March 1, 2013

NRC 2013-0022
10 CFR 50.90

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

Supplement to License Amendment Request 252
Technical Specification 5.6.5, Reactor Coolant System (RCS)
Pressure and Temperature Limits Report (PTLR)

References:

(1) NextEra Energy Point Beach, LLC letter to NRC, dated January 15, 2012, License Amendment Request 252 Technical Specification 5.6.5, Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR) (ML13016A028)
(2) NRC E-Mail to NextEra Energy Point Beach, dated 1 February 2013, Supplement to LAR re: P-T Limit Curves and LTOP Limits (TAC Nos. MF0532 & MF0533)

In Reference (1), NextEra Energy Point Beach, LLC (NextEra) submitted a license amendment request to amend renewed Facility Operating License Nos. DPR-24 and DPR-27 for Point Beach Nuclear Plant (PBNP), Units 1 and 2, respectively. The proposed amendments would revise the PBNP Technical Specifications (TS) to allow the use of two new methodologies: Framatome ANP Topical Report BAW-2308, Revisions 1-A and 2-A, "Initial RT_{NDT} of Linde 80 Weld Materials," and Westinghouse Owners Group (WOG) WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves." The revision would add BAW-2308, Revisions 1-A and 2-A and WCAP-14040-A, Revision 4, as approved methodologies to TS 5.6.5, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," for determining RCS pressure-temperature (PT) limits.

In Reference (2), the NRC informed NextEra that supplemental information would be required in order for the LAR to meet acceptance review criteria. This letter provides the requested information.

Enclosure 1 provides an engineering evaluation of the applicability of the PTLR using the requested methodologies. Enclosure 2 provides a mark-up of TRM 2.2, Pressure Temperature Limits Report. Enclosure 3 provides WCAP-16669-NP, "Point Beach Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation."

The information contained in this letter does not alter the no significant hazards consideration contained in Reference (1) and continues to satisfy the criteria of 10 CFR 51.22 for categorical exclusion from the requirements of an environmental assessment.

Approval of the proposed amendment is requested by January 1, 2014. NextEra will implement the amendment within 180 days of Commission Approval.

This letter contains no new Regulatory Commitments and no revisions to existing Regulatory Commitments.

The supplemental information to the LAR has been reviewed by the Plant Operations Review Committee.

In accordance with 10 CFR 50.91, a copy of this letter is being provided to the designated Wisconsin Official.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on March 1, 2013.

Very truly yours,

NextEra Energy Point Beach, LLC

Larry Meyer
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

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

SUPPLEMENT TO LICENSE AMENDMENT REQUEST 252
TECHNICAL SPECIFICATION 5.6.5, REACTOR COOLANT SYSTEM (RCS)
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

ENGINEERING EVALUATION
OF THE APPLICABILITY OF THE PTLR
USING THE REQUESTED METHODOLOGIES (MASTER CURVE)

1. PURPOSE AND SCOPE

This engineering evaluation provides the basis for the mark-up of TRM 2.2, Pressure Temperature Limits Report (PTLR). The current PTLR curves in TRM 2.2 are applicable to 35.9 effective full power years (EFPY). This evaluation provides the basis for new PTLR applicable to 50 EFPY. The PTLR mark-up is based on WCAP-16669-NP which uses the two new methodologies, BAW-2308, Revision 1-A, and WCAP-14040-A, Revision 4, for determining RCS pressure and temperature (PT) limits with updated fluences from power uprate as identified in WCAP-16983-P.

The enclosed PTLR mark-up in Enclosure 2 is based on 53 EFPY fluence projections which assumed 10% power uprates occurred for both units in 2008. This engineering evaluation determines the effective period for these PTLR curves using fluence projections with 17% power uprates for both units in 2011 (WCAP-16983-P).

2. BACKGROUND

License Amendment Request 252 (Reference 5) proposes to amend the Point Beach Nuclear Plant (PBNP) Technical Specifications (TS) to allow the use of two new methodologies; Framatome ANP Topical Report BAW-2308, Revisions 1-A (Reference 2) and 2-A (Reference 3), "Initial RT_{NDT} of Linde 80 Weld Materials," and Westinghouse Owners Group (WOG) WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" (Reference 6). The revision would add BAW-2308, Revisions 1-A and 2-A and WCAP-14040-A, Revision 4, as approved methodologies to TS 5.6.5, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," for determining RCS pressure-temperature (PT) limits.

On January 31, 2013, a teleconference between the NRC and PBNP staffs was held to discuss the NextEra Energy Point Beach, LLC (NextEra) submittal. The submittal involves a significant change in PTLR methodologies and the NRC staff did not have sufficient information to evaluate use of the two new methodologies at PBNP. NextEra was informed by the NRC they would need to provide a marked-up PTLR that used the proposed new methodologies. Also, any supporting documents would need to be provided.

3. APPROACH

Section 3 of WCAP-16669-NP describes the overall approach for the criteria for allowable pressure-temperature relationships. WCAP-16669-NP states the ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{Ic}, for the metal temperature at that time. K_{Ic} is obtained from the reference fracture toughness curve, defined in the 1998 Edition through the 2000 Addenda of Section XI, Appendix G of the ASME Code. The K_{Ic} curve is given by the following equation:

$$K_{Ic} = 33.2+20.734*e^{[0.02(T-RT_{NDT})]}$$

where,

K_{Ic} = reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature RT_{NDT}

The following equation is used to calculate values of ART for each weld and plate or forging in the reactor vessel beltline.

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin}$$

ΔRT_{NDT} is the mean value of the transition temperature shift, or change in ΔRT_{NDT}, due to irradiation, and is calculated using the following equation:

$$\Delta RT_{NDT} = (CF)*f^{(0.28 - 0.10 \log f)}$$

The fluence value, f, is the end-of-life neutron fluence, in units of 10^{19} n/cm^2 (E > 1.0 MeV), for each vessel beltline material.

For each beltline material, the only variable that would affect the applicability of the PTLR curves is the end-of-life neutron fluence value. The fluence values for each beltline material in WCAP-16669-NP will be compared to corresponding fluence versus EFPY tables in WCAP-16983-P. As long as the fluence value in WCAP-16983-P is less than the fluence value in WCAP-16669-NP for a given material, the proposed PTLR curves will remain effective.

4. PROPOSED PTLR CURVES IN WCAP-16669-NP

WCAP-16669-NP, "Point Beach Unit 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," was issued in December 2006. WCAP-16669-NP used fluence projections for 53 EFPY (End of Extended Life) based on the following assumptions:

- Power uprates of 10% (1678 MWth) for both units would be implemented in 2008, and
- Hafnium absorbers in the peripheral fuel assembly locations would be removed from both units in 2008.

WCAP-16669-NP provides the methodology and results of the generation of heatup and cooldown PT limit curves for normal operation of the PBNP Units 1 and 2 reactor vessels. These PT curves were generated based on the reactor vessel information and calculated fluences based on the plant operating conditions listed in Table 1-1 of WCAP-16669-NP.

Regulatory Guide 1.99, Revision 2, is used for the calculation of Adjusted Reference Temperature (ART) values ($IRT_{NDT} + \Delta RT_{NDT}$ + margins for uncertainties) at the surface, 1/4T and 3/4T locations, where T is the thickness of the vessel at the beltline region measured from the clad/base metal interface.

The heatup and cooldown curves in WCAP-16669-NP were generated using the most limiting ART values and the NRC approved methodology documented in WCAP-14040-NP-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves." Specifically, the "axial flaw" and "circ flaw" methodologies of the 1998 ASME Code, Section XI through the 2000 Addenda was used, which makes use of the K_{Ic} methodology.

The ART values were obtained from AREVA calculation 32-9019240-000, "ART Values for Point Beach Units 1 and 2." This document makes use of the NRC approved Topical Report BAW-2308, Revision 1-A, "Initial RT_{NDT} of Linde 80 Weld Materials." The fluence projections are based on the Ferret Code best-estimate values verified by Westinghouse in letter

WEP-06-13, dated February 14, 2006, as approved for use by the NRC in WCAP-16083-NP-A, Revision 0.

The 2008 PT limit curves were generated for 43 and 53 EFPY using heatup rates of 60 and 100°F/hr, cooldown rates of 0, 20, 40, 60 and 100°F/hr, using uprated conditions for both units, and with/without hafnium removal. The curves were developed without margins for instrumentation errors. These curves can be found in Figures 5-1 through 5-8 of WCAP-16669-NP.

The PT limit curves for Case 3 in WCAP-16669-NP (10% power uprate in 2008, 53 EFPY, and hafnium rods removed in 2008) shown in Table 1, were selected since this case has the highest fluence values.

Table 1

Plant Operating Conditions Reflecting Development of PT Limit Curves for 53 EFPY

Case	Unit	Power (MWth)	EFPY	Hafnium Rods?
3	1	1518.5 startup to 2/3/2003 1540.0 2/3/2003 to 10/2008 1678.0 10/2008 to 53 EFPY	53	Removal October 2008
	2	1518.5 startup to 2/3/2003 1540.0 2/3/2003 to 4/2008 1678.0 4/2008 to 53 EFPY	53	Removal April 2008

5. EXTENDED POWER UPRATE CONDITIONS IN WCAP-16983-P

WCAP-16983-P, Point Beach Units 1 and 2 Extended Power Uprate (EPU) Engineering Report, Section 5.1.2.4 provides results of the neutron fluence evaluations, including pressure vessel neutron exposure. The fluence projections are based on a 17% uprate (1800 MWth) of both units in 2011.

The fast neutron fluence (E > 1.0 MeV) values applicable to the clad/base metal interface for the PBNP Units 1 and 2 pressure vessels are provided in Tables 5.1.2-1 and 5.1.2-2 of WCAP-16983-P. The neutron exposure tabulations include fuel-cycle-specific results for Unit 1 at the conclusion of Cycle 31 (29.7 EFPY) and for Unit 2 at the conclusion of Cycle 29 (29.1 EFPY). The tabulations also include projections for both units for operation through the new period of PT curve applicability based on an assumed core power distribution without hafnium absorbers in peripheral fuel assembly locations.

As presented, the data in Tables 5.1.2-1 and 5.1.2-2 of WCAP-16983-P represent the maximum exposure at the clad/base metal interface for each of the materials making up the beltline region of the reactor pressure vessel. The beltline region is defined as that portion of the pressure vessel that is anticipated to be exposed to a neutron fluence (E > 1.0 MeV) greater than 1.0E+17 n/cm^2 over the service life of the unit. Pressure vessel materials not included in Tables 5.1.2-1 and 5.1.2-2 WCAP-16983-P are projected to be subjected to a neutron fluence < 1.0E+17 n/cm^2 and are, therefore, not considered a part of the beltline region.

The Low Temperature Overpressure Protection System (LTOPS) setpoint analysis for PBNP Units 1 and 2, January 2007, calculated the following LTOPS single setpoint applicable for 53 EFPY with and without hafnium rods for the entire LTOPS operating region:

LTOPS Single Setpoint = 420 psig (with RCP operating restriction, no more than one RCP in operation for RCS temperatures ≤ 180°F)

Section 4.3.4 of WCAP-16983-P concluded that for the EPU, the same LTOPS single setpoint remains acceptable with an arming temperature equal to 285°F.

6. COMPARISON OF PROJECTED FLUENCE VALUES

The fluence values for each beltline material in WCAP-16669-NP were compared to corresponding fluence versus EFPY tables in WCAP-16983-P at 50 EFPY as shown in Table 2.

Table 2

Comparison of Projected Fluence for Point Beach Nuclear Plant Units 1 and 2

Component Description	Heat or Heat/Lot	ID Fluence (x10^{19}) WCAP-16669-NP 53 EFPY	ID Fluence (x10^{19}) WCAP-16983-P 50 EFPY
Unit 1			
Nozzle Belt Forging	122P237	0.36	0.36
Intermediate Shell Plate	A9811-1	4.90	4.79
Lower Shell Plate	C1423-1	4.55	4.35
Nozzle Belt to Intermed. Shell Circ Weld (100%)	SA-1426	0.36	0.36
Intermediate Shell Long Seam (ID 27%)	SA-812	3.19	3.13
Intermediate Shell Long Seam (OD 73%)	SA-775	3.19	3.13
Intermed. to Lower Shell Circ. Weld (100%)	SA-1101	4.43	4.25
Lower Shell Long Seam (100%)	SA-847	3.05	2.94
Unit 2			
Nozzle Belt Forging	123V352	0.50	0.49
Intermediate Shell Forging	123V500	5.05	4.76
Lower Shell Forging	122W195	4.90	4.57
Nozzle Belt to Intermed. Shell Circ Weld (100%)	21935	0.50	0.49
Intermed. to Lower Shell Circ. Weld (100%)	72442 (SA-1484)	4.65	4.34

As illustrated in Table 2, the projected fluence values for all of the beltline materials are less than the fluence values used to generate the proposed PTLR curves. Thus, the PTLR curves remain conservative and valid through a 50 EFPY period of applicability.

Reactor vessel total EFPY values were determined in December 2012 to be 33.35 EFPY for Unit 1 and 32.8 EFPY for Unit 2. Assuming the units operate 95% of the time in the future (5% for future refueling outages), the period of applicability for the PTLR curves would be:

(50 EFPY – 33.35 EFPY) / 0.95 = 17.5 years

This additional 17.5 years would make the PTLR effective through the end of 2029.

7. CONCLUSION

The proposed RCS Pressure/Temperature (P/T) limits for PBNP Units 1 and 2 Technical Specifications are valid and conservative through 50 Effective Full Power Years (EFPY) (approximately through 2029).

8. REFERENCES

1. WCAP-16669-NP, “Point Beach Unit 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation”, dated December 2006
2. Framatome ANP Topical Report BAW-2308, Revision 1-A, “Initial RT_{NDT} of Linde 80 Weld Materials,” approved August 2005
3. Framatome ANP Topical Report BAW-2308, Revision 2-A, “Initial RT_{NDT} of Linde 80 Weld Materials,” approved March 2008
4. WCAP-16983-P, “Point Beach Units 1 and 2 Extended Power Uprate (EPU) Engineering Report”, dated September 2009
5. NextEra Energy Point Beach, LLC letter to NRC, dated January, 15 2012, License Amendment Request 252 Technical Specification 5.6.5, Reactor Coolant System (RCS) Pressure and Temperature Limits Report (ML13016A028)
6. Westinghouse Owners Group (WOG) WCAP-14040-A, Revision 4, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves”, dated May 2004

ENCLOSURE 2

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

SUPPLEMENT TO LICENSE AMENDMENT REQUEST 252
TECHNICAL SPECIFICATION 5.6.5, REACTOR COOLANT SYSTEM (RCS)
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

MARK-UP OF TRM 2.2
PRESSURE TEMPERATURE LIMITS REPORT

16 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, and XXXXXXX..

The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto (Ref 5.19). Based upon fluence values in Westinghouse ~~report LTR-REA-08-144~~ WCAP-16983-P (Ref 5.15), this PTLR is effective for ~~35.9~~ 50 EFPY (approximately ~~June 2014~~ 2029). ~~(Ref 5.8)~~

The Technical Specifications addressed in this report are listed below:

1.1 3.4.3 Pressure/Temperature (P-T) Limits

1.2 3.4.12 Low Temperature Overpressure Protection (LTOP) System

2.0 OPERATING LIMITS

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

2.1 RCS Pressure and Temperature Limits (LCO 3.4.3)

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

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

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

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

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

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

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

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

2.3 Low Temperature Overpressure Protection System Setpoints (LCO 3.4.12)

Pressurizer Power-Operated Relief Valve Lift Setting Limits

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

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

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

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

2.4 Criticality and Hydrostatic Leak Test Limits

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

3.0 REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM

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

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

The pressure vessel surveillance program is in compliance with Appendix H to 10 CFR 50, entitled, "Reactor Vessel Radiation Surveillance Program." The material test requirements and the acceptance standard utilize the nil-ductility temperature, RT_{NDT}, which is determined in accordance with ASTM E208. The empirical relationship between RT_{NDT} and the fracture toughness of the reactor vessel steel is developed in accordance with Appendix G, "Protection Against Non-Ductile Failure," to Section XI of the ASME Boiler and Pressure Vessel Code. The surveillance capsule removal schedule meets the requirements of ASTM E185-82.

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

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

4.0 SUPPLEMENTAL DATA INFORMATION

The limiting RT_{PTS} values for the PBNP limiting beltline materials at ~~35.9~~ 50 EFPY are:

- Unit 1 - Intermediate to Lower Shell Circ Weld = ~~281.0~~ 230.7ºF; Lower Shell Axial Weld = ~~250.3~~ 220.0ºF (Ref. ~~5.8, Attachment A~~ 5.2, Table 4-10)
- Unit 2 - Intermediate to Lower Shell Circ Weld = ~~295.1~~ 265.2ºF; Intermediate Shell Forging = ~~150.2~~ 150.6ºF (Ref. ~~5.8, Attachment A~~ 5.2, Table 4-10)

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~~ 4, May 2004

5.2 ~~WCAP-15976~~, WCAP-16669, "Point Beach Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," Revision 1, ~~March 2008~~ January 2009

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~~ Low Temperature Overpressure Protection System (LTOPS) Setpoint Analysis, January 2007

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

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

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

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

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

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

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

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

5.13 Deleted

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

~~5.15 Westinghouse Letter LTR-REA-08-144, "Summary of Neutron Fluence Evaluations for the Point Beach Units 1 and 2 Extended Power Uprate," dated January 2009~~ WCAP-16983-P, Point Beach Units 1 and 2 Extended Power Uprate (EPU) Engineering Report

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

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

5.18 Deleted

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

5.20 CL 4C, Low Temperature Overpressurization Protection Unit 1

5.21 CL 4C, Low Temperature Overpressurization Protection Unit 2

5.22 OP 3C, Hot Standby to Cold Shutdown

5.23 OP 4B, Reactor Coolant Pump Operation

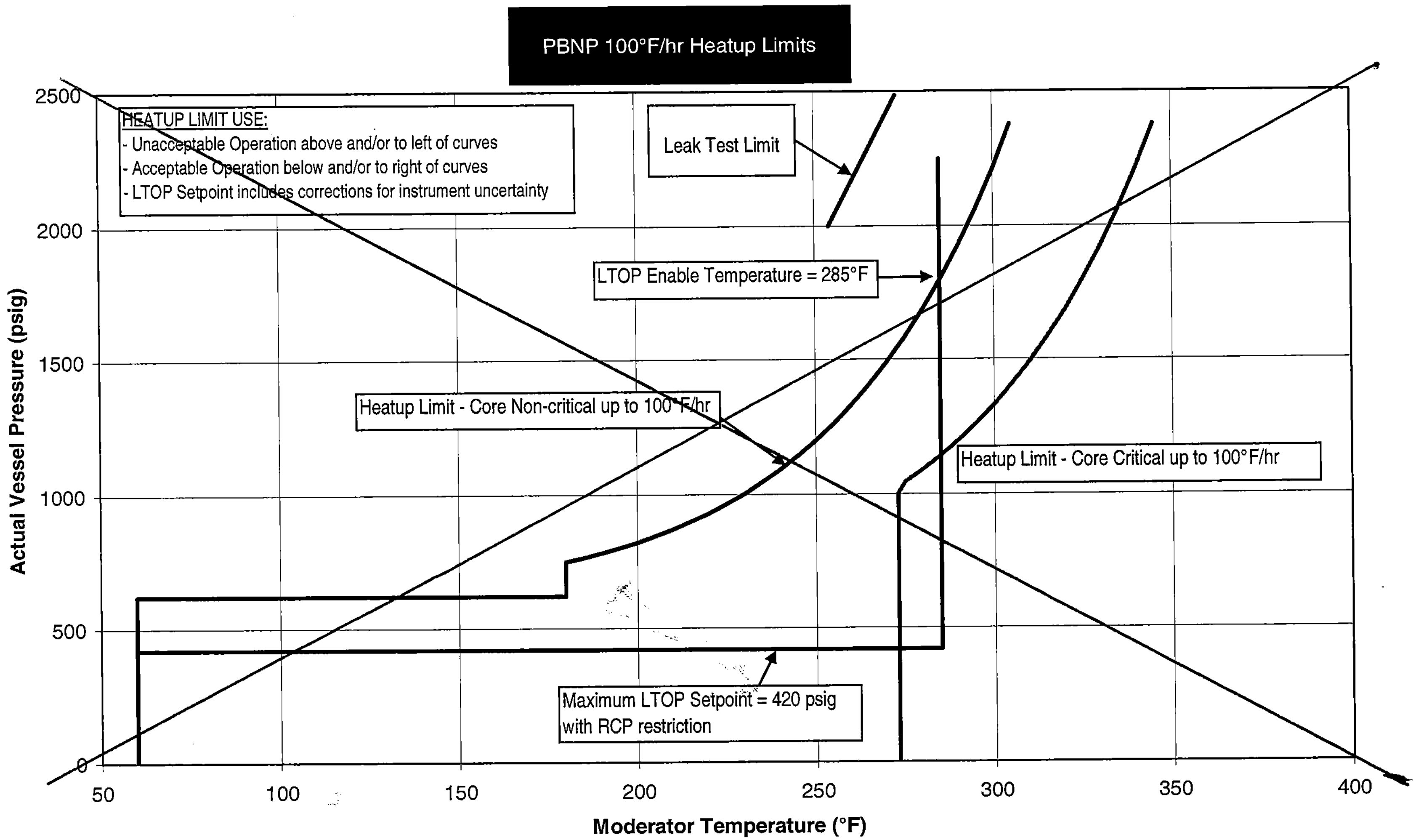
5.24 OP 1A, Cold Shutdown to Hot Standby

