 2.3.



PRESSURE TEMPERATURE LIMITS REPORT

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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 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 per hour.
- b. A maximum cooldown rate of 100°F per hour.
- c. An average temperature change of  $\leq 10^\circ\text{F}$  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, 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.

PRESSURE TEMPERATURE LIMITS REPORT

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2.2 Low Temperature Overpressure Protection System Enable Temperature (LCO 3.4.12)

2.2.1 The enable temperature for the Low Temperature Overpressure Protection System is 270°F.

2.3 Low Temperature Overpressure Protection System Setpoints (LCO 3.4.12)

2.3.1 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.

PRESSURE TEMPERATURE LIMITS REPORT

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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

PRESSURE TEMPERATURE LIMITS REPORT

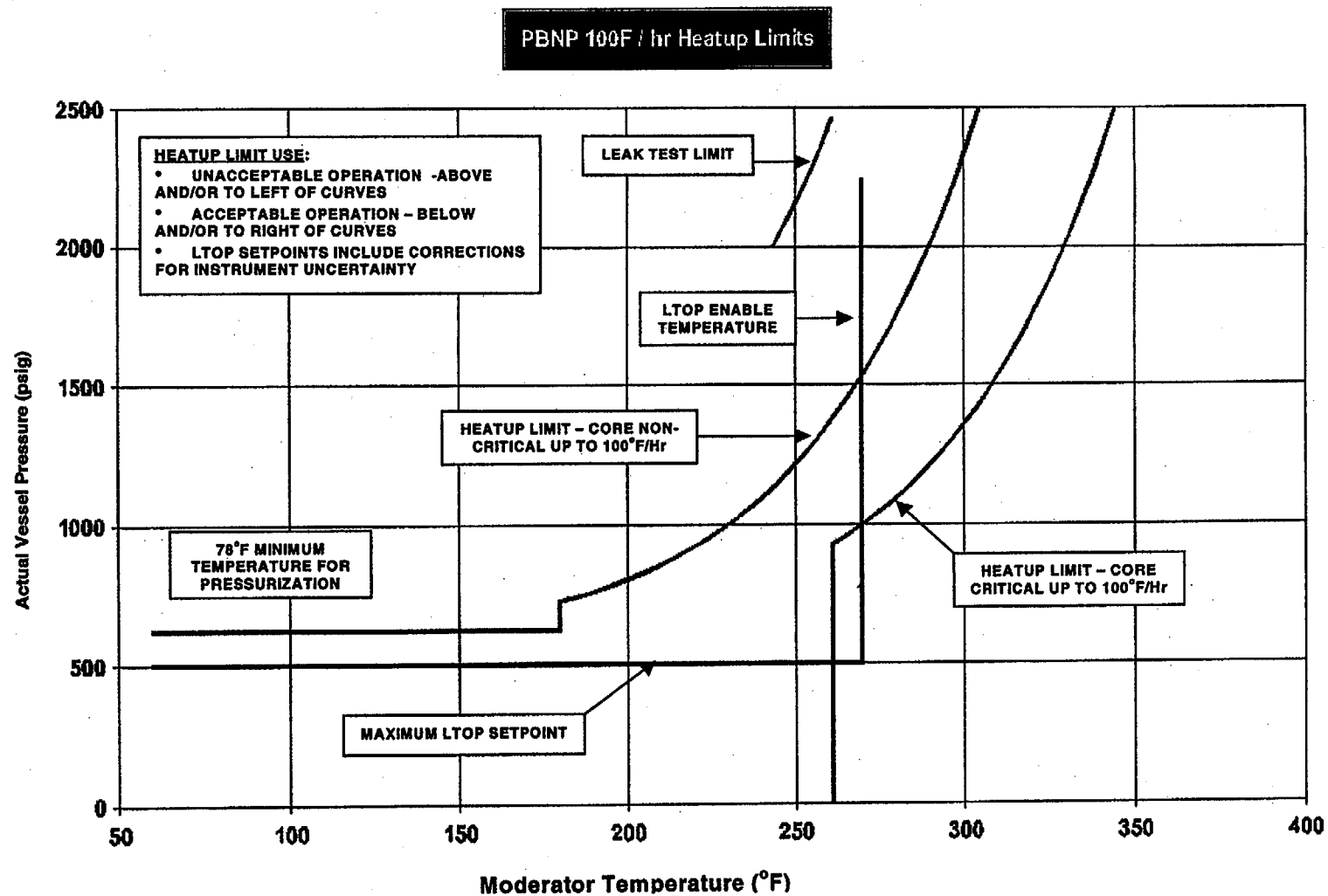
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5.0 REFERENCES

- 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

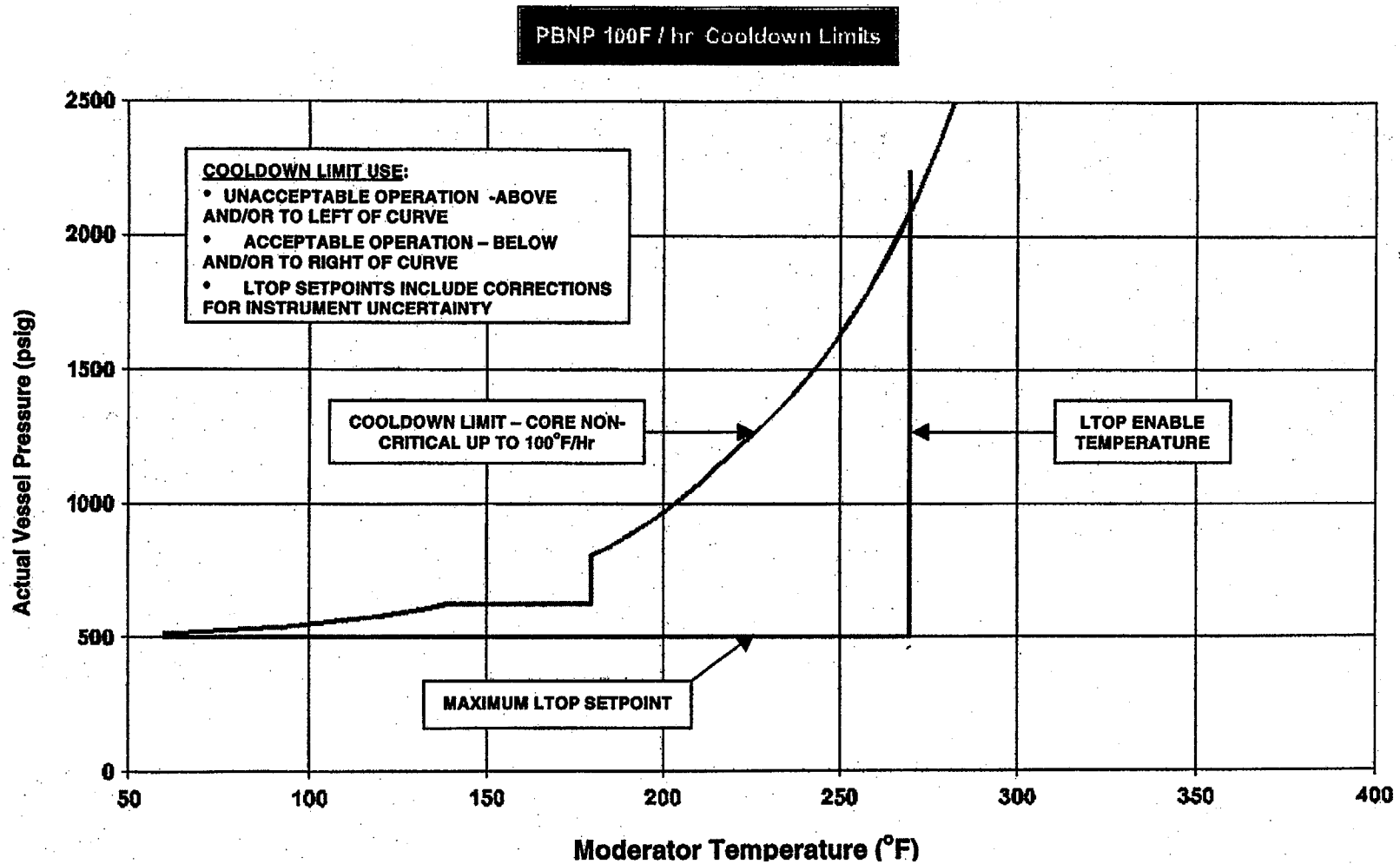
PRESSURE TEMPERATURE LIMITS REPORT

FIGURE 1  
RCS PRESSURE-TEMPERATURE LIMITS FOR HEATUP



PRESSURE TEMPERATURE LIMITS REPORT

FIGURE 2  
RCS PRESSURE-TEMPERATURE LIMITS FOR COOLDOWN



PRESSURE TEMPERATURE LIMITS REPORT

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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.

PRESSURE TEMPERATURE LIMITS REPORT

TABLE 3  
POINT BEACH UNIT 1 RPV BELTLINE 25.59 EFPY  $\phi_{Calc}$  (32.2 EFPY  $\phi_{Best.Est.}$ ) VALUES<sup>(E)</sup>

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 <sup>(D)</sup>

Component Description	Heat or Heat/Lot	32 EFPY $\phi_{Best.Est.}$ Inside Surface Fluence (E19 n/cm <sup>2</sup> )	32.2 EFPY $\phi_{Best.Est.}$ Inside Surface Fluence (E19 n/cm <sup>2</sup> ) <sup>(A)</sup>	32.2 EFPY $\phi_{Best.Est.}$ 1/4T Fluence (E19 n/cm <sup>2</sup> ) <sup>(B)</sup>	32.2 EFPY $\phi_{Best.Est.}$ 1/4T Fluence Factor <sup>(C)</sup>	32.2 EFPY $\phi_{Best.Est.}$ 3/4T Fluence (E19 n/cm <sup>2</sup> ) <sup>(B)</sup>	32.2 EFPY $\phi_{Best.Est.}$ 3/4T Fluence Factor <sup>(C)</sup>
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:

- <sup>(A)</sup> Interpolation of neutron exposure (in units of E19 n/cm<sup>2</sup>, 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,  

$$\text{fluence} = 0.547 + \left( \frac{0.796 - 0.547}{8 \text{ EFPY} - 32 \text{ EFPY}} \right) \times (32.2 \text{ EFPY} - 32.0 \text{ EFPY}) = 0.550 \text{ E19 n/cm}^2$$