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

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

REPLACE FIGURE WITH FIGURE 5-7 OF WCAP-16669-NP

Figure 1
RCS PRESSURE-TEMPERATURE LIMITS FOR HEATUP



MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Intermediate Shell Longitudinal Welds SA-812 (ID) and SA-775 (OD)
LIMITING ART VALUES AT 53 EFPY (Hafnium Removal): 1/4T, 220.0°F
3/4T, 184.6°F

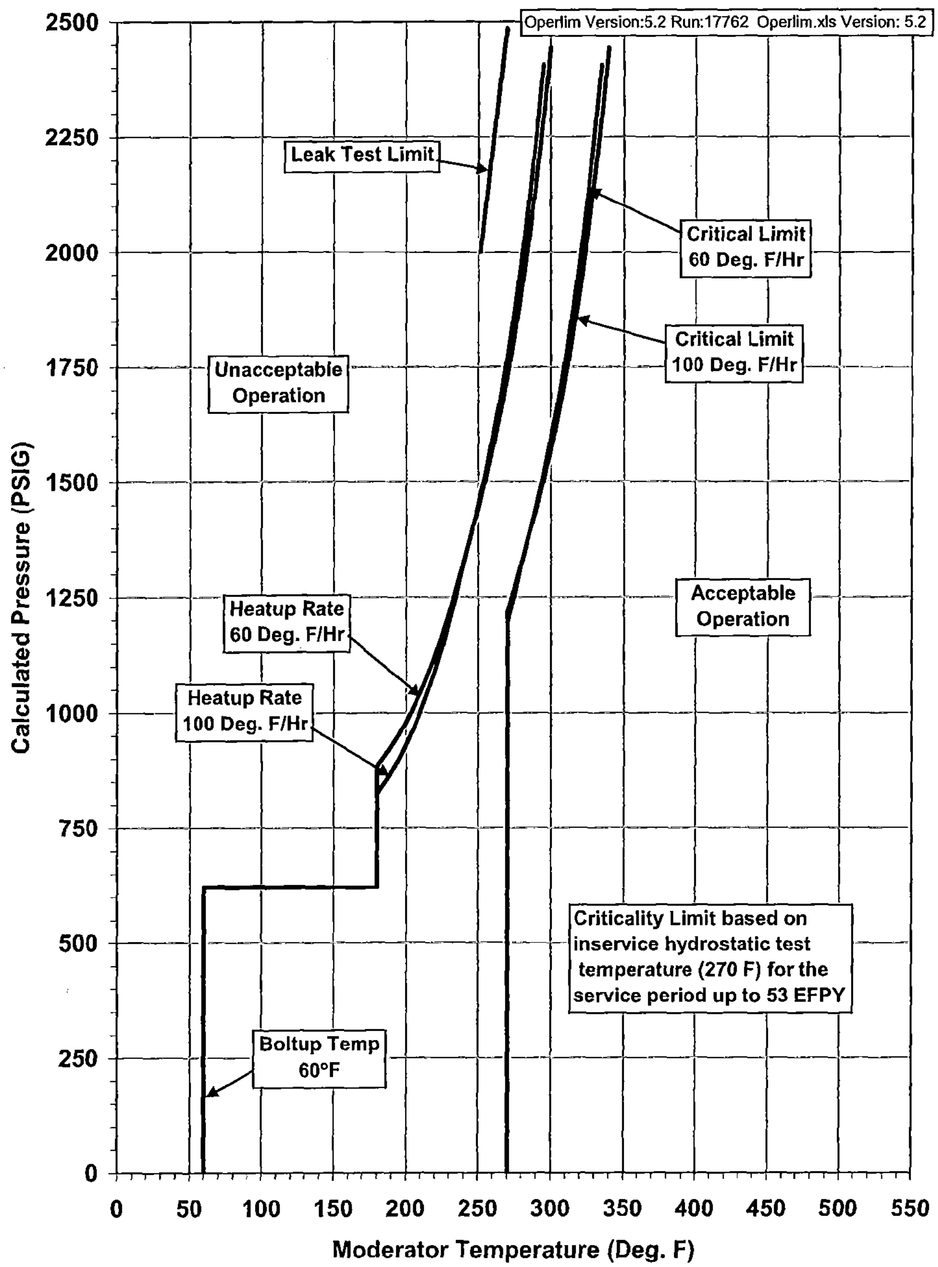
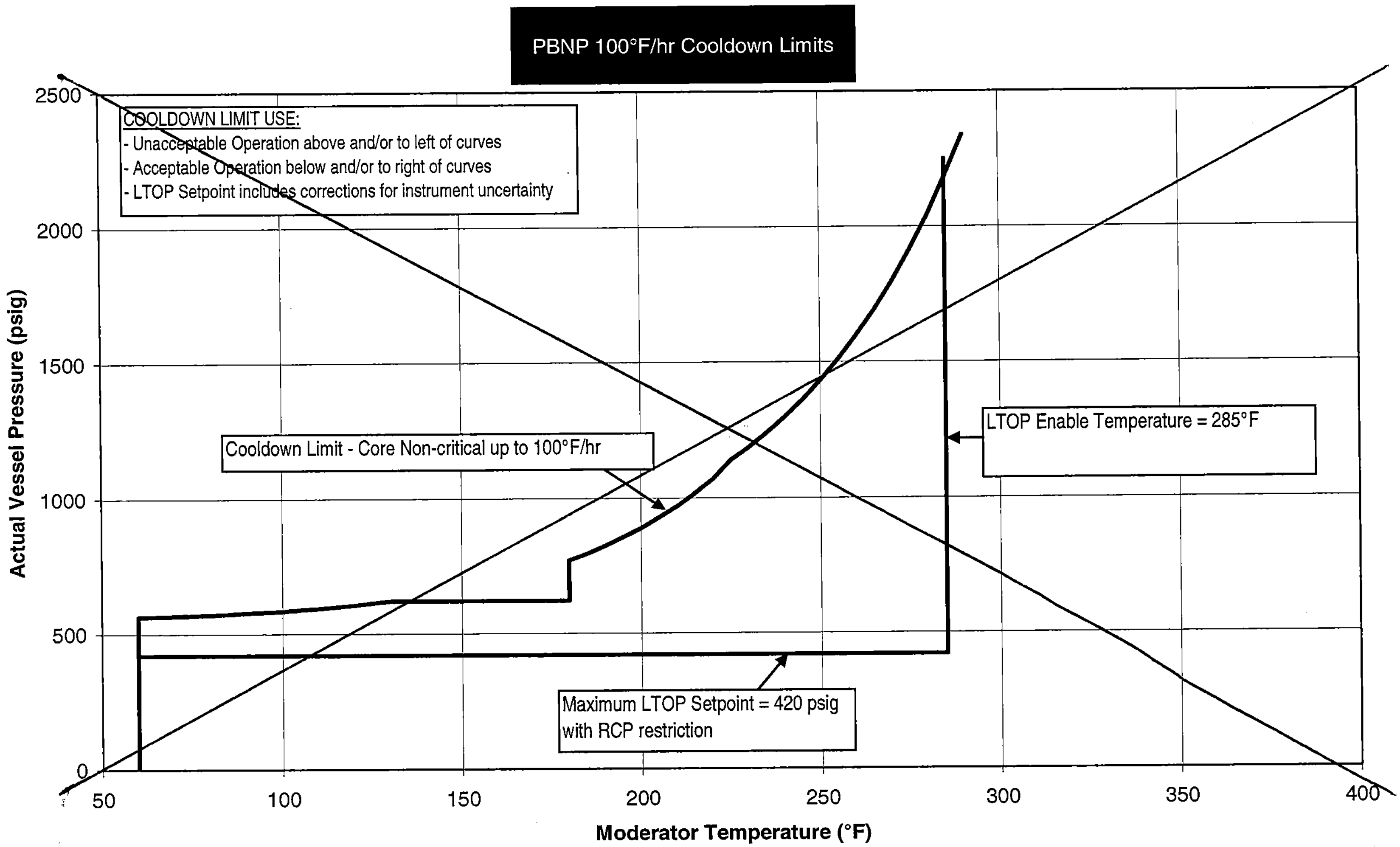


Figure 5-7 **Point Beach Units 1 and 2 Reactor Coolant System Heatup Limitations (Heatup Rates of 60 and 100°F/hr) Applicable for 53 EFPY (with Hafnium Removal and without Margins for Instrumentation Errors) Using 1998 App. G Methodology (w/K_{Ic})**

REPLACE FIGURE WITH FIGURE 5-8 OF WCAP-16669-NP

Figure 2
RCS PRESSURE-TEMPERATURE LIMITS FOR COOLDOWN



MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Intermediate Shell Longitudinal Welds SA-812 (ID) and SA-775 (OD)
LIMITING ART VALUES AT 53 EFPY (Hafnium Removal): 1/4T, 220.0°F
3/4T, 184.6°F

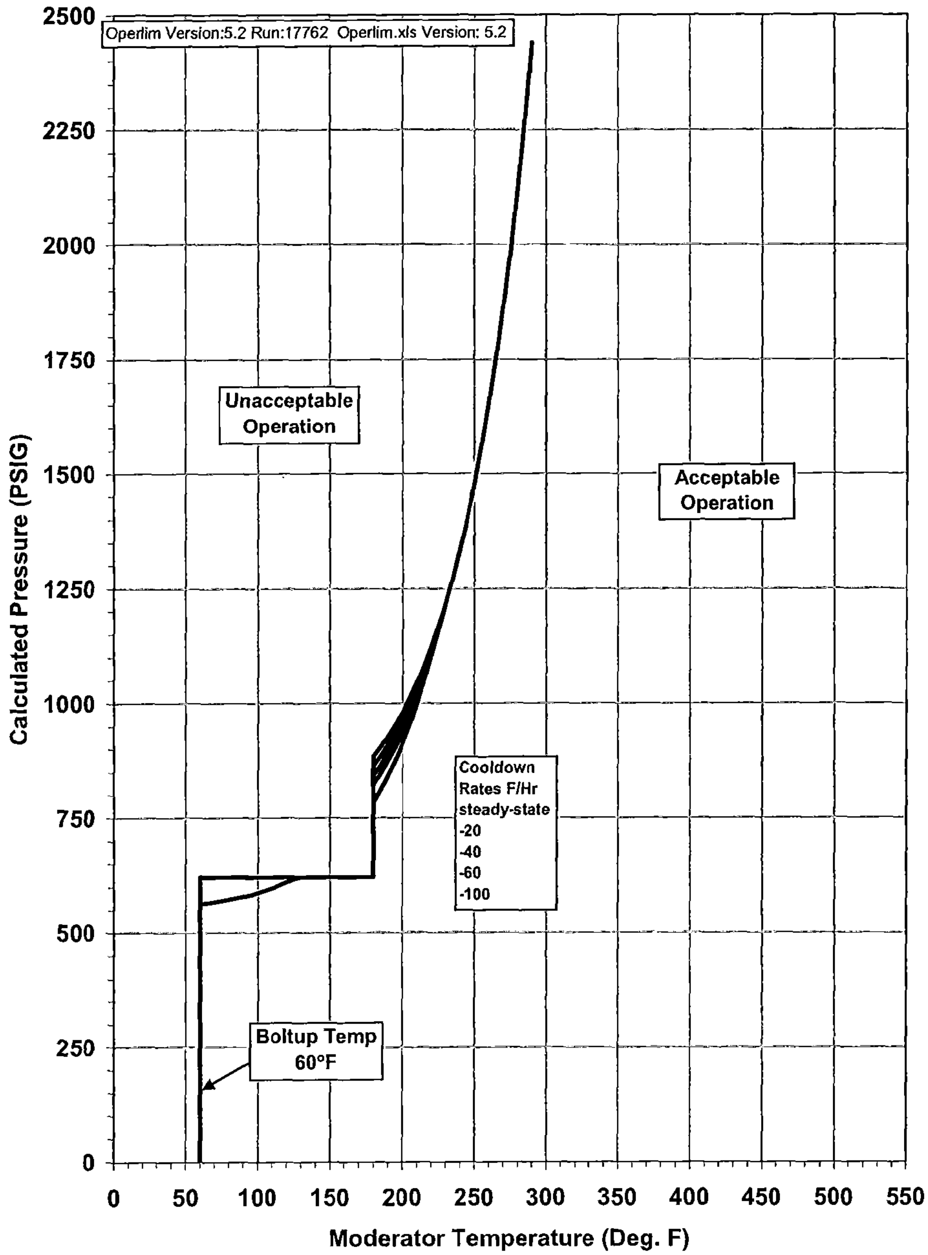


Figure 5-8 Point Beach Units 1 and 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for 53 EFPY (with Hafnium Removal and without Margins for Instrumentation Errors) Using 1998 App. G Methodology (w/K_{Ic})

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

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

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

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

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

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

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

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

TABLE 3
POINT BEACH UNIT 1 RPV BELTLINE ~~35.9~~ 50 EFPY VALUES[(E)]

Based on Westinghouse Report ~~WCAP-15976,~~WCAP-16669, Revision 1, "Point Beach Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," (Ref 5.2). ~~Note that the estimated fluence at a specific point in time is not linearly interpolated between zero and the estimated fluence at 36.9 EFPY, due to changes in core design at certain points in the operating history of the unit. Although the analysis in WCAP-15976 is based on 36.9 EFPY, the applicability of the analysis is now 35.9 EFPY per LTR-PCAM-08-57, "Point Beach Units 1 and 2 EPU P-T Limit Curve Applicability Determination and Related Calculations" (Ref 5.8).~~

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

Component Description	Heat or Heat/Lot	~~35.9 EFPY [(E)]~~ Inside Surface Fluence (E19 n/cm^2)	~~35.9 EFPY [(E)]~~ 1/4T Fluence (E19 n/cm^2) [(B)]	~~35.9 EFPY [(E)] 1/4T Fluence Factor [(C)]~~	~~35.9 EFPY [(E)]~~ 3/4T Fluence (E19 n/cm^2) [(B)]	~~35.9 EFPY [(E)] 3/4T Fluence Factor [(C)]~~
Nozzle Belt Forging	122P237	~~0.25~~ 0.36	~~0.17~~ 0.24	~~0.53~~	~~0.08~~ 0.11	~~0.37~~
Intermediate Shell Plate	A9811-1	~~3.38~~ 4.90	~~2.29~~ 3.32	~~1.22~~	~~1.05~~ 1.52	~~1.01~~
Lower Shell Plate	C1423-1	~~3.04~~ 4.55	~~2.06~~ 3.09	~~1.20~~	~~0.94~~ 1.41	~~0.98~~
Nozzle Belt to Intermed. Shell Circ Weld (100%)	8T1762 (SA-1426)	~~0.25~~ 0.36	~~0.17~~ 0.24	~~0.53~~	~~0.08~~ 0.11	~~0.37~~
Intermediate Shell Long Seam (ID 27%)	1P0815 (SA-812)	~~2.19~~ 3.19	~~1.48~~ 2.16	~~1.11~~	N/A	~~N/A~~
Intermediate Shell Long Seam (OD 73%) [(A)]	1P0661 (SA-775)	~~2.19~~ 3.19	N/A	~~N/A~~	~~0.68~~ 0.99	~~0.89~~
Intermed. to Lower Shell Circ. Weld (100%)	71249 (SA-1101)	~~3.05~~ 4.43	~~2.07~~ 3.00	~~1.20~~	~~0.95~~ 1.38	~~0.99~~
Lower Shell Long Seam [(A)] (100%)	61782 (SA-847)	~~2.08~~ 3.05	~~1.41~~ 2.07	~~1.10~~	~~0.65~~ 0.95	~~0.88~~

Footnotes:

(A) Limiting material

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

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

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

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

TABLE 4
POINT BEACH UNIT 2 RPV BELTLINE ~~35.9~~ 50 EFPY VALUES[(E)]

Based on Westinghouse Report ~~WCAP-15976,~~WCAP-16669, Revision 1, "Point Beach Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," (Ref 5.2). ~~Note that the estimated fluence at a specific point in time is not linearly interpolated between zero and the estimated fluence at 36.9 EFPY, due to changes in core design at certain points in the operating history of the unit. Although the analysis in WCAP-15976 is based on 36.9 EFPY, the applicability of the analysis is now 35.9 EFPY per LTR-PCAM-08-57, "Point Beach Units 1 and 2 EPU P-T Limit Curve Applicability Determination and Related Calculations" (Ref 5.8).~~

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

Component Description	**Heat or Heat/Lot**	**~~35.9 EFPY~~ [~~(E)~~] Inside Surface Fluence (E19 n/cm^2)**	**~~35.9 EFPY~~ [~~(E)~~] 1/4T Fluence (E19 n/cm^2) [(B)]**	**~~35.9 EFPY~~ [~~(E)~~] ~~1/4T Fluence Factor~~ [~~(C)~~]**	**~~35.9 EFPY~~ [~~(E)~~] 3/4T Fluence (E19 n/cm^2) [(B)]**	**~~35.9 EFPY~~ [~~(E)~~] ~~3/4T Fluence Factor~~ [~~(C)~~]**
Nozzle Belt Forging	123V352	~~0.34~~ 0.50	~~0.23~~ 0.34	~~0.60~~	~~0.11~~ 0.16	~~0.44~~
Intermediate Shell Forging [(A)]	123V500	~~3.38~~ 5.05	~~2.29~~ 3.42	~~1.22~~	~~1.05~~ 1.57	~~1.01~~
Lower Shell Forging	122W195	~~3.30~~ 4.90	~~2.23~~ 3.32	~~1.22~~	~~1.02~~ 1.52	~~1.01~~
Nozzle Belt to Intermed. Shell Circ Weld (100%)	21935	~~0.34~~ 0.50	~~0.23~~ 0.34	~~0.60~~	~~0.11~~ 0.16	~~0.44~~
Intermed. to Lower Shell Circ Weld (100%) [(A)]	72442 (SA-1484)	~~3.13~~ 4.65	~~2.12~~ 3.15	~~1.20~~	~~0.97~~ 1.44	~~0.99~~

Footnotes:

(A) Limiting Material

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

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

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

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

TABLE 5
POINT BEACH UNIT 1 RPV 1/4T BELTLINE MATERIAL ADJUSTED REFERENCE TEMPERATURES AT ~~35.9~~ 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. 5.6) and WCAP-15976,~~WCAP-16669, Revision 1, "Point Beach Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," (Ref 5.2). ~~Although the analysis in WCAP-15976 is based on 36.9 EFPY, the applicability of the analysis is now 35.9 EFPY per LTR-PCAM-08-57, "Point Beach Units 1 and 2 EPU P-T Limit Curve Applicability Determination and Related Calculations" (Ref 5.8).~~

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

Component Description	Heat or Heat/Lot	Initial RT_{NDT} (°F)	%Cu	%Ni	CF	CF Method	~~1/4T 35.9 EFPY(H) Fluence Factor(A)~~	ΔRT_{NDT} (°F)	~~σ_I~~	~~σ_Δ~~	Margin (°F)	ART (°F)(E)
Nozzle Belt Forging	122P237	+50	0.11	0.82	77	Table	~~0.53~~	~~40.8~~ 47.4	~~0~~	~~17~~	34	~~125~~ 131.4
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~~ 104.1	"	~~8.5~~	56.4	~~154~~ 161.5
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~~ 46.4	"	~~8.5~~	56.4	~~100~~ 103.8
Nozzle Belt to Intermed. Shell Circ Weld (100%)	8T1762 (SA-1426)	~~-5~~ -47.6	0.19	0.57	~~152.4~~ 167.0	Table	~~0.53~~	~~80.8~~ 102.9	~~19.7~~	~~28~~	~~68.47~~ 65.7	~~144~~ 121.0
Intermediate Shell Long Seam (ID 27%)	1P0815 (SA-812)	~~-5~~ -47.6	0.17	0.52	~~138.2~~ 167.0	Table	~~1.11~~	~~153.4~~ 201.9	~~19.7~~	~~28~~	~~68.47~~ 65.7	~~217~~ 220.0
Intermediate Shell Long Seam (OD 73%)	1P0661 (SA-775)	~~-5~~ -47.6	0.17	0.64	~~157.6~~ 167.0	Table	~~N/A~~	N/A	~~19.7~~	~~28~~	N/A	N/A
Intermed. to Lower Shell Circ. Weld (100%)	71249 (SA-1101)	~~+10~~ -47.4	0.23	0.59	167.6	Table(C)	~~1.20~~	~~201.1~~ 216.4	~~0~~	~~28~~	~~56~~ 61.7	~~267~~ 230.7
Lower Shell Long Seam (100%)	61782 (SA-847)	~~-5~~ -47.6	0.23	0.52	~~157.4~~ 167.0	Table	~~1.10~~	~~173.1~~ 199.9	~~19.7~~	~~28~~	~~68.47~~ 65.7	~~237~~ 218.1
"	"	"			~~163.3~~	~~Surv. Data(D)~~	"	~~179.6~~	"	~~14~~	~~48.34~~	~~223~~

Footnotes:

(A) See Table 3

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

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

(D) Credible Surveillance Data; see WE Calculation Addendum 98-0156-00-A, "Evaluation of New Surveillance Data on Chemistry Factor for Weld Wire Heat 61782, Point Beach Unit 1," (Ref.5.3) utilizing latest time-weighted temperature data for Point Beach Unit 1, which supersedes BAW-2325.

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

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

(G) Deleted.

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

TABLE 6
POINT BEACH UNIT 2 RPV 1/4T BELTLINE MATERIAL ADJUSTED REFERENCE TEMPERATURES AT ~~35.9~~ 50 EFPY[(I)]

Unless otherwise noted, all ART input data obtained from ~~BAW-2325, "Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity," May 1998 (Ref. 5.6) and WCAP-15976,~~ WCAP-16669, Revision 1, "Point Beach Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," (Ref 5.2). ~~Although the analysis in WCAP-15976 is based on 36.9 EFPY, the applicability of the analysis is now 35.9 EFPY per LTR-PCAM-08-57, "Point Beach Units 1 and 2 EPU P-T Limit Curve Applicability Determination and Related Calculations" (Ref 5.8).~~

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

Component Description	Heat or Heat/Lot	Initial RT_{NDT} (°F)	%Cu	%Ni	CF	CF Method	~~1/4T 35.9~~ 50 ~~(I) EFPY Fluence Factor(A)~~	ΔRT_{NDT} (°F)	~~σ_I~~	~~σ_Δ~~	Margin (°F)	ART (°F)[(E)]
Nozzle Belt Forging	123V352	+40	0.11	0.73	76	Table	~~0.60~~	~~45.6~~ 53.5	~~0~~	~~17~~	34	~~120~~ 127.5
Intermediate Shell Forging	123V500	+40	0.09	0.70	58	Table[(B)]	~~1.22~~	~~70.8~~ 76.6	~~0~~	~~17~~	34	~~145~~ 150.6
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~~ 56.2	~~0~~	~~8.5~~	17	~~110~~ 113.2
Nozzle Belt to Intermed. Shell Circ Weld (100%)	21935	-56	0.18	0.70	170	Table[(H)]	~~0.60~~	~~102~~ 120.0	~~17~~	~~28~~	65.5	~~112~~ 129.5
Intermed. to Lower Shell Circ. Weld (100%)	72442 (SA-1484)	~~-5~~ -30	0.26	0.60	180	Table[(D)]	~~1.20~~	~~216.0~~ 234.4	~~19.7~~	~~28~~	~~68.47~~ 60.8	~~280~~ 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 $RT_{NDT} + \Delta RT_{NDT}$ + 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 Revision 1 (Ref 5.2) for discussion of EFPY values. The 36.9 EFPY values listed in WCAP-15976, Revision 1, are now applicable to 35.9 EFPY.~~

TABLE 7
POINT BEACH UNIT 1 RPV 3/4T BELTLINE MATERIAL ADJUSTED REFERENCE TEMPERATURES AT ~~35.9~~ **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. 5.6) and ~~WCAP-15976,~~WCAP-16669, Revision 1, "Point Beach Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," (Ref 5.2). ~~Although the analysis in WCAP-15976 is based on 36.9 EFPY, the applicability of the analysis is now 35.9 EFPY per LTR-PCAM-08-57, "Point Beach Units 1 and 2 EPU P-T Limit Curve Applicability Determination and Related Calculations" (Ref 5.8).~~

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

Component Description	Heat or Heat/Lot	Initial RT_{NDT} (°F)	%Cu	%Ni	CF	CF Method	~~3/4T 35.9 EFPY[H] Fluence Factor[A]~~	ΔRT_{NDT} (°F)	~~σ_I~~	~~σ_Δ~~	Margin (°F)	ART (°F)[E]
Nozzle Belt Forging	122P237	+50	0.11	0.82	77	Table	~~0.37~~	~~28.5~~ 33.7	~~0~~	~~17~~	~~34~~ 33.7	~~113~~ 117.4
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~~ 88.5	~~26.9~~	~~8.5~~	56.4	~~138~~ 145.9
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~~ 39.2	~~26.9~~	~~8.5~~	56.4	~~93~~ 96.6
Nozzle Belt to Intermed. Shell Circ Weld (100%)	8T1762 (SA-1426)	~~-5~~ -47.6	0.19	0.57	~~152.4~~ 167.0	Table	~~0.37~~	~~56.4~~ 73.2	~~19.7~~	~~28~~	~~68.47~~ 65.7	~~120~~ 91.3
Intermediate Shell Long Seam (ID 27%)	1P0815 (SA-812)	~~-5~~ -47.6	0.17	0.52	~~138.2~~ 167.0	Table	~~N/A~~	N/A	~~19.7~~	~~28~~	~~68.47~~ N/A	N/A
Intermediate Shell Long Seam (OD 73%)	1P0661 (SA-775)	~~-5~~ -47.6	0.17	0.64	~~157.6~~ 167.0	Table	~~0.89~~	~~140.3~~ 166.5	~~19.7~~	~~28~~	~~68.47~~ 65.7	204 184.6
Intermed. To Lower Shell Circ. Weld (100%)	71249 (SA-1101)	~~+10~~ -47.4	0.23	0.59	167.6	Table[C]	~~0.99~~	~~165.9~~ 182.3	~~0~~	~~28~~	~~56~~ 61.7	232 196.6
Lower Shell Long Seam (100%)	61782 (SA-847)	~~-5~~ -47.6	0.23	0.52	~~157.4~~ 167.0	Table	~~0.88~~	~~138.5~~ 164.5	~~19.7~~	~~28~~	~~68.47~~ 65.7	202 182.7
"	"	~~"~~			~~163.3~~	~~Surv. Data[D]~~	~~"~~	~~143.7~~	~~"~~	~~14~~	~~48.34~~	~~187~~

Footnotes:

(A) See Table 3.

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

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

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

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

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

(G) Deleted.

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

TABLE 8
POINT BEACH UNIT 2 RPV 3/4T BELTLINE MATERIAL ADJUSTED REFERENCE TEMPERATURES AT ~~35.9~~ 50 EFPY [I]

Unless otherwise noted, all ART input data obtained from ~~BAW-2325, "Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity," May 1998 (Ref. 5.6) and WCAP-15976,~~ WCAP-16669, Revision 1, "Point Beach Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," (Ref 5.2). ~~Although the analysis in WCAP-15976 is based on 36.9 EFPY, the applicability of the analysis is now 35.9 EFPY per LTR-PCAM-08-57, "Point Beach Units 1 and 2 EPU P-T Limit Curve Applicability Determination and Related Calculations" (Ref 5.8).~~

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

Component Description	Heat or Heat/Lot	Initial RT_{NDT} (°F)	%Cu	%Ni	CF	CF Method	~~3/4T 35.9 EFPY[I] Fluence Factor~~[A]	ΔRT_{NDT} (°F)	~~σ_I~~	~~σ_Δ~~	Margin (°F)	ART (°F)[E]
Nozzle Belt Forging	123V352	+40	0.11	0.73	76	Table	~~0.44~~	~~33.4~~ 38.9	~~0~~	~~17~~	34	~~107~~ 112.9
Intermediate Shell Forging	123V500	+40	0.09	0.70	58	Table[B]	~~1.01~~	~~58.6~~ 65.2	~~0~~	~~17~~	34	~~133~~ 139.2
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~~ 47.8	"	~~8.5~~	17	~~100~~ 104.8
Nozzle Belt to Intermed. Shell Circ Weld (100%)	21935	-56	0.18	0.70	170	Table[H]	~~0.44~~	~~74.8~~ 87.3	~~17~~	~~28~~	65.5	~~84~~ 96.8
Intermed. to Lower Shell Circ. Weld (100%)	72442 (SA-1484)	~~-5~~ -30	0.26	0.60	180	Table [D]	~~0.99~~	~~178.2~~ 198.4	~~19.7~~	~~28~~	~~68.47~~ 60.8	~~242~~ 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 $RT_{NDT} + \Delta RT_{NDT}$ + Margin, where ΔRT_{NDT} = Chemistry Factor × Fluence Factor, and Margin = $2(\sigma_I^2 + \sigma_\Delta^2)^{0.5}$, with σ_I defined as the standard deviation of the Initial RT_{NDT}, and σ_Δ defined as the standard deviation of ΔRT_{NDT}. ~~Calculated ART values are rounded to the nearest °F in accordance with the rounding-off method of ASTM Practice E29.~~
(F) Instruction Manual, Reactor Vessel, Point Beach Nuclear Plant No. 2, Combustion Engineering, CE Book #4869, October 1970.
(G) Deleted.
(H) Table CF value based on best-estimate chemistry data from CEOG Report "Best Estimate Copper and Nickel Values in CE Fabricated Reactor Vessel Welds," CE NPSD-1039, Revision 2, Final Report, June 1997
(I) EFPY value listed here is based on various reactor fuel management strategies and reactor power levels. ~~See WCAP-15976 Revision 1 (Ref 5.2) for discussion of EFPY values. The 36.9 EFPY values listed in WCAP-15976, Revision 1, are now applicable to 35.9 EFPY.~~

ENCLOSURE 3

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

SUPPLEMENT TO LICENSE AMENDMENT REQUEST 252
TECHNICAL SPECIFICATION 5.6.5, REACTOR COOLANT SYSTEM (RCS)
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

WCAP-16669-NP
POINT BEACH UNITS 1 AND 2
HEATUP AND COOLDOWN LIMIT CURVEES
FOR NORMAL OPERATION

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Point Beach Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation

Westinghouse

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Point Beach Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation

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PREFACE

This report has been technically reviewed and verified by:

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RECORD OF REVISION

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EXECUTIVE SUMMARY

This report provides the methodology and results of the generation of heatup and cooldown pressure temperature (PT) limit curves for normal operation of the Point Beach Units 1 and 2 reactor vessels. The PT curves were generated based on the latest available reactor vessel information and updated calculated fluences.

The new Point Beach Unit 1 and 2 heatup and cooldown pressure-temperature limit curves were generated using adjusted reference temperature (ART) values that bound both units. The highest ART values from the two units were from the Unit 2 intermediate to lower shell girth weld, however the limiting materials are actually the intermediate shell axial welds from Unit 1, depending on the vessel thickness [¼ thickness (1/4T) or ¾ thickness (3/4T) locations]. The axial welds become limiting over the girth weld through use of "circ-flaw" methodology from ASME Code Case N-588. This methodology is less restrictive than the standard "axial-flaw" methodology from the 1998 ASME Code, Section XI through 2000 Addenda. In addition to the use of Code Case N-588, the PT curves also made use of the K_{Ic} methodology detailed in ASME Code Case N-640. Both ASME Code Case N-588 and N-640 were joined together under ASME Code Case N-641 and incorporated into the 1998 ASME Code, Section XI, through 2000 Addenda.

The PT limit curves were generated for 43 and 53 EFPY using heatup rates of 60 and 100°F/hr, cooldown rates of 0, 20, 40, 60 and 100°F/hr, with uprated conditions for both units, and with/without hafnium removal in 2008. The curves were developed without margins for instrumentation errors. These curves can be found in Figures 5-1 through 5-8.

1 INTRODUCTION

Heatup and cooldown limit curves are calculated using the adjusted RT_{NDT} (reference nil-ductility temperature) corresponding to the limiting beltline region material of the reactor vessel. The adjusted RT_{NDT} of the limiting material in the core region of the reactor vessel is determined by using the unirradiated reactor vessel material fracture toughness properties, estimating the radiation-induced ΔRT_{NDT}, and adding a margin. The unirradiated RT_{NDT} is designated as the higher of either the drop weight nil-ductility transition temperature (NDTT) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (normal to the major working direction) minus 60°F.

RT_{NDT} increases as the material is exposed to fast-neutron radiation. Therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the unirradiated RT_{NDT} (IRT_{NDT}). The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and nickel) present in reactor vessel steels. The Nuclear Regulatory Commission (NRC) has published a method for predicting radiation embrittlement in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials" [Reference 1]. Regulatory Guide 1.99, Revision 2, is used for the calculation of Adjusted Reference Temperature (ART) values (IRT_{NDT} + ΔRT_{NDT} + margins for uncertainties) at the surface, 1/4T and 3/4T locations, where T is the thickness of the vessel at the beltline region measured from the clad/base metal interface.

The heatup and cooldown curves documented in this report were generated using the most limiting ART values and the NRC approved methodology documented in WCAP-14040-NP-A, Revision 4 [Reference 2], "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves." Specifically, the "axial flaw" and "circ flaw" methodologies of the 1998 ASME Code, Section XI through the 2000 Addenda [Reference 3] was used, which makes use of the K_{Ic} methodology.

The ART values were obtained from AREVA calculation 32-9019240-000, "ART Values for Point Beach Units 1 and 2" [Reference 4]. This document makes use of the NRC approved Topical Report BAW-2308, Revision 1-A, "Initial RTNDT of Linde 80 Weld Materials" [Reference 5]. The fluence projections are based on the Ferret Code best-estimate values verified by Westinghouse in letter WEP-06-13, dated February 14, 2006 [Reference 6] as approved for use by the NRC in WCAP-16083-NP-A, Revision 0 [Reference 7].

The purpose of this report is to present the calculations and the development of the Point Beach Units 1 and 2 heatup and cooldown curves for 43 and 53 EFPY for plant conditions that reflect a power uprate and with/without removal of hafnium rods. The four cases analyzed for each unit are shown in Table 1-1. This report documents the calculated ART values and the development of the PT limit curves for normal operation. The PT curves herein were generated without instrumentation errors. The PT curves include a hydrostatic leak test limit curve from 2485 psig to 2000 psig, along with the pressure-temperature limits for the vessel flange region per the requirements of 10 CFR Part 50, Appendix G [Reference 8].

TABLE 1-1
Plant Operating Conditions Reflecting Development of PT Limit Curves for 43 and 53 EFPY

Case	Unit	Power (MWth)	EFPY	Hafnium Rods?
1	1	1518.5 startup to 2/3/2003 1540.0 2/3/2003 to 10/2008 1678.0 10/2008 to 53 EFPY	53	Yes
	2	1518.5 startup to 2/3/2003 1540.0 2/3/2003 to 4/2008 1678.0 4/2008 to 53 EFPY	53	Yes
2	1	1518.5 startup to 2/3/2003 1540.0 2/3/2003 to 10/2008 1678.0 10/2008 to 43 EFPY	43	Yes
	2	1518.5 startup to 2/3/2003 1540.0 2/3/2003 to 4/2008 1678.0 4/2008 to 43 EFPY	43	Yes
3	1	1518.5 startup to 2/3/2003 1540.0 2/3/2003 to 10/2008 1678.0 10/2008 to 53 EFPY	53	Removal October 2008
	2	1518.5 startup to 2/3/2003 1540.0 2/3/2003 to 4/2008 1678.0 4/2008 to 53 EFPY	53	Removal April 2008
4	1	1518.5 startup to 2/3/2003 1540.0 2/3/2003 to 10/2008 1678.0 10/2008 to 43 EFPY	43	Removal October 2008
	2	1518.5 startup to 2/3/2003 1540.0 2/3/2003 to 4/2008 1678.0 4/2008 to 43 EFPY	43	Removal April 2008

2 FRACTURE TOUGHNESS PROPERTIES

The fracture-toughness properties of the ferritic materials in the reactor coolant pressure boundary are determined in accordance with the NRC Standard Review Plan [Reference 9]. The beltline material properties of the Point Beach Units 1 and 2 reactor vessel are presented in Table 2-1. The unirradiated RT_{NDT} values for the closure head and vessel flange are documented in Table 2-2.

The Regulatory Guide 1.99, Revision 2 methodology used to develop the heatup and cooldown curves documented in this report is the same as that documented in WCAP-14040, Revision 4 [Reference 2]. The chemistry factors (CFs) were calculated using Regulatory Guide 1.99 Revision 2, Positions 1.1 and 2.1. Position 1.1 uses the Tables from the Reg. Guide along with the best estimate copper and nickel weight percents, which are presented in Table 2-1. Position 2.1 uses the surveillance capsule data from all capsules withdrawn to date. The determination of CFs using Positions 1.1 and 2.1 are documented in Reference 4. Table 2-3 summarizes the Positions 1.1 and 2.1 CFs determined for the Point Beach Units 1 and 2 beltline materials.

It should be noted that in the calculations of Position 2.1 chemistry factors in Table 2-3, the ratio procedure described in Reference 1 was applied to account for chemistry differences between the vessel weld material and the surveillance weld material [Reference 4]. No temperature adjustments are required for the Point Beach Units 1 and 2 data since it is being applied to their own plants.