- <sup>(B)</sup> 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<sup>2</sup>, 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"),  $f = 0.550 \times e^{-0.24(1.625)} = 0.3724 \text{ E19 n/cm}^2$ .
- <sup>(C)</sup> 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<sup>2</sup>. For example, the 32.2 EFPY 1/4T fluence factor for nozzle belt forging, heat no. 122P237,  $ff = 0.3724^{(0.28 - 0.10 \log 0.3724)} = 0.7269$ .
- <sup>(D)</sup> Instruction Manual, 132-Inch I.D. Reactor Pressure Vessel, Babcock & Wilcox, September 1969 (Ref. 12).
- <sup>(E)</sup> 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  $K = \phi_{Best.Est.} E \sigma_{\gamma} / \phi_{Calc} = 0.838$ .



PRESSURE TEMPERATURE LIMITS REPORT

TABLE 4  
POINT BEACH UNIT 2 RPV BELTLINE 30.51 EFPY  $\phi_{\text{Calc}}$  (34.0 EFPY  $\phi_{\text{Best.Est.}}$ ) VALUES<sup>(E)</sup>

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 <sup>(D)</sup>

Component Description	Heat or Heat/Lot	32 EFPY $\phi_{\text{Best.Est.}}$ Inside Surface Fluence (E19 n/cm <sup>2</sup> )	34.0 EFPY $\phi_{\text{Best.Est.}}$ Inside Surface Fluence (E19 n/cm <sup>2</sup> ) <sup>(A)</sup>	34.0 EFPY $\phi_{\text{Best.Est.}}$ 1/4T Fluence (E19 n/cm <sup>2</sup> ) <sup>(B)</sup>	34.0 EFPY $\phi_{\text{Best.Est.}}$ 1/4T Fluence Factor <sup>(C)</sup>	34.0 EFPY $\phi_{\text{Best.Est.}}$ 3/4T Fluence (E19 n/cm <sup>2</sup> ) <sup>(B)</sup>	34.0 EFPY $\phi_{\text{Best.Est.}}$ 3/4T Fluence Factor <sup>(C)</sup>
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:

<sup>(A)</sup> Interpolation of neutron exposure (in units of E19 n/cm<sup>2</sup>, 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,

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

<sup>(B)</sup> 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:  $f = f_{\text{surf}} \times e^{-0.24x}$ , where  $f_{\text{surf}}$  is expressed in units of E19 n/cm<sup>2</sup>, 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"),  $f = 0.5775 \times e^{-0.24(1.625)} = 0.3910 \text{ E19 n/cm}^2$ .

<sup>(C)</sup> 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<sup>2</sup>. For example, the 34.0 EFPY 1/4T fluence factor for nozzle belt forging, heat no. 123V352,  $ff = 0.3910^{(0.28 - 0.10 \log 0.3910)} = 0.7399$ .

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

<sup>(E)</sup> 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  $\phi_{\text{Best.Est.}}$  based upon  $K = \phi_{\text{Best.Est.}} \cdot E \cdot \sigma_{\text{t}} / \phi_{\text{Calc}} = 0.921$ .

PRESSURE TEMPERATURE LIMITS REPORT

TABLE 5  
POINT BEACH UNIT 1 RPV 1/4T BELTLINE MATERIAL ADJUSTED REFERENCE TEMPERATURES AT 25.59 EFPY  $\phi_{Calc}$   
(32.2 EFPY  $\phi_{Best.Est.}$ )<sup>(H)</sup>

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 <sup>(F)</sup>

Component Description	Heat or Heat/Lot	Initial RT <sub>NDT</sub> (°F)	%Cu	%Ni	CF	CF Method	1/4T 32.2 EFPY $\phi_{Best.Est.}$ Fluence Factor <sup>(A)</sup>	$\Delta RT_{NDT}$ (°F)	$\sigma_I$	$\sigma_{\Delta}$	Margin (°F)	ART (°F) <sup>(E)</sup>
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 <sup>(B)</sup>	"	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 <sup>(B)</sup>	"	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 <sup>(C)</sup>	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 <sup>(D)</sup>	"	165.42	"	14	48.34	209 <sup>(G)</sup>

Footnotes:

- <sup>(A)</sup> See Table 1.
- <sup>(B)</sup> Credible Surveillance Data; see BAW-2325 for evaluation.
- <sup>(C)</sup> Non-credible surveillance data; see BAW-2325 for evaluation. Table CF conservative because difference between ratio-adjusted measure  $\Delta RT_{NDT}$  and predicted  $\Delta RT_{NDT}$  based on Table CF is less than  $2\sigma$  (56°F).
- <sup>(D)</sup> 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.
- <sup>(E)</sup> Adjusted reference temperature (ART) calculated per Regulatory Guide 1.99, Rev. 2.  $ART = Initial RT_{NDT} + \Delta RT_{NDT} + Margin$ , where  $\Delta RT_{NDT} = Chemistry Factor \times Fluence Factor$ , and  $Margin = 2(\sigma_I^2 + \sigma_{\Delta}^2)^{0.5}$ , with  $\sigma_I$  defined as the standard deviation of the Initial RT<sub>NDT</sub> and  $\sigma_{\Delta}$  defined as the standard deviation of  $\Delta RT_{NDT}$ . For example, for nozzle belt forging, heat no.122P237,  $ART = 50 + (77 \times 0.7269) + 34 = 140^\circ F$ . Calculated ART values are rounded to the nearest °F in accordance with the rounding-off method of ASTM Practice E29.
- <sup>(F)</sup> Instruction Manual, 132-Inch I.D. Reactor Pressure Vessel, Babcock & Wilcox, September 1969.
- <sup>(G)</sup> By inspection, these are the limiting material properties.
- <sup>(H)</sup> 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  $K = \phi_{Best.Est.} / \phi_{Calc} = 0.838$ .

**PRESSURE TEMPERATURE LIMITS REPORT**

**TABLE 6**  
**POINT BEACH UNIT 2 RPV 1/4T BELTLINE MATERIAL ADJUSTED REFERENCE TEMPERATURES AT 30.51 EFPY  $\phi_{Calc}$**   
**(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 <sup>(F)</sup>

Component Description	Heat or Heat/Lot	Initial RT <sub>NDT</sub> (°F)	%Cu	%Ni	CF	CF Method	1/4T 34.0 EFPY $\phi_{Best.Est.}^{(A)}$ Fluence Factor	$\Delta RT_{NDT}$ (°F)	$\sigma_1$	$\sigma_\Delta$	Margin (°F)	ART (°F) <sup>(E)</sup>
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 <sup>(B)</sup>	1.208	70.06	0	17	34	144 <sup>(G)</sup>
Lower Shell Forging	122W195	+40	0.05	0.72	31	Table	1.161	35.99	0	17	34	110
"	"	"			42.8	Surv. Data <sup>(C)</sup>	"	49.69	"	8.5	17	107
Nozzle Belt to Intermed. Shell Circ Weld (100%)	21935	-56	0.18	0.70	170	Table <sup>(H)</sup>	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 <sup>(D)</sup>	1.156	208.08	19.7	28	68.47	272 <sup>(G)</sup>

**Footnotes:**

<sup>(A)</sup> See Table 2.

<sup>(B)</sup> Non-credible surveillance data; see BAW-2325 for evaluation. Table CF conservative because difference between measured  $\Delta RT_{NDT}$  and predicted  $\Delta RT_{NDT}$  based on Table CF is less than  $2\sigma$  (34°F)

<sup>(C)</sup> Credible surveillance data; see BAW-2325 for evaluation.

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

<sup>(E)</sup> Adjusted reference temperature (ART) calculated per Regulatory Guide 1.99, Rev. 2.  $ART = Initial\ RT_{NDT} + \Delta RT_{NDT} + Margin$ , where  $\Delta RT_{NDT} = Chemistry\ Factor \times Fluence\ Factor$ , and  $Margin = 2(\sigma_1^2 + \sigma_\Delta^2)^{0.5}$ , with  $\sigma_1$  defined as the standard deviation of the Initial  $RT_{NDT}$ , and  $\sigma_\Delta$  defined as the standard deviation of  $\Delta RT_{NDT}$ . For example, for nozzle belt forging, heat no. 123V352,  $ART = 40 + (76 \times 0.7399) + 34 = 130^\circ F$ . Calculated ART values are rounded to the nearest °F in accordance with the rounding-off method of ASTM Practice E29.

<sup>(F)</sup> Instruction Manual, Reactor Vessel, Point Beach Nuclear Plant Unit 2, Combustion Engineering, CE Book #4869, October 1970.

<sup>(G)</sup> By inspection, these are the limiting material properties.

<sup>(H)</sup> 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).

<sup>(I)</sup> 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  $K = \phi_{Best.Est.} E \sigma / \phi_{Calc} = 0.921$ .

PRESSURE TEMPERATURE LIMITS REPORT

TABLE 7  
POINT BEACH UNIT 1 RPV 3/4T BELTLINE MATERIAL ADJUSTED REFERENCE TEMPERATURES AT 25.59 EFPY  $\phi_{Calc}$   
(32.2 EFPY  $\phi_{Best.Est.}$ )<sup>(H)</sup>