TABLE 2-1
Summary of the Best Estimate Cu and Ni Weight Percent and Initial RT_{NDT} Values for the Point Beach Units 1 and 2 Reactor Vessel Materials [Reference 4]

Material Description				Chemical Composition		Initial RT_{NDT}
Reactor Vessel Beltline Region Location	Matl. Ident.	Heat Number	Type	Cu wt%	Ni wt%	
Point Beach Unit 1						
Nozzle Belt Forging (NB)	122P237	122P237	SA-508 Cl. 2	0.11	0.82	50°F
Intermediate Shell Plate (IS)	A9811-1	A9811-1	SA-508 Cl. 2	0.20	0.06	1°F
Lower Shell Plate (LS)	C1423-1	C1423-1	SA-508 Cl. 2	0.12	0.07	1°F
NB to IS Circ. Weld (100%)	SA-1426	8T1762	Linde 80 Flux	0.19	0.57	-47.6°F
IS Long. Weld (ID 27%)	SA-812	1P0815	Linde 80 Flux	0.17	0.52	-47.6°F
IS Long. Weld (OD 73%)	SA-775	1P0661	Linde 80 Flux	0.17	0.64	-47.6°F
Intermediate to LS Circ. Weld (100%)	SA-1101	71249	Linde 80 Flux	0.23	0.59	-47.4°F
LS Long. Weld (100%)	SA-847	61782	Linde 80 Flux	0.23	0.52	-47.6°F
Point Beach Unit 2						
Nozzle Belt Forging (NB)	123V352	123V352	SA-508 Cl. 2	0.11	0.73	40°F
Intermediate Shell Forging (IS)	123V500	123V500	SA-508 Cl. 2	0.09	0.70	40°F
Lower Shell Forging (LS)	123W195	123W195	SA-508 Cl. 2	0.05	0.72	40°F
NB to IS Circ. Weld (100%)	21935	21935	Linde 1092 Flux	0.18	0.70	-56°F
Intermediate to LS Circ. Weld (100%)	SA-1484	72442	Linde 80 Flux	0.26	0.60	-30°F

TABLE 2-2
Summary of the Initial RT_{NDT} Values for the Point Beach Units 1 and 2 Closure Head and Vessel Flanges

Material Identification	Initial RT_{NDT} (°F)
Point Beach Unit 1	
Closure Head Flange	-68[(a)]
Vessel Flange	48[(b)]
Point Beach Unit 2	
Closure Head Flange	-60[(a)]
Vessel Flange	60[(b)]

Notes:
(a) Certified Material Test Report Initial RT_{NDT} values documented in References 12 and 13 for the replacement reactor vessel closure heads.
(b) From WCAP-15121, Revision 1 [Reference 10]

TABLE 2-3
Summary of the Point Beach Units 1 and 2 Reactor Vessel Beltline Material Chemistry Factors [Reference 4]

Beltline Materials	Unit	Material Ident.	Cu wt%	Ni wt%	Chemistry Factor
Nozzle Belt Forging (NB)	1	122P237	0.11	0.82	77.0
Intermediate Shell (IS)	1	A9811-1	0.20	0.06	79.3[(a)]
Lower Shell (LS)	1	C1423-1	0.12	0.07	35.8[(a)]
NB to IS Circ. Weld (100%)	1	SA-1426	0.19	0.57	167.0
IS Long. Weld (ID 27%)	1	SA-812	0.17	0.52	167.0
IS Long. Weld (OD 73%)	1	SA-775	0.17	0.64	167.0
Intermediate to LS Circ. Weld (100%)	1	SA-1101	0.23	0.59	167.6
LS Long. Weld (100%)	1	SA-847	0.23	0.52	167.0
Nozzle Belt Forging (NB)	2	123V352	0.11	0.73	76.0
Intermediate Shell Forging (IS)	2	123V500	0.09	0.70	58.0
Lower Shell Forging (LS)	2	123W195	0.05	0.72	42.8[(a)]
NB to IS Circ. Weld (100%)	2	21935	0.18	0.70	170.5
Intermediate to LS Circ. Weld (100%)	2	SA-1484	0.26	0.60	180.0

Notes:
(a) Determined from surveillance data

3 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

3.1 OVERALL APPROACH

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{Ic}, for the metal temperature at that time. K_{Ic} is obtained from the reference fracture toughness curve, defined in the 1998 Edition through the 2000 Addenda of Section XI, Appendix G of the ASME Code [Reference 3]. The K_{Ic} curve is given by the following equation:

$$K_{Ic} = 33.2 + 20.734 * e^{[0.02(T - RT_{NDT})]} \tag{1}$$

where,

K_{Ic} = reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature RT_{NDT}

This K_{Ic} curve is based on the lower bound of static critical K_I values measured as a function of temperature on specimens of SA-533 Grade B Class_1, SA-508-1, SA-508-2, SA-508-3 steel.

3.2 METHODOLOGY FOR PRESSURE-TEMPERATURE LIMIT CURVE DEVELOPMENT

The governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C^* K_{Im} + K_{It} < K_{Ic} \tag{2}$$

where,

K_{Im} = stress intensity factor caused by membrane (pressure) stress

K_{It} = stress intensity factor caused by the thermal gradients

K_{Ic} = function of temperature relative to the RT_{NDT} of the material

C = 2.0 for Level A and Level B service limits

C = 1.5 for hydrostatic and leak test conditions during which the reactor core is not critical

For membrane tension, the corresponding K_I for the postulated defect is:

$$K_{Im} = M_m \times (pR_i / t) \tag{3}$$

where, M_m for an inside surface flaw is given by:

M_m = 1.85 for $\sqrt{t} < 2$,
M_m = $0.926\sqrt{t}$ for $2 \le \sqrt{t} \le 3.464$,
M_m = 3.21 for $\sqrt{t} > 3.464$

Similarly, M_m for an outside surface flaw is given by:

M_m = 1.77 for $\sqrt{t} < 2$,
M_m = $0.893\sqrt{t}$ for $2 \le \sqrt{t} \le 3.464$,
M_m = 3.09 for $\sqrt{t} > 3.464$

and

p = internal pressure, Ri = vessel inner radius, and t = vessel wall thickness.

For bending stress, the corresponding K_I for the postulated defect is:

K_{Ib} = M_b * Maximum Stress, where M_b is two-thirds of M_m

The maximum K_I produced by radial thermal gradient for the postulated inside surface defect of G-2120 is $K_{It} = 0.953 \times 10^{-3} \times CR \times t^{2.5}$, where CR is the cooldown rate in °F/hr., or for a postulated outside surface defect, $K_{It} = 0.753 \times 10^{-3} \times HU \times t^{2.5}$, where HU is the heatup rate in °F/hr.

The through-wall temperature difference associated with the maximum thermal K_I can be determined from Fig. G-2214-1. The temperature at any radial distance from the vessel surface can be determined from Fig. G-2214-2 for the maximum thermal K_I.

(a) The maximum thermal K_I relationship and the temperature relationship in Fig. G-2214-1 are applicable only for the conditions given in G-2214.3(a)(1) and (2).

(b) Alternatively, the K_I for radial thermal gradient can be calculated for any thermal stress distribution and at any specified time during cooldown for a ¼-thickness inside surface defect using the relationship:

$$K_{It} = (1.0359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3) * \sqrt{\pi a} \tag{4}$$

or similarly, K_{IT} during heatup for a ¼-thickness outside surface defect using the relationship:

$$K_{It} = (1.043C_0 + 0.630C_1 + 0.481C_2 + 0.401C_3) * \sqrt{\pi a} \quad (5)$$

where the coefficients C_0, C_1, C_2 and C_3 are determined from the thermal stress distribution at any specified time during the heatup or cooldown using the form:

$$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3 \quad (6)$$

and x is a variable that represents the radial distance from the appropriate (i.e., inside or outside) surface to any point on the crack front and a is the maximum crack depth.

Note, that Equations 3, 4 and 5 were implemented in the OPERLIM computer code, which is the program used to generate the pressure-temperature (PT) limit curves. No other changes were made to the OPERLIM computer code with regard to PT calculation methodology. Therefore, the PT curve methodology is unchanged from that described in WCAP-14040-NP-A, Revision 4 "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" [Reference 2] Section 2.6 (equations 2.6.2-4 and 2.6.3-1) with the exceptions just described above.

At any time during the heatup or cooldown transient, K_{Ic} is determined by the metal temperature at the tip of a postulated flaw at the 1/4T and 3/4T location, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from the temperature gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, K_{It}, for the reference flaw are computed. From Equation 2, the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

For the calculation of the allowable pressure versus coolant temperature during cooldown, the reference flaw of Appendix G to the ASME Code is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on the measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel inner diameter. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the ΔT (temperature) developed during cooldown results in a higher value of K_{Ic} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist so that the increase in K_{Ic} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location and, therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and ensures conservative operation of the system for the entire cooldown period.

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the wall. The heatup results in compressive stresses at the inside surface that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{Ic} for the 1/4T crack during heatup is lower than the K_{Ic} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist so that the effects of compressive thermal stresses and lower K_{Ic} values do not offset each other, and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to ensure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of the pressure-temperature limitations for the case in which a 1/4T flaw located at the 1/4T location from the outside surface is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and therefore tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced by constructing a composite curve based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration. The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist wherein, over the course of the heatup ramp, the controlling condition switches from the inside to the outside, and the pressure limit must at all times be based on analysis of the most critical criterion.

3.3 CLOSURE HEAD/VESSEL FLANGE REQUIREMENTS

10 CFR Part 50, Appendix G [Reference 8] addresses the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure flange regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (3107 psig for Point Beach Unit 1 and 3125 psig for Point Beach Unit 2), which is calculated to be 621 psig (the more limiting value) for both Units 1 and 2. The limiting unirradiated RT_{NDT} of 60°F occurs in the vessel flange of the Point Beach Unit 2 reactor vessel, so the minimum allowable temperature of this region is 180°F at pressures greater than 621 psig (without instrument uncertainties). This limit is shown in Figures 5-1 through 5-8 wherever applicable.

4 CALCULATION OF ADJUSTED REFERENCE TEMPERATURE

From Regulatory Guide 1.99, Revision 2, the adjusted reference temperature (ART) for each material in the beltline region is given by the following expression:

$$\text{ART} = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \quad (7)$$

Initial RT_{NDT} is the reference temperature for the unirradiated material as defined in paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code [Reference 9]. If measured values of initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may be used if there are sufficient test results to establish a mean and standard deviation for the class.

ΔRT_{NDT} is the mean value of the adjustment in reference temperature caused by irradiation and should be calculated as follows:

$$\Delta RT_{NDT} = CF * f^{(0.28 - 0.10 \log f)} \quad (8)$$

To calculate ΔRT_{NDT} at any depth (e.g., at 1/4T or 3/4T), the following formula must first be used to attenuate the fluence at the specific depth.

$$f_{(depth\ x)} = f_{surface} * e^{(-0.24x)} \quad (9)$$

where x inches (vessel beltline thickness is 6.5 inches) is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then placed in Equation 8 to calculate the ΔRT_{NDT} at the specific depth.

The Westinghouse Radiation Engineering and Analysis Group evaluated the vessel fluence projections and the results are presented in Tables 4-1 through 4-4 [Reference 11]. The Ferret Code best estimate values used from Reference 11 are an exception to the methods presented in WCAP-14040-NP-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" [Reference 2]. Their use is permitted based on NRC approval documented in WCAP-16083-NP-A, Revision 0 [Reference 7] and has been verified by Westinghouse in letter WEP-06-13 [Reference 6]. Tables 4-1 through 4-4 also provide a summary of the vessel fluence projections at the 1/4T and 3/4T locations. Tables 4-5 and 4-8 contain the 1/4T and 3/4T calculated fluences and fluence factors, per the Regulatory Guide 1.99, Revision 2, used to calculate the 43 and 53 EFPY ART values for all beltline materials in the Point Beach Units 1 and 2 reactor vessels.

Margin is calculated as, $M = 2\sqrt{\sigma_i^2 + \sigma_\Delta^2}$. The standard deviation for the initial RT_{NDT} margin term (σ_i) is 0°F when the initial RT_{NDT} is a measured value and 17°F when a generic value is available. The standard deviation for the ΔRT_{NDT} margin term, σ_Δ, is 17°F for plates or forgings, and 8.5°F for plates or forgings when credible surveillance data is used. For welds, σ_Δ is equal to 28°F when surveillance capsule data is not used, and is 14°F (half the value) when credible surveillance capsule data is used. σ_Δ need not exceed 0.5 times the mean value of ΔRT_{NDT}.

Contained in Tables 4-5 through 4-8 are the Point Beach (Units 1 and 2) 43 and 53 EFPY ART calculations with and without hafnium removal used for generation of the heatup and cooldown curves.

Table 4-1
Fluence (E>1.0 MeV) Values for the Point Beach Unit 1 and Unit 2 Vessel Beltline Materials with Uprate, with hafnium rods, through 53 EFPY [Reference 4]

Beltline Materials	Unit	Material Ident.	53 EFPY Fluence, n/cm^2		
			Inner Wetted Surface	1/4T Location (x=1.625)	3/4T Location (x=4.875)
Nozzle Belt Forging (NB)	1	122P237	2.84E+18	1.92E+18	8.81E+17
Intermediate Shell (IS)	1	A9811-1	4.86E+19	3.29E+19	1.51E+19
Lower Shell (LS)	1	C1423-1	3.70E+19	2.51E+19	1.15E+19
NB to IS Circ. Weld (100%)	1	SA-1426	2.84E+18	1.92E+18	8.81E+17
IS Long. Weld (ID 27%)	1	SA-812	3.10E+19	2.10E+19	N/A
IS Long. Weld (OD 73%)	1	SA-775	3.10E+19	N/A	9.62E+18
Intermediate to LS Circ. Weld (100%)	1	SA-1101	3.71E+19	2.51E+19	1.15E+19
LS Long. Weld (100%)	1	SA-847	2.60E+19	1.76E+19	8.07E+18
Nozzle Belt Forging (NB)	2	123V352	4.12E+18	2.79E+18	1.28E+18
Intermediate Shell Forging (IS)	2	123V500	4.66E+19	3.16E+19	1.44E+19
Lower Shell Forging (LS)	2	123W195	4.15E+19	2.81E+19	1.29E+19
NB to IS Circ. Weld (100%)	2	21935	4.12E+18	2.79E+18	1.28E+18
Intermediate to LS Circ. Weld (100%)	2	SA-1484	3.75E+19	2.54E+19	1.16E+19

Table 4-2
Fluence (E>1.0 MeV) Values for the Point Beach Unit 1 and Unit 2 Vessel Beltline Materials with Uprate, with hafnium rods, through 43 EFPY [Reference 4]

Beltline Materials	Unit	Material Ident.	43 EFPY Fluence, n/cm²		
			Inner Wetted Surface	1/4T Location (x=1.625)	3/4T Location (x=4.875)
Nozzle Belt Forging (NB)	1	122P237	2.38E+18	1.61E+18	7.39E+17
Intermediate Shell (IS)	1	A9811-1	3.95E+19	2.67E+19	1.23E+19
Lower Shell (LS)	1	C1423-1	3.11E+19	2.11E+19	9.65E+18
NB to IS Circ. Weld (100%)	1	SA-1426	2.38E+18	1.61E+18	7.39E+17
IS Long. Weld (ID 27%)	1	SA-812	2.52E+19	1.71E+19	N/A
IS Long. Weld (OD 73%)	1	SA-775	2.52E+19	N/A	7.82E+18
Intermediate to LS Circ. Weld (100%)	1	SA-1101	3.10E+19	2.10E+19	9.62E+18
LS Long. Weld (100%)	1	SA-847	2.15E+19	1.46E+19	6.67E+18
Nozzle Belt Forging (NB)	2	123V352	3.42E+18	2.32E+18	1.06E+18
Intermediate Shell Forging (IS)	2	123V500	3.81E+19	2.58E+19	1.18E+19
Lower Shell Forging (LS)	2	123W195	3.45E+19	2.34E+19	1.07E+19
NB to IS Circ. Weld (100%)	2	21935	3.42E+18	2.32E+18	1.06E+18
Intermediate to LS Circ. Weld (100%)	2	SA-1484	3.14E+19	2.13E+19	9.75E+18

Table 4-3
Fluence (E>1.0 MeV) Values for the Point Beach Unit 1 and Unit 2 Vessel Beltline Materials with Uprate, with hafnium rod removal October 2008 (Unit 1) and April 2008 (Unit 2), through 53 EFPY [Reference 4]

Beltline Materials	Unit	Material Ident.	53 EFPY Fluence, n/cm^2 Inner Wetted Surface	1/4T Location (x=1.625)	3/4T Location (x=4.875)
Nozzle Belt Forging (NB)	1	122P237	3.58E+18	2.42E+18	1.11E+18
Intermediate Shell (IS)	1	A9811-1	4.90E+19	3.32E+19	1.52E+19
Lower Shell (LS)	1	C1423-1	4.55E+19	3.08E+19	1.41E+19
NB to IS Circ. Weld (100%)	1	SA-1426	3.58E+18	2.42E+18	1.11E+18
IS Long. Weld (ID 27%)	1	SA-812	3.19E+19	2.16E+19	N/A
IS Long. Weld (OD 73%)	1	SA-775	3.19E+19	N/A	9.90E+18
Intermediate to LS Circ. Weld (100%)	1	SA-1101	4.43E+19	3.00E+19	1.38E+19
LS Long. Weld (100%)	1	SA-847	3.05E+19	2.07E+19	9.47E+18
Nozzle Belt Forging (NB)	2	123V352	5.04E+18	3.41E+18	1.56E+18
Intermediate Shell Forging (IS)	2	123V500	5.05E+19	3.42E+19	1.57E+19
Lower Shell Forging (LS)	2	123W195	4.90E+19	3.32E+19	1.52E+19
NB to IS Circ. Weld (100%)	2	21935	5.04E+18	3.41E+18	1.56E+18
Intermediate to LS Circ. Weld (100%)	2	SA-1484	4.65E+19	3.15E+19	1.44E+19

Table 4-4
Fluence (E>1.0 MeV) Values for the Point Beach Unit 1 and Unit 2 Vessel Beltline Materials with Uprate, with hafnium rod removal October 2008 (Unit 1) and April 2008 (Unit 2), through 43 EFPY [Reference 4]

Beltline Materials	Unit	Material Ident.	43 EFPY Fluence, n/cm^2		
			Inner Wetted Surface	1/4T Location (x=1.625)	3/4T Location (x=4.875)
Nozzle Belt Forging (NB)	1	122P237	2.80E+18	1.90E+18	8.69E+17
Intermediate Shell (IS)	1	A9811-1	3.97E+19	2.69E+19	1.23E+19
Lower Shell (LS)	1	C1423-1	3.59E+19	2.43E+19	1.11E+19
NB to IS Circ. Weld (100%)	1	SA-1426	2.80E+18	1.90E+18	8.69E+17
IS Long. Weld (ID 27%)	1	SA-812	2.57E+19	1.74E+19	N/A
IS Long. Weld (OD 73%)	1	SA-775	2.57E+19	N/A	7.98E+18
Intermediate to LS Circ. Weld (100%)	1	SA-1101	3.51E+19	2.38E+19	1.09E+19
LS Long. Weld (100%)	1	SA-847	2.42E+19	1.64E+19	7.51E+18
Nozzle Belt Forging (NB)	2	123V352	3.95E+18	2.67E+18	1.23E+18
Intermediate Shell Forging (IS)	2	123V500	4.04E+19	2.74E+19	1.25E+19
Lower Shell Forging (LS)	2	123W195	3.88E+19	2.63E+19	1.20E+19
NB to IS Circ. Weld (100%)	2	21935	3.95E+18	2.67E+18	1.23E+18
Intermediate to LS Circ. Weld (100%)	2	SA-1484	3.67E+19	2.49E+19	1.14E+19

Table 4-5
Adjusted Reference Temperature Evaluation for the Point Beach Unit 1 and Unit 2 Reactor Vessel Beltline Materials with Uprate, with hafnium rods, through 53 EFPY [Reference 4]

Material Description			Initial RT_{NDT}	Chemistry Factor	53 EFPY Fluence, n/cm^2			ΔRT_{NDT}, °F at 53 EFPY		Margin		ART, °F at 53 EFPY	
Reactor Vessel Beltline Region Location	Matl. Ident.	Heat Number			Clad/Low Alloy Steel Interface	1/4T	3/4T	1/4T	3/4T	1/4T	3/4T	1/4T	3/4T
Point Beach Unit 1 Evaluation													
Nozzle Belt Forging (NB)	122P237	122P237	50	77.0	2.84E+18	1.92E+18	8.81E+17	43.1	30.2	34.0	30.2	127.1	110.4
Intermediate Shell Plate (IS)	A9811-1	A9811-1	1	79.3*	4.86E+19	3.29E+19	1.51E+19	104.0	88.3	56.4	56.4	161.4	145.7
Lower Shell Plate (LS)	C1423-1	C1423-1	1	35.8*	3.70E+19	2.51E+19	1.15E+19	44.6	37.2	56.4	56.4	102.0	94.6
NB to IS Circ. Weld (100%)	SA-1426	8T1762	-47.6	167.0	2.84E+18	1.92E+18	8.81E+17	93.5	65.5	65.7	65.7	111.6	83.6
IS Long. Weld (ID 27%)	SA-812	1P0815	-47.6	167.0	3.10E+19	2.10E+19	N/A	200.7	N/A	65.7	N/A	218.8	N/A
IS Long. Weld (OD 73%)	SA-775	1P0661	-47.6	167.0	3.10E+19	N/A	9.62E+18	N/A	165.2	N/A	65.7	N/A	183.3
Intermediate to LS Circ. Weld (100%)	SA-1101	71249	-47.4	167.6	3.71E+19	2.51E+19	1.15E+19	209.1	174.1	61.7	61.7	223.4	188.4
LS Long. Weld (100%)	SA-847	61782	-47.6	167.0	2.60E+19	1.76E+19	8.07E+18	192.9	157.0	65.7	65.7	211.1	175.3
Point Beach Unit 2 Evaluation													
Nozzle Belt Forging (NB)	123V352	123V352	40	76.0	4.12E+18	2.79E+18	1.28E+18	49.6	35.6	34.0	34.0	123.6	109.6
Intermediate Shell Forging (IS)	123V500	123V500	40	58.0	4.66E+19	3.16E+19	1.45E+19	75.6	63.9	34.0	34.0	149.6	137.9
Lower Shell Forging (LS)	123W195	123W195	40	42.8*	4.15E+19	2.81E+19	1.29E+19	54.6	45.8	17.0	17.0	111.6	102.8
NB to IS Circ. Weld (100%)	21935	21935	-56	170.5	4.12E+18	2.79E+18	1.29E+18	111.2	79.8	65.5	65.5	120.7	89.3
Intermediate to LS Circ. Weld (100%)	SA-1484	72442	-30	180.0	3.75E+19	2.54E+19	1.16E+19	225.0	187.6	60.8	60.8	255.8	218.4

* - Determined from surveillance data.

Table 4-6
Adjusted Reference Temperature Evaluation for the Point Beach Unit 1 and Unit 2 Reactor Vessel Beltline Materials with Uprate, with hafnium rods, through 43 EFPY [Reference 4]

Material Description			Initial RT_{NDT}	Chemistry Factor	43 EFPY Fluence, n/cm^2			ΔRT_{NDT}, °F at 43 EFPY		Margin		ART, °F at 43 EFPY	
Reactor Vessel Beltline Region Location	Matl. Ident.	Heat Number			Clad/Low Alloy Steel Interface	1/4T	3/4T	1/4T	3/4T	1/4T	3/4T	1/4T	3/4T
Point Beach Unit 1 Evaluation													
Nozzle Belt Forging (NB)	122P237	122P237	50	77.0	2.38E+18	1.61E+18	7.39E+17	40.0	27.6	34.0	27.6	124.0	105.2
Intermediate Shell Plate (IS)	A9811-1	A9811-1	1	79.3*	3.95E+19	2.67E+19	1.23E+19	100.2	83.8	56.4	56.4	157.6	141.2
Lower Shell Plate (LS)	C1423-1	C1423-1	1	35.8*	3.11E+19	2.11E+19	9.65E+18	43.1	35.4	56.4	56.4	100.5	92.8
NB to IS Circ. Weld (100%)	SA-1426	8T1762	-47.6	167.0	2.38E+18	1.61E+18	7.40E+17	86.7	60.0	65.7	65.7	104.8	78.1
IS Long. Weld (ID 27%)	SA-812	1P0815	-47.6	167.0	2.52E+19	1.71E+19	N/A	191.6	N/A	65.7	N/A	209.7	N/A
IS Long. Weld (OD 73%)	SA-775	1P0661	-47.6	167.0	2.52E+19	N/A	7.82E+18	N/A	155.5	N/A	65.7	N/A	173.6
Intermediate to LS Circ. Weld (100%)	SA-1101	71249	-47.4	167.6	3.10E+19	2.10E+19	9.62E+18	201.5	165.8	61.7	61.7	215.8	180.1
LS Long. Weld (100%)	SA-847	61782	-47.6	167.0	2.15E+19	1.46E+19	6.67E+18	184.4	148.1	65.7	65.7	202.5	166.4
Point Beach Unit 2 Evaluation													
Nozzle Belt Forging (NB)	123V352	123V352	40	76.0	3.42E+18	2.32E+18	1.06E+18	46.0	32.6	34.0	32.6	120.0	105.2
Intermediate Shell Forging (IS)	123V500	123V500	40	58.0	3.81E+19	2.58E+19	1.18E+19	72.7	60.7	34.0	34.0	146.7	134.7
Lower Shell Forging (LS)	123W195	123W195	40	42.8*	3.45E+19	2.34E+19	1.07E+19	52.6	43.6	17.0	17.0	109.6	100.6
NB to IS Circ. Weld (100%)	21935	21935	-56	170.5	3.42E+18	2.32E+18	1.06E+18	103.2	73.1	65.5	65.5	112.7	82.6
Intermediate to LS Circ. Weld (100%)	SA-1484	72442	-30	180.0	3.14E+19	2.13E+19	9.75E+18	216.9	178.7	60.8	60.8	247.7	209.5

* - Determined from surveillance data.

Table 4-7
Adjusted Reference Temperature Evaluation for the Point Beach Unit 1 and Unit 2 Reactor Vessel Beltline Materials with Uprate, with hafnium removal October 2008 (Unit 1) and April 2008 (Unit 2), through 53 EFPY [Reference 4]

Material Description			Initial RT_{NDT}	Chemistry Factor	53 EFPY Fluence, n/cm²			ΔRT_{NDT}, °F at 53 EFPY		Margin		ART, °F at 53 EFPY	
Reactor Vessel Beltline Region Location	**Matl. Ident.**	**Heat Number**			**Clad/Low Alloy Steel Interface**	**1/4T**	**3/4T**	**1/4T**	**3/4T**	**1/4T**	**3/4T**	**1/4T**	**3/4T**
Point Beach Unit 1 Evaluation													
Nozzle Belt Forging (NB)	122P237	122P237	50	77.0	3.58E+18	2.42E+18	1.11E+18	47.4	33.7	34.0	33.7	131.4	117.4
Intermediate Shell Plate (IS)	A9811-1	A9811-1	1	79.3*	4.90E+19	3.32E+19	1.52E+19	104.1	88.5	56.4	56.4	161.5	145.9
Lower Shell Plate (LS)	C1423-1	C1423-1	1	35.8*	4.55E+19	3.09E+19	1.41E+19	46.4	39.2	56.4	56.4	103.8	96.6
NB to IS Circ. Weld (100%)	SA-1426	8T1762	-47.6	167.0	3.58E+18	2.42E+18	1.11E+18	102.9	73.2	65.7	65.7	121.0	91.3
IS Long. Weld (ID 27%)	SA-812	1P0815	-47.6	167.0	3.19E+19	2.16E+19	N/A	201.9	N/A	65.7	N/A	220.0	N/A
IS Long. Weld (OD 73%)	SA-775	1P0661	-47.6	167.0	3.19E+19	N/A	9.90E+18	N/A	166.5	N/A	65.7	N/A	184.6
Intermediate to LS Circ. Weld (100%)	SA-1101	71249	-47.4	167.6	4.43E+19	3.00E+19	1.38E+19	216.4	182.3	61.7	61.7	230.7	196.6
LS Long. Weld (100%)	SA-847	61782	-47.6	167.0	3.05E+19	2.07E+19	9.47E+18	199.9	164.5	65.7	65.7	218.1	182.7
Point Beach Unit 2 Evaluation													
Nozzle Belt Forging (NB)	123V352	123V352	40	76.0	5.04E+18	3.41E+18	1.56E+18	53.5	38.9	34.0	34.0	127.5	112.9
Intermediate Shell Forging (IS)	123V500	123V500	40	58.0	5.05E+19	3.42E+19	1.57E+19	76.6	65.2	34.0	34.0	150.6	139.2
Lower Shell Forging (LS)	123W195	123W195	40	42.8*	4.90E+19	3.32E+19	1.52E+19	56.2	47.8	17.0	17.0	113.2	104.8
NB to IS Circ. Weld (100%)	21935	21935	-56	170.5	5.04E+18	3.41E+18	1.56E+18	120.0	87.3	65.5	65.5	129.5	96.8
Intermediate to LS Circ. Weld (100%)	SA-1484	72442	-30	180.0	4.65E+19	3.15E+19	1.44E+19	234.4	198.4	60.8	60.8	265.2	229.2

* - Determined from surveillance data.

Table 4-8
Adjusted Reference Temperature Evaluation for the Point Beach Unit 1 and Unit 2 Reactor Vessel Beltline Materials with Uprate, with hafnium removal October 2008 (Unit 1) and April 2008 (Unit 2), through 43 EFPY [Reference 4]

Material Description			Initial RT_{NDT}	Chemistry Factor	43 EFPY Fluence, n/cm^2			ΔRT_{NDT}, °F at 43 EFPY		Margin		ART, °F at 43 EFPY	
Reactor Vessel Beltline Region Location	Matl. Ident.	Heat Number			Clad/Low Alloy Steel Interface	1/4T	3/4T	1/4T	3/4T	1/4T	3/4T	1/4T	3/4T
Point Beach Unit 1 Evaluation													
Nozzle Belt Forging (NB)	122P237	122P237	50	77.0	2.80E+18	1.90E+18	8.69E+17	42.9	30.0	34.0	30.0	126.9	110.0
Intermediate Shell Plate (IS)	A9811-1	A9811-1	1	79.3*	3.97E+19	2.69E+19	1.23E+19	100.2	83.9	56.4	56.4	157.6	141.3
Lower Shell Plate (LS)	C1423-1	C1423-1	1	35.8*	3.59E+19	2.43E+19	1.11E+19	44.4	36.9	56.4	56.4	101.8	94.3
NB to IS Circ. Weld (100%)	SA-1426	8T1762	-47.6	167.0	2.80E+18	1.890E+18	8.69E+17	93.0	65.0	65.7	65.7	111.1	83.1
IS Long. Weld (ID 27%)	SA-812	1P0815	-47.6	167.0	2.57E+19	1.74E+19	N/A	192.4	N/A	65.7	N/A	210.5	N/A
IS Long. Weld (OD 73%)	SA-775	1P0661	-47.6	167.0	2.57E+19	N/A	7.98E+18	N/A	156.5	N/A	65.7	N/A	174.6
Intermediate to LS Circ. Weld (100%)	SA-1101	71249	-47.4	167.6	3.51E+19	2.38E+19	1.09E+19	206.7	171.6	61.7	61.7	221.0	185.9
LS Long. Weld (100%)	SA-847	61782	-47.6	167.0	2.42E+19	1.64E+19	7.51E+18	189.7	153.6	65.7	65.7	207.8	171.9
Point Beach Unit 2 Evaluation													
Nozzle Belt Forging (NB)	123V352	123V352	40	76.0	3.95E+18	2.67E+18	1.23E+18	48.7	34.9	34.0	34.0	122.7	108.9
Intermediate Shell Forging (IS)	123V500	123V500	40	58.0	4.04E+19	2.74E+19	1.25E+19	73.5	61.7	34.0	34.0	147.5	135.7
Lower Shell Forging (LS)	123W195	123W195	40	42.8*	3.88E+19	2.63E+19	1.20E+19	53.9	45.0	17.0	17.0	110.9	102.0
NB to IS Circ. Weld (100%)	21935	21935	-56	170.5	3.95E+18	2.67E+18	1.23E+18	109.3	78.3	65.5	65.5	118.8	87.8
Intermediate to LS Circ. Weld (100%)	SA-1484	72442	-30	180.0	3.67E+19	2.49E+19	1.14E+19	224.1	186.5	60.8	60.8	254.9	217.3

*** - Determined from surveillance data.**

Contained in Tables 4-9 and 4-10 is a summary of the limiting ART values used in the generation of the Point Beach Units 1 and 2 reactor vessel PT limit curves. The limiting materials for the "axial-flaw" methodology are Point Beach Unit 1 Intermediate Shell Axial Welds SA-812 (1/4T location) and SA-775 (3/4T location). The limiting material for the "circ-flaw" methodology is Point Beach Unit 2 Intermediate to Lower Shell Circumferential Weld SA-1484.

TABLE 4-9
Summary of the Limiting ART Values (with Hafnium) Used in the Generation of the Point Beach Units 1 and 2 Heatup/Cooldown Curves

EFPY	Limiting "Circ-Flaw" ART		Limiting "Axial-Flaw" ART	
	1/4T (°F)	3/4T (°F)	1/4T (°F)	3/4T (°F)
Point Beach Unit 1				
43	215.8	180.1	**209.7**	**173.6**
53	223.4	188.4	**218.8**	**183.3**
Point Beach Unit 2				
43	**247.7**	**209.5**	146.7	134.7
53	**255.8**	**218.4**	149.6	137.9

TABLE 4-10
Summary of the Limiting ART Values (without Hafnium) Used in the Generation of the Point Beach Units 1 and 2 Heatup/Cooldown Curves

EFPY	Limiting "Circ-Flaw" ART		Limiting "Axial-Flaw" ART	
	1/4T (°F)	3/4T (°F)	1/4T (°F)	3/4T (°F)
Point Beach Unit 1				
43	221.0	185.9	**210.5**	**174.6**
53	230.7	196.6	**220.0**	**184.6**
Point Beach Unit 2				
43	**254.9**	**217.3**	147.5	135.7
53	**265.2**	**229.2**	150.6	139.2

5 HEATUP AND COOLDOWN PRESSURE-TEMPERATURE LIMIT CURVES

Pressure-temperature limit curves for normal heatup and cooldown of the primary reactor coolant system have been calculated for the pressure and temperature in the reactor vessel beltline region using the methods discussed in Sections 3 and 4 of this report. This approved methodology is also presented in WCAP-14040-NP-A, Revision 4.

Figure 5-1 presents the limiting heatup curves without margins for possible instrumentation errors using heatup rates of 60 and 100°F/hr applicable for 43 EFPY with the "Flange-Notch" requirement and hafnium and using the "Axial-flaw" methodology. This curve was generated using the 1998 ASME Code Section XI, Appendix G. Figure 5-2 presents the limiting cooldown curve without margins for possible instrumentation errors using cooldown rates of 0, 20, 40, 60 and 100°F/hr applicable for 43 EFPY with the "Flange-Notch" requirement and hafnium. Again, this curve was generated using the 1998 ASME Code Section XI, Appendix G. These PT limit curves bound those generated using the "Circ-flaw" methodology with the limiting circ-weld ART value from the Unit 2 intermediate to lower shell girth weld.

Figure 5-3 presents the limiting heatup curve without margins for possible instrumentation errors using heatup rates of 60 and 100°F/hr applicable for 53 EFPY with the "Flange-Notch" requirement and hafnium. This curve was generated using the 1998 ASME Code Section XI, Appendix G. Figure 5-4 presents the limiting cooldown curve without margins for possible instrumentation errors using cooldown rates of 0, 20, 40, 60 and 100°F/hr applicable for 53 EFPY with the "Flange-Notch" requirement and hafnium. Again, this curve was generated using the 1998 ASME Code Section XI, Appendix G. These PT limit curves bound those generated using the "Circ-flaw" methodology with the limiting circ-weld ART value from the Unit 2 intermediate to lower shell girth weld.

Figure 5-5 presents the limiting heatup curves without margins for possible instrumentation errors using heatup rates of 60 and 100°F/hr applicable for 43 EFPY with the "Flange-Notch" requirement and hafnium removal in 2008. This curve was generated using the 1998 ASME Code Section XI, Appendix G. Figure 5-6 presents the limiting cooldown curve without margins for possible instrumentation errors using cooldown rates of 0, 20, 40, 60 and 100°F/hr applicable for 43 EFPY with the "Flange-Notch" requirement and hafnium removal in 2008. Again, this curve was generated using the 1998 ASME Code Section XI, Appendix G. These PT limit curves bound those generated using the "Circ-flaw" methodology with the limiting circ-weld ART value from the Unit 2 intermediate to lower shell girth weld.

Figure 5-7 presents the limiting heatup curve without margins for possible instrumentation errors using heatup rates of 60 and 100°F/hr applicable for 53 EFPY with the "Flange-Notch" requirement and hafnium removal in 2008. This curve was generated using the 1998 ASME Code Section XI, Appendix G. Figure 5-8 presents the limiting cooldown curve without margins for possible instrumentation errors using cooldown rates of 0, 20, 40, 60 and 100°F/hr applicable for 53 EFPY with the "Flange-Notch" requirement and hafnium removal in 2008. Again, this curve was generated using the 1998 ASME Code Section XI, Appendix G. These PT limit curves bound those generated using the "Circ-flaw"

methodology with the limiting circ-weld ART value from the Unit 2 intermediate to lower shell girth weld.

Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown in Figures 5-1 through 5-8. This is in addition to other criteria, which must be met before the reactor is made critical, as discussed below in the following paragraphs.

The reactor must not be made critical until pressure-temperature combinations are to the right of the criticality limit line shown in Figures 5-1 and 5-7 (heatup curves only). The straight-line portion of the criticality limit is at the minimum permissible temperature for the 2485 psig inservice hydrostatic test as required by Appendix G to 10 CFR Part 50. The governing equation for the hydrostatic test is defined in the 1998 ASME Code Section XI, Appendix G as follows:

$$1.5\ K_{Im} < K_{Ic}$$

where,

K_{Im} is the stress intensity factor covered by membrane (pressure) stress,

$K_{Ic} = 33.2 + 20.734\ e^{[0.02\ (T - RT_{NDT})]}$,

T is the minimum permissible metal temperature, and

RT_{NDT} is the metal reference nil-ductility temperature.

The criticality limit curve specifies pressure-temperature limits for core operation to provide additional margin during actual power production as specified in Reference 8. The pressure-temperature limits for core operation (except for low power physics tests) are that: 1) the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and 2) the reactor vessel must be at least 40°F higher than the minimum permissible temperature in the corresponding pressure-temperature curve for heatup and cooldown calculated as described in Section 4 of this report. For the heatup and cooldown curves without margins for instrumentation errors, the minimum temperatures for the in service hydrostatic leak tests for the Point Beach Units 1 and 2 reactor vessel at 43 EFPY (with hafnium) is 260°F, at 43 EFPY (without hafnium) is 261°F, at 53 EFPY (with hafnium) is 269°F, and at 53 EFPY (without hafnium) is 270°F. The vertical line drawn from these points on the pressure-temperature curve, intersecting a curve 40°F higher than the pressure-temperature limit curve constitutes the limit for core operation for the reactor vessel.

Figures 5-1 through 5-8 define all of the above limits for ensuring prevention of non-ductile failure for the Point Beach Units 1 and 2 reactor vessel for 43 and 53 EFPY with the "Flange-Notch" requirement, without instrumentation uncertainties, and with/without hafnium [Reference 8]. The data points used for developing the heatup and cooldown pressure-temperature limit curves shown in Figures 5-1 through 5-8 are presented in Tables 5-1 through 5-8..