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 <sup>(F)</sup>

Component Description	Heat or Heat/Lot	Initial RT <sub>NDT</sub> (°F)	%Cu	%Ni	CF	CF Method	3/4T 32.2 EFPY $\phi_{Best.Est.}$ Fluence Factor <sup>(A)</sup>	$\Delta RT_{NDT}$ (°F)	$\sigma_1$	$\sigma_\Delta$	Margin (°F)	ART (°F) <sup>(E)</sup>
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 <sup>(B)</sup>	"	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 <sup>(B)</sup>	"	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 <sup>(G)</sup>
Intermed. To Lower Shell Circ. Weld (100%)	71249 (SA-1101)	+10	0.23	0.59	167.6	Table <sup>(C)</sup>	0.8993	150.72	0	28	56	217 <sup>(G)</sup>
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 <sup>(D)</sup>	"	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  $\Delta RT_{NDT}$  are predicted  $\Delta RT_{NDT}$  based on Table CF is less than  $2\sigma$  (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.  $ART = Initial RT_{NDT} + \Delta RT_{NDT} + Margin$ , where  $\Delta RT_{NDT} = Chemistry Factor \times Fluence Factor$ , and  $Margin = 2(\sigma_1^2 + \sigma_\Delta^2)^{0.5}$ , with  $\sigma_1$  defined as the standard deviation of the Initial  $RT_{NDT}$ , and  $\sigma_\Delta$  defined as the standard deviation of  $\Delta RT_{NDT}$ . For example, for nozzle belt forging, heat no. 122P237,  $ART = 50 + (77 \times 0.5322) + 34 = 125^\circ F$ . 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  $K = \phi_{Best.Est.} E \sigma \tau / \phi_{Calc} = 0.838$ .

PRESSURE TEMPERATURE LIMITS REPORT

TABLE 8  
POINT BEACH UNIT 2 RPV 3/4T BELTLINE MATERIAL ADJUSTED REFERENCE TEMPERATURES AT 30.51 EFPY  $\phi_{Calc}$   
(34.0 EFPY  $\phi_{Best.Est.}$ )<sup>(I)</sup>

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 <sup>(F)</sup>

Component Description	Heat or Heat/Lot	Initial RT <sub>NDT</sub> (°F)	%Cu	%Ni	CF	CF Method	3/4T 32.2 EFPY $\phi_{Best.Est.}$ Fluence Factor <sup>(A)</sup>	$\Delta RT_{NDT}$ (°F)	$\sigma_1$	$\sigma_A$	Margin (°F)	ART (°F) <sup>(E)</sup>
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 <sup>(B)</sup>	0.9958	57.76	0	17	34	132 <sup>(G)</sup>
Lower Shell Forging	122W195	+40	0.05	0.72	31	Table	0.9456	29.31	0	17	34	103
	"	"			42.8	Surv. Data <sup>(C)</sup>	"	40.47	"	8.5	17	97
Nozzle Belt to Intermed. Shell Circ Weld (100%)	21935	-56	0.18	0.70	170	Table <sup>(H)</sup>	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 <sup>(D)</sup>	0.9405	169.29	19.7	28	68.47	233 <sup>(G)</sup>

Footnotes:

<sup>(A)</sup> See Table 2.

<sup>(B)</sup> Non-credible surveillance data; see BAW-2325 for evaluation. Table CF conservative because difference between measured  $\Delta RT_{NDT}$  and predicted  $\Delta RT_{NDT}$  based on Table CF is less than  $2\sigma$  (56°F).

<sup>(C)</sup> Credible surveillance data; see BAW-2325 for evaluation.

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

<sup>(E)</sup> Adjusted reference temperature (ART) calculated per Regulatory Guide 1.99, Rev. 2.  $ART = Initial\ RT_{NDT} + \Delta RT_{NDT} + Margin$ , where  $\Delta RT_{NDT} = Chemistry\ Factor \times Fluence\ Factor$ , and  $Margin = 2(\sigma_1^2 + \sigma_A^2)^{0.5}$ , with  $\sigma_1$  defined as the standard deviation of the Initial  $RT_{NDT}$ , and  $\sigma_A$  defined as the standard deviation of  $\Delta RT_{NDT}$ . For example, for nozzle belt forging, heat no. 123V352,  $ART = 40 + (76 \times 0.5435) + 34 = 115^\circ F$ . Calculated ART values are rounded to the nearest °F in accordance with the rounding-off method of ASTM Practice E29.

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

<sup>(G)</sup> By inspection, these are the limiting material properties.

<sup>(H)</sup> 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

<sup>(I)</sup> 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  $K = \phi_{Best.Est.} / \phi_{Calc} = 0.921$ .