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Intermediate Shell Longitudinal Welds SA-812 (ID) and SA-775 (OD)
LIMITING ART VALUES AT 43 EFPY (with Hafnium): 1/4T, 209.7°F
3/4T, 173.6°F

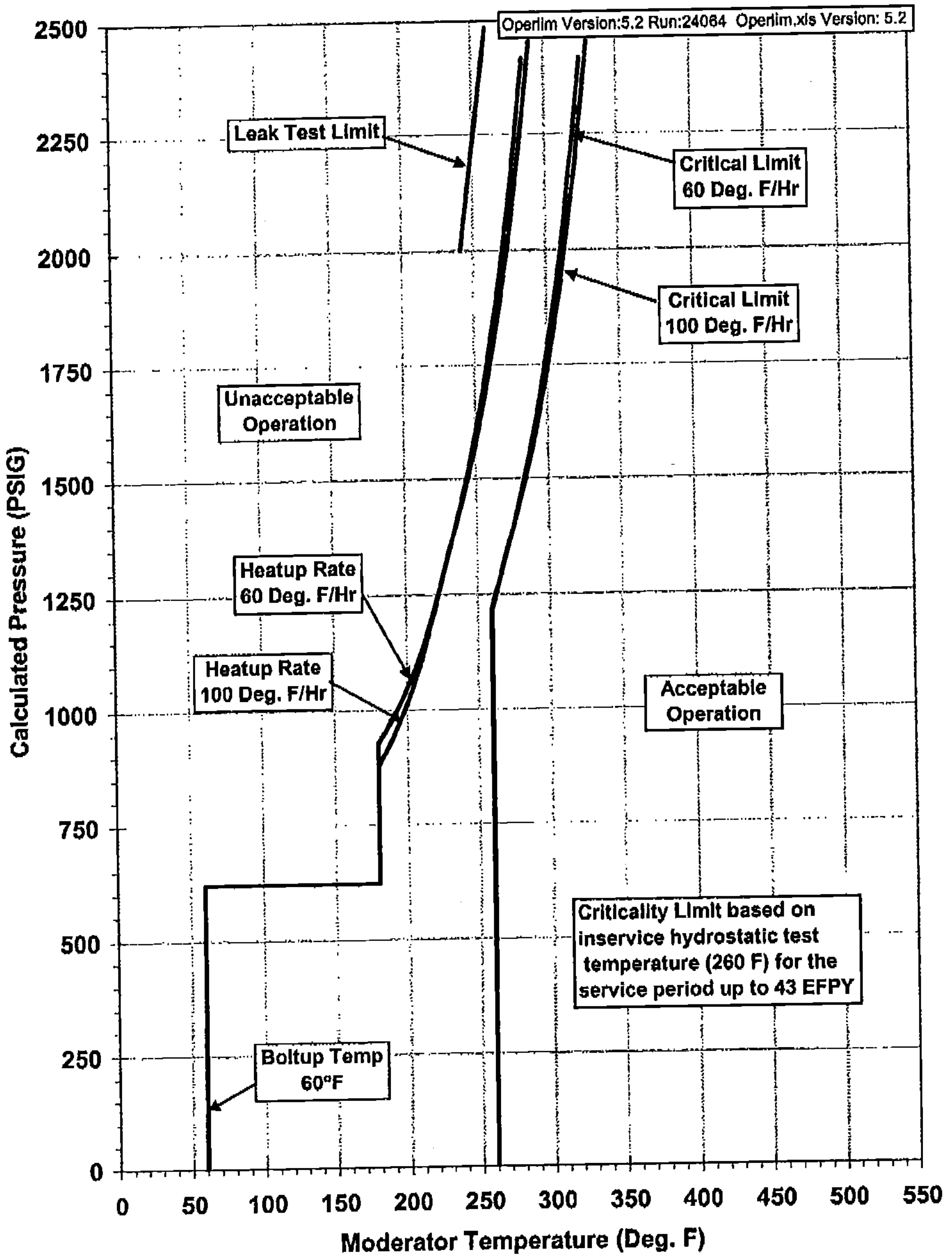

Figure 5-1 **Point Beach Units 1 and 2 Reactor Coolant System Heatup Limitations (Heatup Rates of 60 and 100°F/hr) Applicable for 43 EFPY (with Hafnium and without Margins for Instrumentation Errors) Using 1998 App. G Methodology (w/K_{Ic})**

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Intermediate Shell Longitudinal Welds SA-812 (ID) and SA-775 (OD)
LIMITING ART VALUES AT 43 EFPY (with Hafnium): 1/4T, 209.7°F
3/4T, 173.6°F

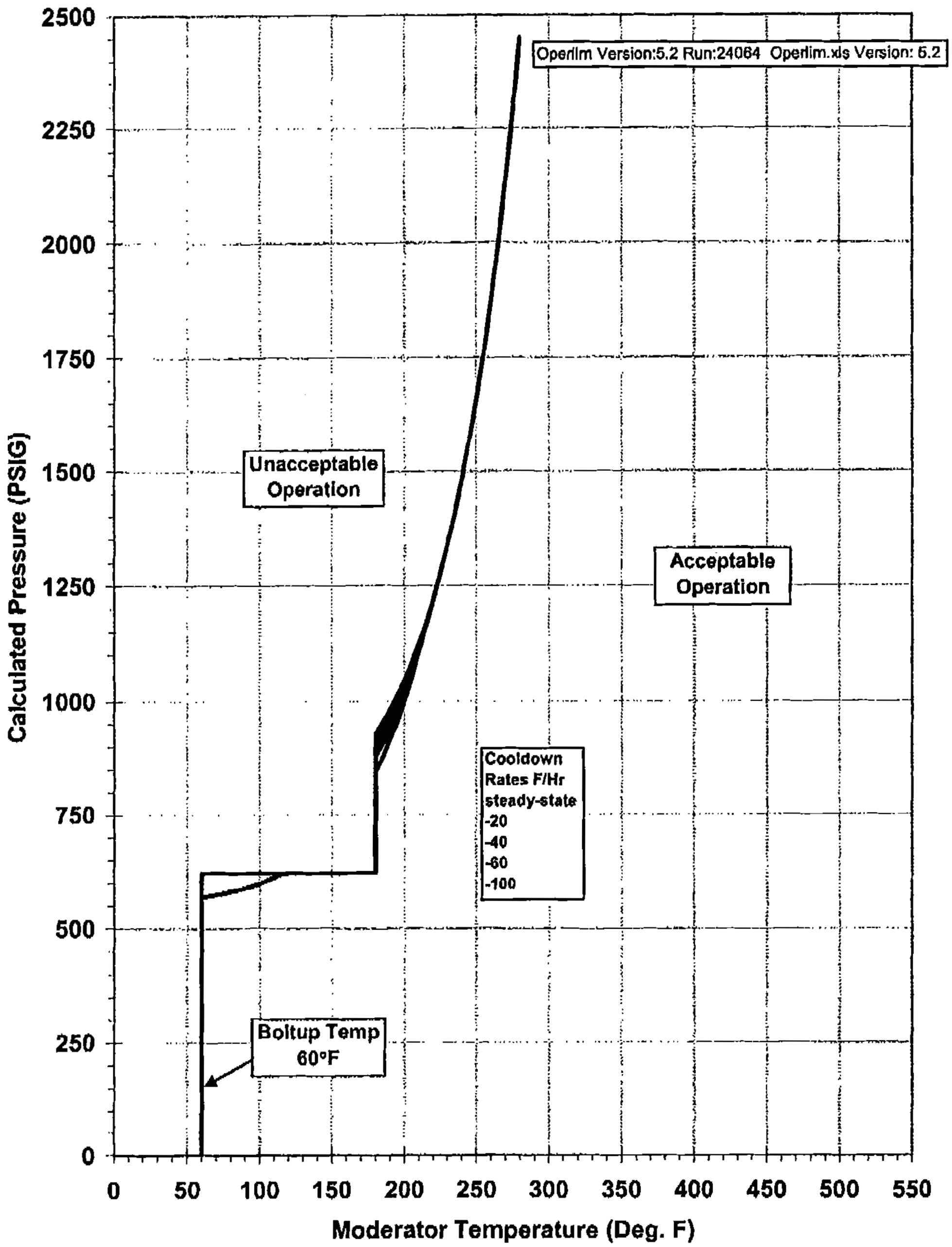

Figure 5-2 Point Beach Units 1 and 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for 43 EFPY (with Hafnium and without Margins for Instrumentation Errors) Using 1998 App. G Methodology (w/K_{Ic})

TABLE 5-1

43 EFPY Heatup Curve Data Points Using 1998 App. G Methodology

(w/Hafnium, w/K_{IC}, w/Flange Notch and w/o Uncertainties for Instrumentation Errors)

60°F/hr Heatup		Criticality Limit		100°F/hr Heatup		Criticality Limit	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
60	0	260	0	60	0	260	0
60	621	260	621	60	621	260	621
65	621	260	621	65	621	260	621
70	621	260	621	70	621	260	621
75	621	260	621	75	621	260	621
80	621	260	621	80	621	260	621
85	621	260	621	85	621	260	621
90	621	260	621	90	621	260	621
95	621	260	621	95	621	260	621
100	621	260	621	100	621	260	621
105	621	260	621	105	621	260	621
110	621	260	621	110	621	260	621
115	621	260	621	115	621	260	621
120	621	260	621	120	621	260	621
125	621	260	621	125	621	260	621
130	621	260	621	130	621	260	621
135	621	260	621	135	621	260	621
140	621	260	621	140	621	260	621
145	621	260	621	145	621	260	621
150	621	260	621	150	621	260	621
155	621	260	621	155	621	260	621
160	621	260	621	160	621	260	621
165	621	260	621	165	621	260	621
170	621	260	621	170	621	260	621
175	621	260	621	175	621	260	621
180	621	260	929	180	621	260	880
180	929	260	954	180	880	260	908
185	954	260	982	185	908	260	940
190	982	260	1012	190	940	260	975
195	1012	260	1046	195	975	260	1014
200	1046	260	1083	200	1014	260	1056
205	1083	260	1125	205	1056	260	1104
210	1125	260	1170	210	1104	260	1156
215	1170	260	1221	215	1156	260	1213
220	1221	265	1277	220	1213	265	1277
225	1277	270	1333	225	1277	270	1338
230	1333	275	1390	230	1338	275	1390
235	1390	280	1454	235	1390	280	1446
240	1454	285	1523	240	1446	285	1508
245	1523	290	1600	245	1508	290	1577
250	1600	295	1685	250	1577	295	1653
255	1685	300	1779	255	1653	300	1736
260	1779	305	1883	260	1736	305	1829

60°F/hr Heatup		Criticality Limit		100°F/hr Heatup		Criticality Limit	
T (°F)	**P (psig)**	**T (°F)**	**P (psig)**	**T (°F)**	**P (psig)**	**T (°F)**	**P (psig)**
265	1883	310	1997	265	1829	310	1930
270	1997	315	2123	270	1930	315	2043
275	2123	320	2263	275	2043	320	2167
280	2263	325	2417	280	2167	325	2303
285	2417			285	2303	330	2454
				290	2454		
Leak Test Limit		**Temperature (°F)**		241	260		
		Pressure (psig)		2000	2485		

TABLE 5-2
43 EFPY Cooldown Curve Data Points Using 1998 App. G Methodology
(w/Hafnium, w/K_{IC}, w/Flange Notch and w/o Uncertainties for Instrumentation Errors)

Steady State		20 °F/hr		40°F/hr		60°F/hr		100°F/hr	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
60	0	60	0	60	0	60	0	60	0
60	621	60	621	60	621	60	621	60	569
65	621	65	621	65	621	65	621	65	572
70	621	70	621	70	621	70	621	70	574
75	621	75	621	75	621	75	621	75	577
80	621	80	621	80	621	80	621	80	581
85	621	85	621	85	621	85	621	85	585
90	621	90	621	90	621	90	621	90	589
95	621	95	621	95	621	95	621	95	594
100	621	100	621	100	621	100	621	100	599
105	621	105	621	105	621	105	621	105	605
110	621	110	621	110	621	110	621	110	612
115	621	115	621	115	621	115	621	115	620
120	621	120	621	120	621	120	621	120	621
125	621	125	621	125	621	125	621	125	621
130	621	130	621	130	621	130	621	130	621
135	621	135	621	135	621	135	621	135	621
140	621	140	621	140	621	140	621	140	621
145	621	145	621	145	621	145	621	145	621
150	621	150	621	150	621	150	621	150	621
155	621	155	621	155	621	155	621	155	621
160	621	160	621	160	621	160	621	160	621
165	621	165	621	165	621	165	621	165	621
170	621	170	621	170	621	170	621	170	621
175	621	175	621	175	621	175	621	175	621
180	621	180	621	180	621	180	621	180	621
180	929	180	911	180	893	180	877	180	845
185	954	185	937	185	921	185	906	185	878
190	982	190	967	190	952	190	939	190	915
195	1012	195	999	195	987	195	975	195	956
200	1046	200	1035	200	1025	200	1016	200	1002
205	1083	205	1074	205	1067	205	1060	205	1052
210	1125	210	1118	210	1113	210	1109	210	1108
215	1170	215	1167	215	1164	215	1164	215	1164
220	1221	220	1220	220	1220	220	1220	220	1220
225	1277	225	1277	225	1277	225	1277	225	1277
230	1338	230	1338	230	1338	230	1338	230	1338
235	1406	235	1406	235	1406	235	1406	235	1406

Steady State		20 °F/hr		40°F/hr		60°F/hr		100°F/hr	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
240	1482	240	1482	240	1482	240	1482	240	1482
245	1565	245	1565	245	1565	245	1565	245	1565
250	1657	250	1657	250	1657	250	1657	250	1657
255	1758	255	1758	255	1758	255	1758	255	1758
260	1871	260	1871	260	1871	260	1871	260	1871
265	1995	265	1995	265	1995	265	1995	265	1995
270	2132	270	2132	270	2132	270	2132	270	2132
275	2283	275	2283	275	2283	275	2283	275	2283
280	2451	280	2451	280	2451	280	2451	280	2451
280.9	2485	280.9	2485	280.9	2485	280.9	2485	280.9	2485

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Intermediate Shell Longitudinal Welds SA-812 (ID) and SA-775 (OD)
LIMITING ART VALUES AT 53 EFPY (with Hafnium): 1/4T, 218.8°F
3/4T, 183.3°F

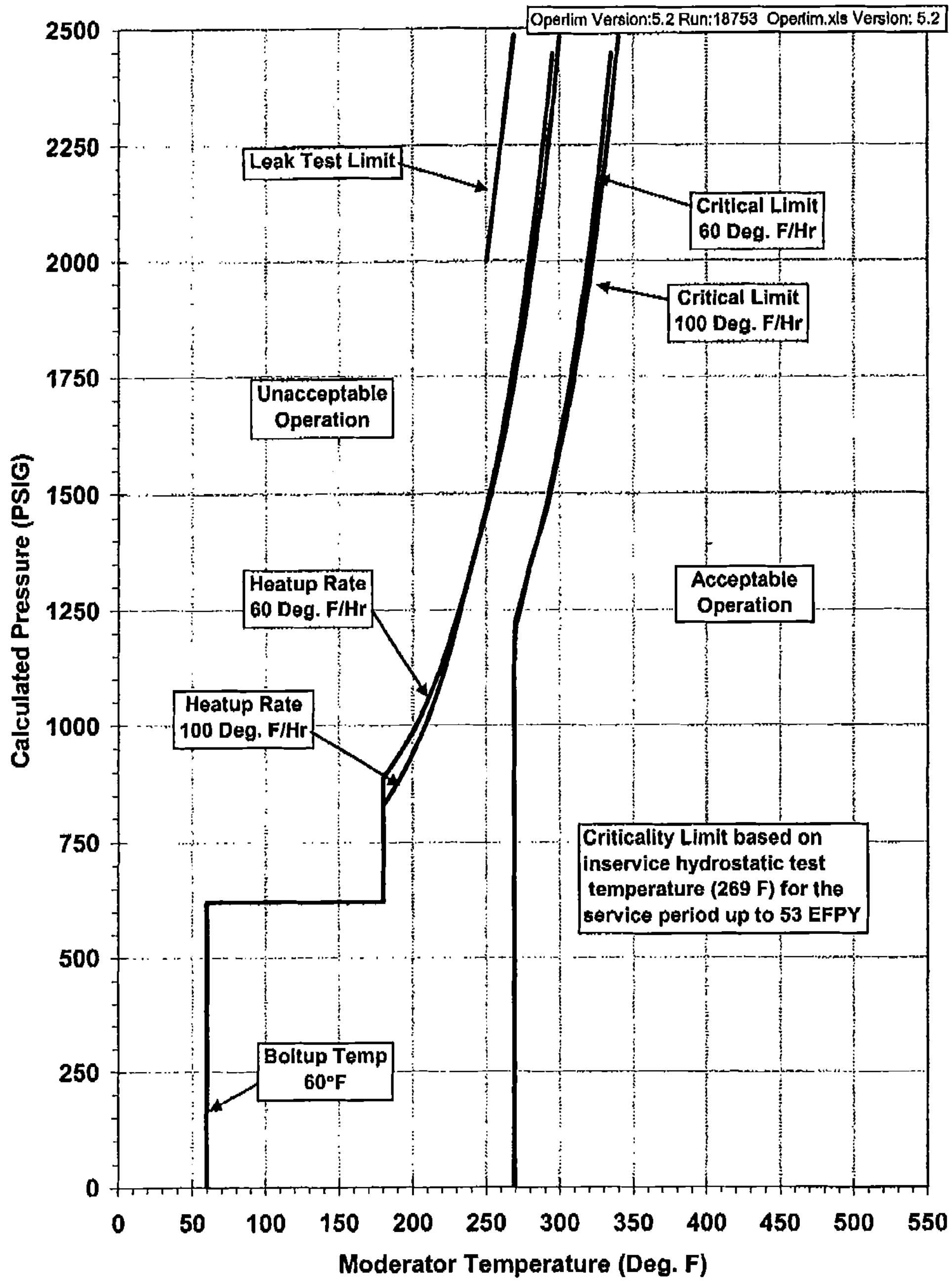

Figure 5-3 Point Beach Units 1 and 2 Reactor Coolant System Heatup Limitations (Heatup Rates of 60 and 100°F/hr) Applicable for 53 EFPY (with Hafnium and without Margins for Instrumentation Errors) Using 1998 App. G Methodology (w/K_{Ic})

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Intermediate Shell Longitudinal Welds SA-812 (ID) and SA-775 (OD)
LIMITING ART VALUES AT 53 EFPY (with Hafnium): 1/4T, 218.8°F
3/4T, 183.3°F

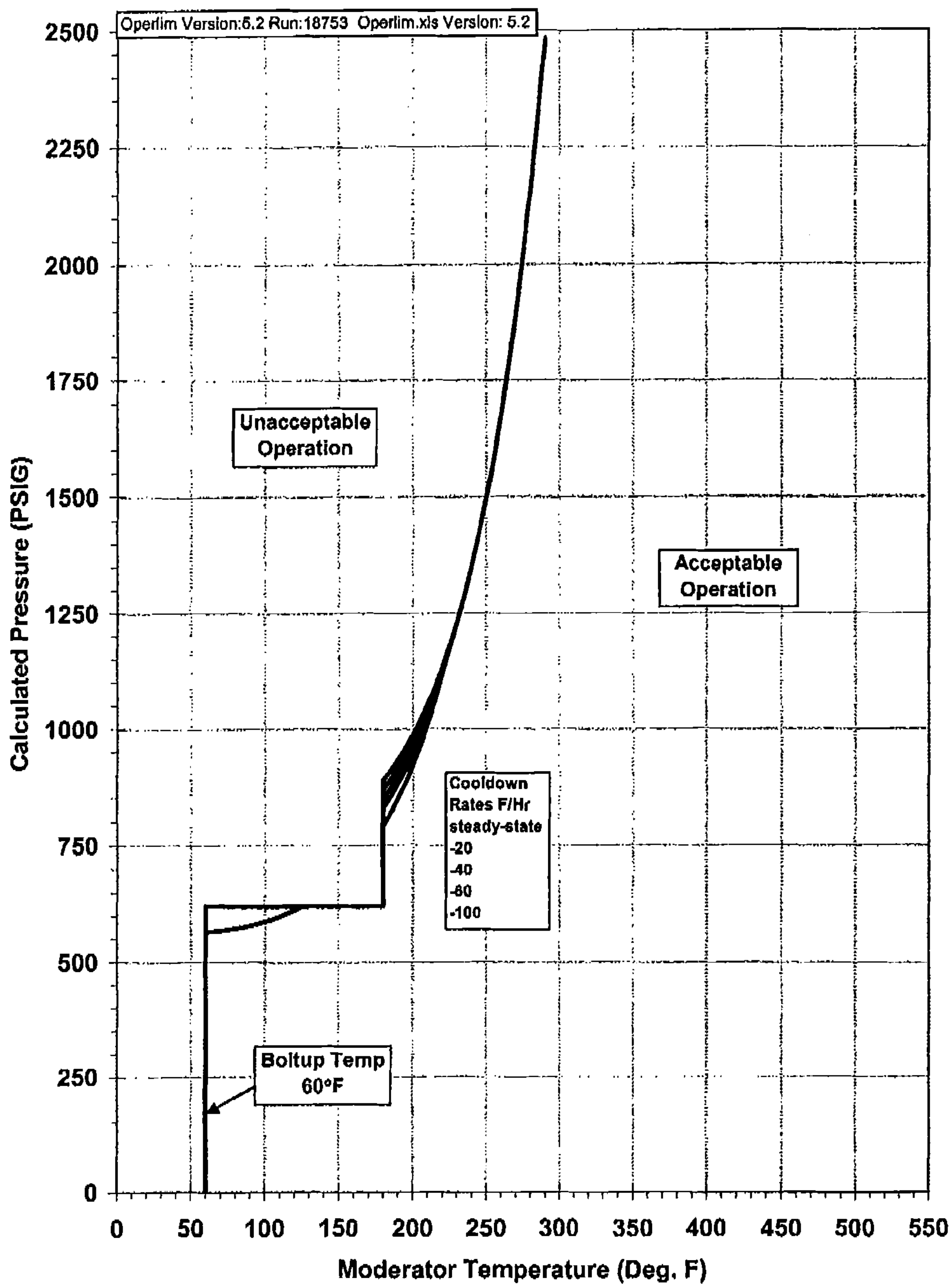

Figure 5-4 Point Beach Units 1 and 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for 53 EFPY (with Hafnium and without Margins for Instrumentation Errors) Using 1998 App. G Methodology (w/K_{Ic})

TABLE 5-3
53 EFPY Heatup Curve Data Points Using 1998 App. G Methodology
(w/Hafnium, w/K_{IC}, w/Flange Notch and w/o Uncertainties for Instrumentation Errors)

60°F/hr Heatup		Criticality Limit		100°F/hr Heatup		Criticality Limit	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
60	0	269	0	60	0	269	0
60	621	269	621	60	621	269	621
65	621	269	621	65	621	269	621
70	621	269	621	70	621	269	621
75	621	269	621	75	621	269	621
80	621	269	621	80	621	269	621
85	621	269	621	85	621	269	621
90	621	269	621	90	621	269	621
95	621	269	621	95	621	269	621
100	621	269	621	100	621	269	621
105	621	269	621	105	621	269	621
110	621	269	621	110	621	269	621
115	621	269	621	115	621	269	621
120	621	269	621	120	621	269	621
125	621	269	621	125	621	269	621
130	621	269	621	130	621	269	621
135	621	269	621	135	621	269	621
140	621	269	621	140	621	269	621
145	621	269	621	145	621	269	621
150	621	269	621	150	621	269	621
155	621	269	621	155	621	269	621
160	621	269	621	160	621	269	621
165	621	269	621	165	621	269	621
170	621	269	621	170	621	269	621
175	621	269	621	175	621	269	621
180	621	269	889	180	621	269	829
180	889	269	910	180	829	269	853
185	910	269	933	185	853	269	879
190	933	269	959	190	879	269	907
195	959	269	987	195	907	269	939
200	987	269	1018	200	939	269	974
205	1018	269	1053	205	974	269	1013
210	1053	269	1091	210	1013	269	1056
215	1091	269	1133	215	1056	269	1103
220	1133	269	1179	220	1103	269	1156
225	1179	270	1230	225	1156	270	1214
230	1230	275	1287	230	1214	275	1277
235	1287	280	1343	235	1277	280	1348
240	1343	285	1401	240	1348	285	1400
245	1401	290	1466	245	1400	290	1457
250	1466	295	1537	250	1457	295	1520
255	1537	300	1615	255	1520	300	1590
260	1615	305	1701	260	1590	305	1667

60°F/hr Heatup		Criticality Limit		100°F/hr Heatup		Criticality Limit	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
265	1701	310	1797	265	1667	310	1752
270	1797	315	1902	270	1752	315	1846
275	1902	320	2018	275	1846	320	1949
280	2018	325	2147	280	1949	325	2063
285	2147	330	2289	285	2063	330	2189
290	2289	335	2445	290	2189	335	2328
295	2445			295	2328	340	2481
				300	2481		
Leak Test Limit		Temperature (°F)		250	269		
		Pressure (psig)		2000	2485		

TABLE 5-4
53 EFPY Cooldown Curve Data Points Using 1998 App. G Methodology
(w/Hafnium, w/K_{IC}, w/Flange Notch and w/o Uncertainties for Instrumentation Errors)

Steady State		20 °F/hr		40°F/hr		60°F/hr		100°F/hr	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
60	0	60	0	60	0	60	0	60	0
60	621	60	621	60	621	60	621	60	565
65	621	65	621	65	621	65	621	65	566
70	621	70	621	70	621	70	621	70	568
75	621	75	621	75	621	75	621	75	571
80	621	80	621	80	621	80	621	80	574
85	621	85	621	85	621	85	621	85	577
90	621	90	621	90	621	90	621	90	580
95	621	95	621	95	621	95	621	95	584
100	621	100	621	100	621	100	621	100	589
105	621	105	621	105	621	105	621	105	593
110	621	110	621	110	621	110	621	110	599
115	621	115	621	115	621	115	621	115	605
120	621	120	621	120	621	120	621	120	612
125	621	125	621	125	621	125	621	125	620
130	621	130	621	130	621	130	621	130	621
135	621	135	621	135	621	135	621	135	621
140	621	140	621	140	621	140	621	140	621
145	621	145	621	145	621	145	621	145	621
150	621	150	621	150	621	150	621	150	621
155	621	155	621	155	621	155	621	155	621
160	621	160	621	160	621	160	621	160	621
165	621	165	621	165	621	165	621	165	621
170	621	170	621	170	621	170	621	170	621
175	621	175	621	175	621	175	621	175	621
180	621	180	621	180	621	180	621	180	621
180	621	180	621	180	621	180	621	180	621
180	889	180	869	180	849	180	829	180	791
185	910	185	891	185	872	185	854	185	819
190	933	190	915	190	898	190	881	190	850
195	959	195	942	195	926	195	911	195	884
200	987	200	972	200	958	200	945	200	922
205	1018	205	1005	205	993	205	982	205	963
210	1053	210	1042	210	1032	210	1023	210	1010
215	1091	215	1082	215	1074	215	1068	215	1061
220	1133	220	1126	220	1122	220	1119	220	1118
225	1179	225	1176	225	1174	225	1174	225	1174
230	1230	230	1230	230	1230	230	1230	230	1230
235	1287	235	1287	235	1287	235	1287	235	1287

Steady State		20 °F/hr		40°F/hr		60°F/hr		100°F/hr	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
240	1350	240	1350	240	1350	240	1350	240	1350
245	1419	245	1419	245	1419	245	1419	245	1419
250	1496	250	1496	250	1496	250	1496	250	1496
255	1581	255	1581	255	1581	255	1581	255	1581
260	1674	260	1674	260	1674	260	1674	260	1674
265	1778	265	1778	265	1778	265	1778	265	1778
270	1892	270	1892	270	1892	270	1892	270	1892
275	2018	275	2018	275	2018	275	2018	275	2018
280	2158	280	2158	280	2158	280	2158	280	2158
285	2312	285	2312	285	2312	285	2312	285	2312
290	2483	290	2483	290	2483	290	2483	290	2483
290.1	2485	290.1	2485	290.1	2485	290.1	2485	290.1	2485

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Intermediate Shell Longitudinal Welds SA-812 (ID) and SA-775 (OD)
LIMITING ART VALUES AT 43 EFPY (Hafnium Removal): 1/4T, 210.5°F
3/4T, 174.6°F

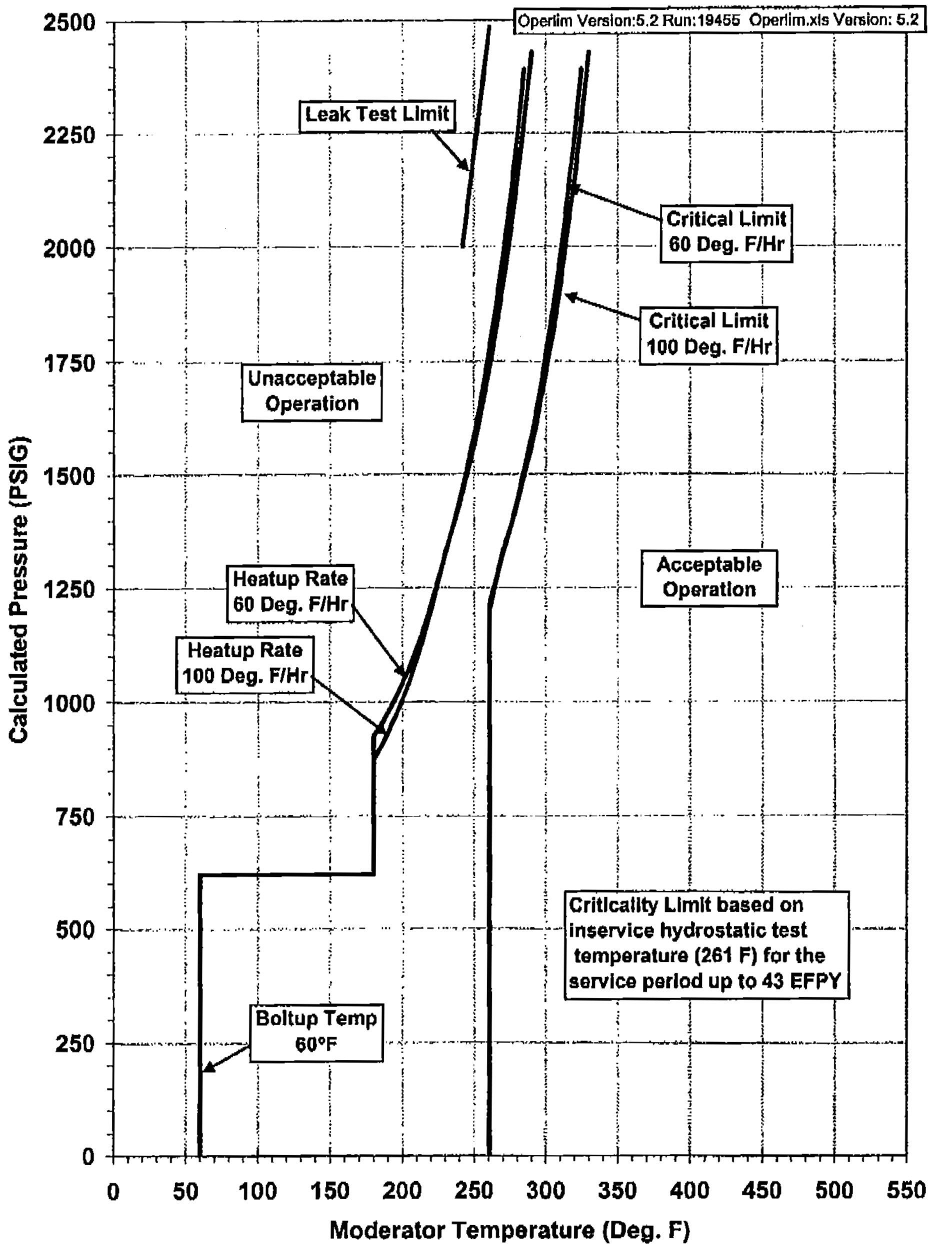

Figure 5-5 Point Beach Units 1 and 2 Reactor Coolant System Heatup Limitations (Heatup Rates of 60 and 100°F/hr) Applicable for 43 EFPY (with Hafnium Removal and without Margins for Instrumentation Errors) Using 1998 App. G Methodology (w/K_{Ic})

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Intermediate Shell Longitudinal Welds SA-812 (ID) and SA-775 (OD)
LIMITING ART VALUES AT 43 EFPY (Hafnium Removal): 1/4T, 210.5°F
3/4T, 174.6°F

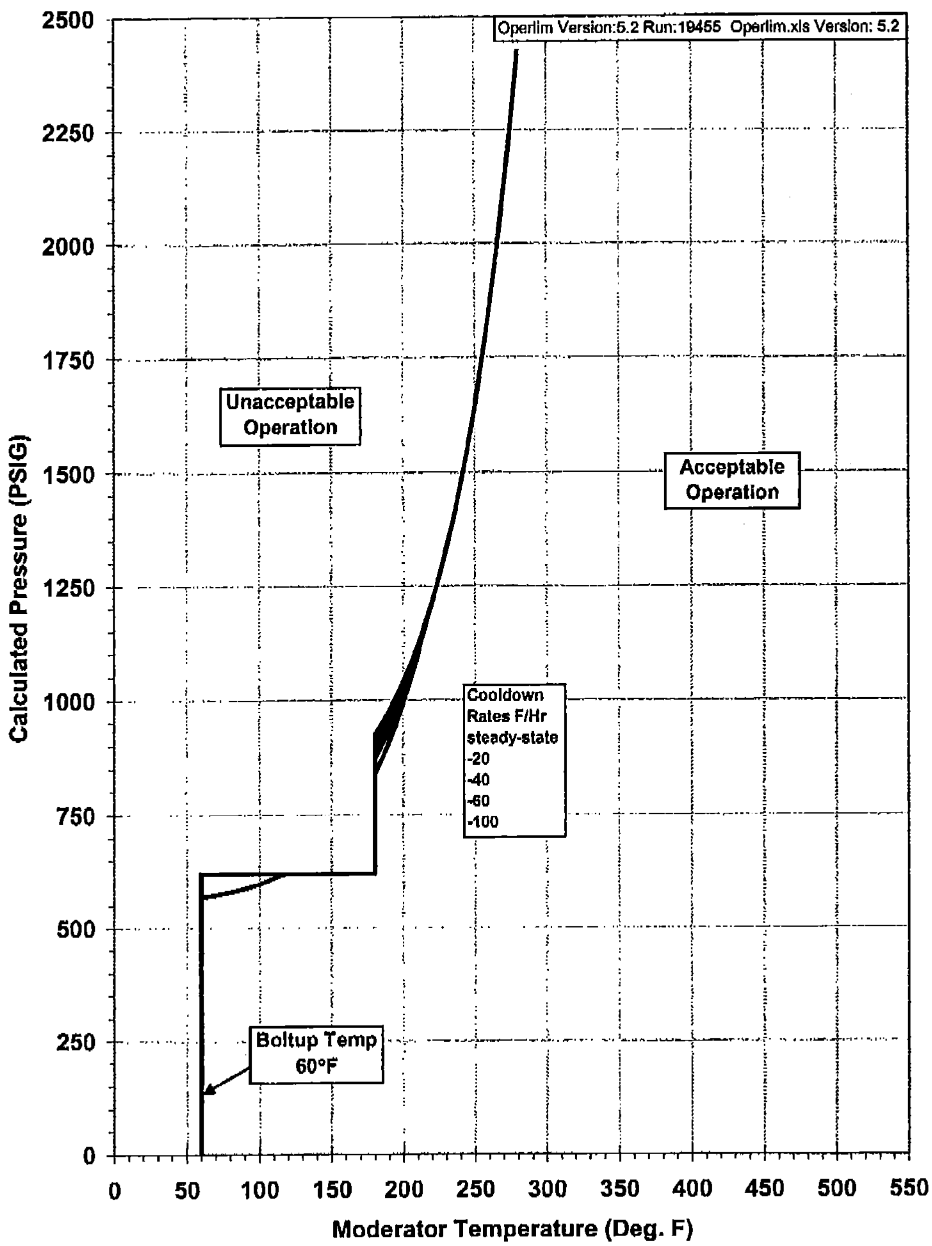

Figure 5-6 Point Beach Units 1 and 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for 43 EFPY (with Hafnium Removal and without Margins for Instrumentation Errors) Using 1998 App. G Methodology (w/K_{Ic})

TABLE 5-5
43 EFPY Heatup Curve Data Points Using 1998 App. G Methodology (w/Hafnium Removal, w/K_{IC}, w/Flange Notch and w/o Uncertainties for Instrumentation Errors)

60°F/hr Heatup		Criticality Limit		100°F/hr Heatup		Criticality Limit	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
60	0	261	0	60	0	261	0
60	621	261	621	60	621	261	621
65	621	261	621	65	621	261	621
70	621	261	621	70	621	261	621
75	621	261	621	75	621	261	621
80	621	261	621	80	621	261	621
85	621	261	621	85	621	261	621
90	621	261	621	90	621	261	621
95	621	261	621	95	621	261	621
100	621	261	621	100	621	261	621
105	621	261	621	105	621	261	621
110	621	261	621	110	621	261	621
115	621	261	621	115	621	261	621
120	621	261	621	120	621	261	621
125	621	261	621	125	621	261	621
130	621	261	621	130	621	261	621
135	621	261	621	135	621	261	621
140	621	261	621	140	621	261	621
145	621	261	621	145	621	261	621
150	621	261	621	150	621	261	621
155	621	261	621	155	621	261	621
160	621	261	621	160	621	261	621
165	621	261	621	165	621	261	621
170	621	261	621	170	621	261	621
175	621	261	621	175	621	261	621
180	621	261	925	180	621	261	874
180	925	261	950	180	874	261	902
185	950	261	977	185	902	261	933
190	977	261	1007	190	933	261	967
195	1007	261	1040	195	967	261	1005
200	1040	261	1077	200	1005	261	1047
205	1077	261	1118	205	1047	261	1093
210	1118	261	1163	210	1093	261	1145
215	1163	261	1212	215	1145	261	1201
220	1212	265	1267	220	1201	265	1263
225	1267	270	1325	225	1263	270	1331
230	1325	275	1381	230	1328	275	1381
235	1381	280	1443	235	1381	280	1437
240	1443	285	1512	240	1437	285	1498
245	1512	290	1587	245	1498	290	1566
250	1587	295	1671	250	1566	295	1640

60°F/hr Heatup		Criticality Limit		100°F/hr Heatup		Criticality Limit	
T (°F)	**P (psig)**	**T (°F)**	**P (psig)**	**T (°F)**	**P (psig)**	**T (°F)**	**P (psig)**
255	1671	300	1763	255	1640	300	1722
260	1763	305	1865	260	1722	305	1813
265	1865	310	1978	265	1813	310	1913
270	1978	315	2102	270	1913	315	2024
275	2102	320	2240	275	2024	320	2146
280	2240	325	2391	280	2146	325	2280
285	2391			285	2280	330	2429
				290	2429		
Leak Test Limit		**Temperature (°F)**		242	261		
		Pressure (psig)		2000	2485		

TABLE 5-6
43 EFPY Cooldown Curve Data Points Using 1998 App. G Methodology (w/Hafnium Removal, w/K_{IC}, w/Flange Notch and w/o Uncertainties for Instrumentation Errors)

Steady State		20 °F/hr		40°F/hr		60°F/hr		100°F/hr	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
60	0	60	0	60	0	60	0	60	0
60	621	60	621	60	621	60	621	60	569
65	621	65	621	65	621	65	621	65	571
70	621	70	621	70	621	70	621	70	574
75	621	75	621	75	621	75	621	75	577
80	621	80	621	80	621	80	621	80	580
85	621	85	621	85	621	85	621	85	584
90	621	90	621	90	621	90	621	90	588
95	621	95	621	95	621	95	621	95	593
100	621	100	621	100	621	100	621	100	598
105	621	105	621	105	621	105	621	105	604
110	621	110	621	110	621	110	621	110	611
115	621	115	621	115	621	115	621	115	618
120	621	120	621	120	621	120	621	120	621
125	621	125	621	125	621	125	621	125	621
130	621	130	621	130	621	130	621	130	621
135	621	135	621	135	621	135	621	135	621
140	621	140	621	140	621	140	621	140	621
145	621	145	621	145	621	145	621	145	621
150	621	150	621	150	621	150	621	150	621
155	621	155	621	155	621	155	621	155	621
160	621	160	621	160	621	160	621	160	621
165	621	165	621	165	621	165	621	165	621
170	621	170	621	170	621	170	621	170	621
175	621	175	621	175	621	175	621	175	621
180	621	180	621	180	621	180	621	180	621
180	925	180	907	180	889	180	872	180	840
185	950	185	933	185	917	185	901	185	873
190	977	190	962	190	947	190	933	190	909
195	1007	195	994	195	981	195	969	195	949
200	1040	200	1029	200	1018	200	1009	200	994
205	1077	205	1068	205	1060	205	1053	205	1043
210	1118	210	1111	210	1105	210	1101	210	1098
215	1163	215	1158	215	1156	215	1155	215	1155
220	1212	220	1211	220	1211	220	1211	220	1211
225	1267	225	1267	225	1267	225	1267	225	1267
230	1328	230	1328	230	1328	230	1328	230	1328

Steady State		20 °F/hr		40°F/hr		60°F/hr		100°F/hr	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
235	1395	235	1395	235	1395	235	1395	235	1395
240	1469	240	1469	240	1469	240	1469	240	1469
245	1551	245	1551	245	1551	245	1551	245	1551
250	1641	250	1641	250	1641	250	1641	250	1641
255	1741	255	1741	255	1741	255	1741	255	1741
260	1852	260	1852	260	1852	260	1852	260	1852
265	1974	265	1974	265	1974	265	1974	265	1974
270	2109	270	2109	270	2109	270	2109	270	2109
275	2258	275	2258	275	2258	275	2258	275	2258
280	2423	280	2423	280	2423	280	2423	280	2423
281.7	2485	281.7	2485	281.7	2485	281.7	2485	281.7	2485

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Intermediate Shell Longitudinal Welds SA-812 (ID) and SA-775 (OD)

LIMITING ART VALUES AT 53 EFPY (Hafnium Removal):	1/4T,	220.0°F
	3/4T,	184.6°F

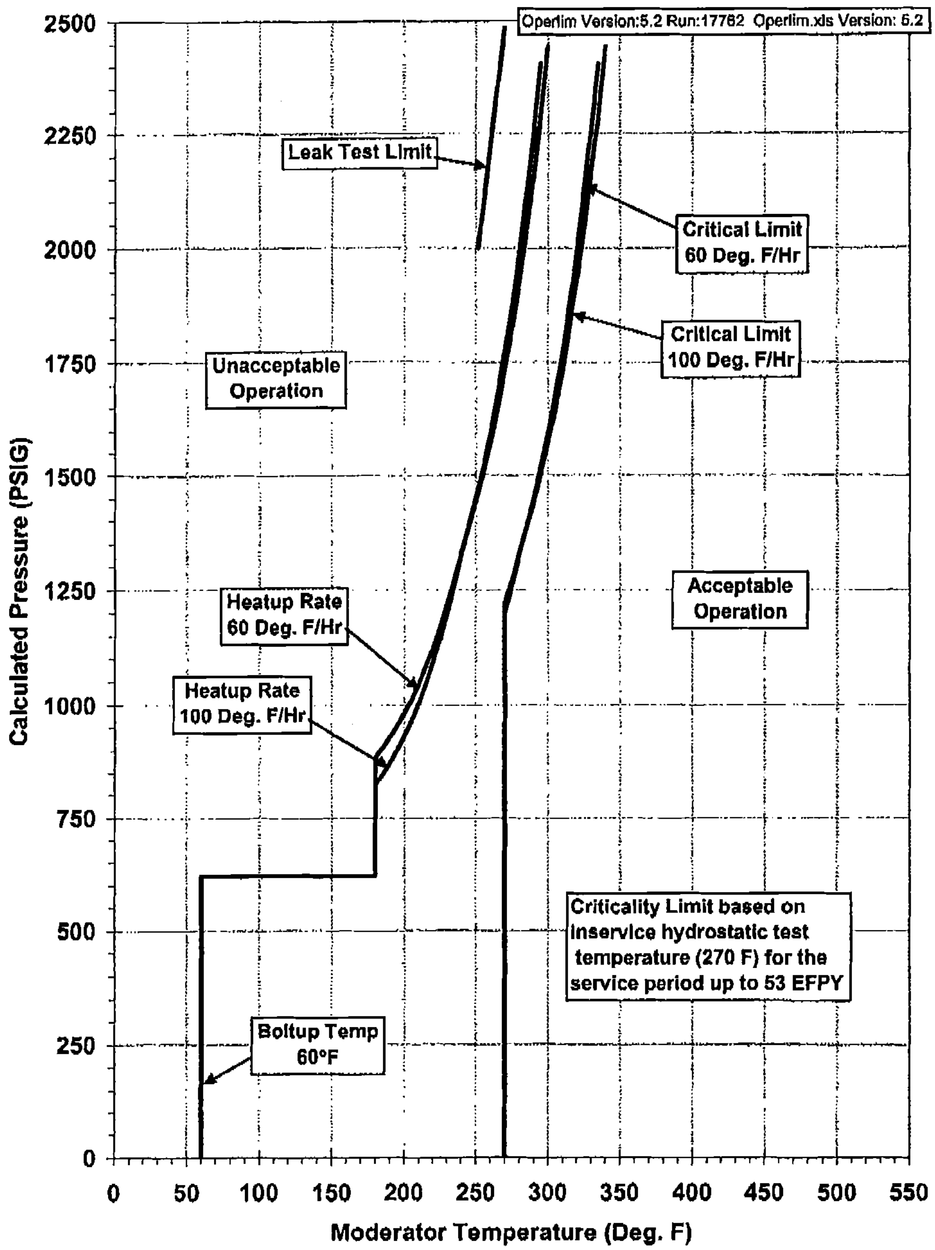

Figure 5-7 Point Beach Units 1 and 2 Reactor Coolant System Heatup Limitations (Heatup Rates of 60 and 100°F/hr) Applicable for 53 EFPY (with Hafnium Removal and without Margins for Instrumentation Errors) Using 1998 App. G Methodology (w/K_{Ic})

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Intermediate Shell Longitudinal Welds SA-812 (ID) and SA-775 (OD)
LIMITING ART VALUES AT 53 EFPY (Hafnium Removal): 1/4T, 220.0°F
3/4T, 184.6°F

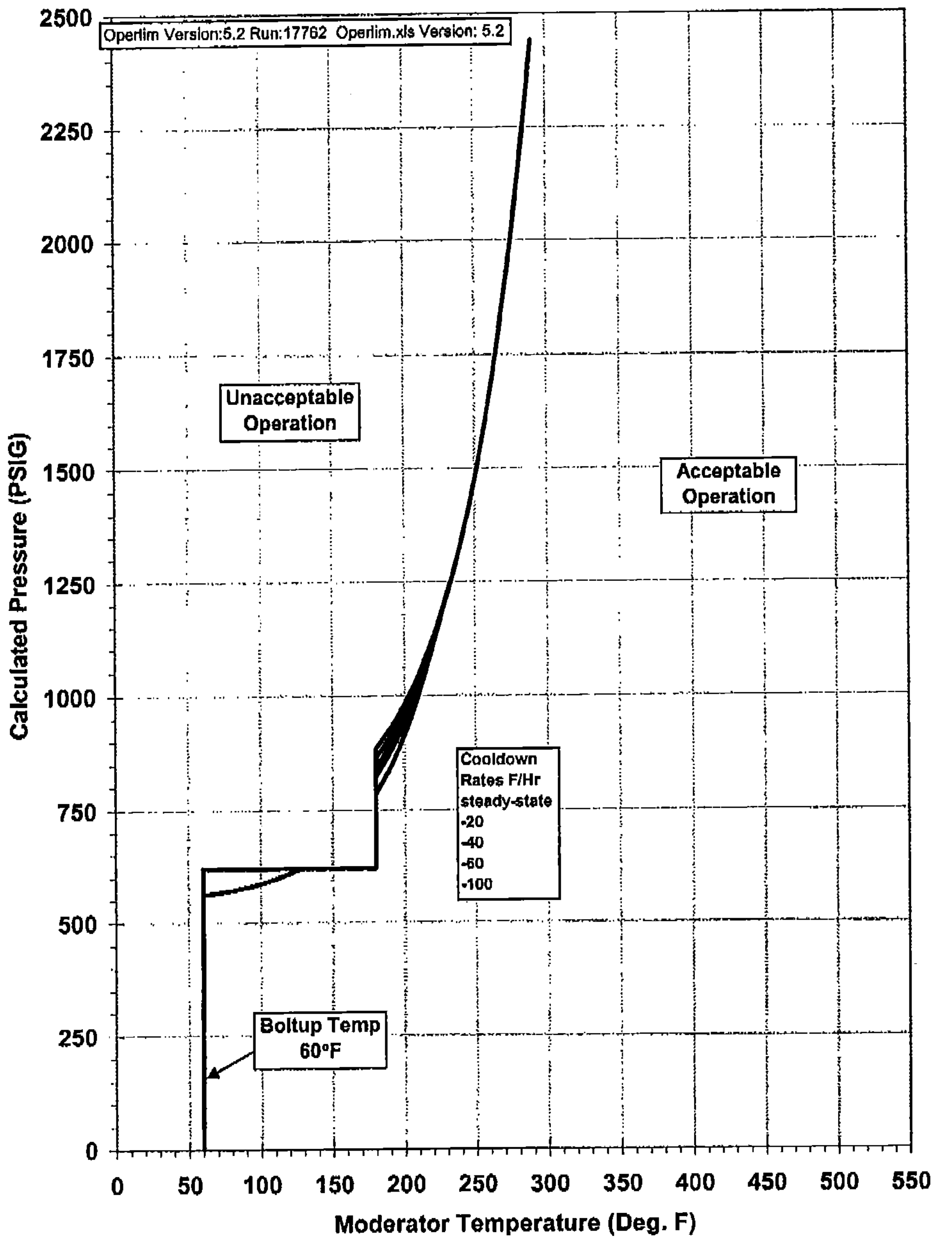

Figure 5-8 **Point Beach Units 1 and 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for 53 EFPY (with Hafnium Removal and without Margins for Instrumentation Errors) Using 1998 App. G Methodology (w/K_{Ic})**

TABLE 5-7
53 EFPY Heatup Curve Data Points Using 1998 App. G Methodology
(w/Hafnium Removal, w/K_{IC}, w/Flange Notch and w/o Uncertainties for Instrumentation Errors)

60°F/hr Heatup		Criticality Limit		100°F/hr Heatup		Criticality Limit	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
60	0	270	0	60	0	270	0
60	621	270	621	60	621	270	621
65	621	270	621	65	621	270	621
70	621	270	621	70	621	270	621
75	621	270	621	75	621	270	621
80	621	270	621	80	621	270	621
85	621	270	621	85	621	270	621
90	621	270	621	90	621	270	621
95	621	270	621	95	621	270	621
100	621	270	621	100	621	270	621
105	621	270	621	105	621	270	621
110	621	270	621	110	621	270	621
115	621	270	621	115	621	270	621
120	621	270	621	120	621	270	621
125	621	270	621	125	621	270	621
130	621	270	621	130	621	270	621
135	621	270	621	135	621	270	621
140	621	270	621	140	621	270	621
145	621	270	621	145	621	270	621
150	621	270	621	150	621	270	621
155	621	270	621	155	621	270	621
160	621	270	621	160	621	270	621
165	621	270	621	165	621	270	621
170	621	270	621	170	621	270	621
175	621	270	621	175	621	270	621
180	621	270	885	180	621	270	823
180	885	270	905	180	823	270	846
185	905	270	928	185	846	270	871
190	928	270	952	190	871	270	899
195	952	270	980	195	899	270	930
200	980	270	1010	200	930	270	965
205	1010	270	1044	205	965	270	1002
210	1044	270	1081	210	1002	270	1044
215	1081	270	1122	215	1044	270	1090
220	1122	270	1168	220	1090	270	1141
225	1168	270	1218	225	1141	270	1198
230	1218	275	1273	230	1198	275	1260
235	1273	280	1330	235	1260	280	1328
240	1330	285	1387	240	1328	285	1387
245	1387	290	1450	245	1387	290	1443
250	1450	295	1519	250	1443	295	1505

60°F/hr Heatup		Criticality Limit		100°F/hr Heatup		Criticality Limit	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
255	1519	300	1595	255	1505	300	1573
260	1595	305	1680	260	1573	305	1648
265	1680	310	1773	265	1648	310	1731
270	1773	315	1876	270	1731	315	1822
275	1876	320	1989	275	1822	320	1923
280	1989	325	2115	280	1923	325	2035
285	2115	330	2253	285	2035	330	2158
290	2253	335	2406	290	2158	335	2293
295	2406			295	2293	340	2443
				300	2443		
Leak Test Limit		Temperature (°F)		251	270		
		Pressure (psig)		2000	2485		

TABLE 5-8
53 EFPY Cooldown Curve Data Points Using 1998 App. G Methodology (w/Hafnium Removal, w/K_{IC}, w/Flange Notch and w/o Uncertainties for Instrumentation Errors)

Steady State		20 °F/hr		40°F/hr		60°F/hr		100°F/hr	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
60	0	60	0	60	0	60	0	60	0
60	621	60	621	60	621	60	621	60	564
65	621	65	621	65	621	65	621	65	566
70	621	70	621	70	621	70	621	70	568
75	621	75	621	75	621	75	621	75	570
80	621	80	621	80	621	80	621	80	573
85	621	85	621	85	621	85	621	85	576
90	621	90	621	90	621	90	621	90	579
95	621	95	621	95	621	95	621	95	583
100	621	100	621	100	621	100	621	100	587
105	621	105	621	105	621	105	621	105	592
110	621	110	621	110	621	110	621	110	598
115	621	115	621	115	621	115	621	115	604
120	621	120	621	120	621	120	621	120	610
125	621	125	621	125	621	125	621	125	618
130	621	130	621	130	621	130	621	130	621
135	621	135	621	135	621	135	621	135	621
140	621	140	621	140	621	140	621	140	621
145	621	145	621	145	621	145	621	145	621
150	621	150	621	150	621	150	621	150	621
155	621	155	621	155	621	155	621	155	621
160	621	160	621	160	621	160	621	160	621
165	621	165	621	165	621	165	621	165	621
170	621	170	621	170	621	170	621	170	621
175	621	175	621	175	621	175	621	175	621
180	621	180	621	180	621	180	621	180	621
180	885	180	864	180	844	180	824	180	785
185	905	185	885	185	866	185	848	185	812
190	928	190	909	190	891	190	874	190	842
195	952	195	936	195	919	195	904	195	875
200	980	200	965	200	950	200	936	200	912
205	1010	205	997	205	984	205	973	205	953
210	1044	210	1032	210	1022	210	1013	210	998
215	1081	215	1072	215	1064	215	1057	215	1048
220	1122	220	1115	220	1110	220	1106	220	1104
225	1168	225	1163	225	1161	225	1160	225	1160
230	1218	230	1217	230	1217	230	1217	230	1217

Steady State		20 °F/hr		40°F/hr		60°F/hr		100°F/hr	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
235	1273	235	1273	235	1273	235	1273	235	1273
240	1334	240	1334	240	1334	240	1334	240	1334
245	1402	245	1402	245	1402	245	1402	245	1402
250	1477	250	1477	250	1477	250	1477	250	1477
255	1560	255	1560	255	1560	255	1560	255	1560
260	1651	260	1651	260	1651	260	1651	260	1651
265	1752	265	1752	265	1752	265	1752	265	1752
270	1864	270	1864	270	1864	270	1864	270	1864
275	1987	275	1987	275	1987	275	1987	275	1987
280	2123	280	2123	280	2123	280	2123	280	2123
285	2274	285	2274	285	2274	285	2274	285	2274
290	2440	290	2440	290	2440	290	2440	290	2440
291.2	2485	291.2	2485	291.2	2485	291.2	2485	291.2	2485

6 REFERENCES

1. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, May 1988.

2. WCAP-14040-NP-A, Revision 4, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", J.D. Andrachek, et al., May 2004.

3. Section XI of the ASME Boiler and Pressure Vessel Code, Appendix G, "Fracture Toughness Criteria for Protection Against Failure." Dated December 1998, through 2000 Addendum.

4. AREVA Document 323-9019240-000, "ART Values for Point Beach Unit 1 and Unit 2," S.B. Davidsaver, et al., dated 6/20/06.

5. AREVA Document 43-2308-02, K. K. Yoon, *"Initial RT_{NDT} of Linde 80 Weld Materials,"* (BAW-2308, Revision 1-A), August 2005.

6. Westinghouse Letter to WEP, WEP-06-13, "Statistical Evaluation of Reactor Vessel Dosimetry – Point Beach Units 1 and 2," Kerry B. Hanahan, dated February 14, 2006.

7. WCAP-16083-NP-A, "Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry," S.L. Anderson, May 2006.

8. Code of Federal Regulations, 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Washington, D.C., Federal Register, Volume 60, No. 243, dated December 19, 1995.

9. "Fracture Toughness Requirements", Branch Technical Position MTEB 5-2, Chapter 5.3.2 in Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-0800, 1981.

10. WCAP-15121, Revision 1, "Point Beach Units 1 and 2 WOG Reactor Vessel 60-Year Evaluation Minigroup Heatup and Cooldown Limit Curves for Normal Operation," J.H. Ledger, April 2001.

11. Westinghouse Letter LTR-REA-04-64, "Pressure Vessel Neutron Exposure Evaluations – Point Beach Units 1 and 2," S.L. Anderson, dated June 4, 2004.

12. Certified Material Test Report MHI-NMC-3277PB1 for Point Beach Unit 1, "CMTR for Closure Head", dated 2/14/05.

13. Certified Material Test Report MHI-NMC-1838PB2 for Point Beach Unit 2, "CMTR for Closure Head", dated 7/2/04.

APPENDIX A

Thermal Stress Intensity Factors (K_{It})

The following pages contain the thermal stress intensity factors (K_{It}) for the maximum heatup and cooldown rates. The vessel radii to the 1/4T and 3/4T locations are as follows:

- 1/4T Radius = 67.781"
- 3/4T Radius = 71.031"

TABLE A-1
K_{It} Values for 100°F/hr Heatup Curve (w/o Margins for Instrument Errors)

Water Temp. (°F)	Vessel Temperature @ 1/4T Location for 100°F/hr Heatup (°F)	1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)	Vessel Temperature @ 3/4T Location for 100°F/hr Heatup (°F)	3/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)
60	56.455	-0.952	55.144	0.524
65	59.695	-2.167	55.835	1.434
70	63.165	-3.048	57.327	2.169
75	66.942	-3.830	59.486	2.775
80	70.966	-4.423	62.167	3.261
85	75.134	-4.934	65.288	3.660
90	79.489	-5.331	68.754	3.983
95	83.928	-5.674	72.503	4.249
100	88.496	-5.943	76.474	4.467
105	93.112	-6.178	80.624	4.649
110	97.816	-6.364	84.919	4.800
115	102.549	-6.530	89.329	4.928
120	107.339	-6.662	93.834	5.035
125	112.148	-6.782	98.414	5.127
130	116.994	-6.879	103.056	5.205
135	121.853	-6.968	107.748	5.274
140	126.736	-7.042	112.480	5.333
145	131.627	-7.113	117.244	5.386
150	136.534	-7.172	122.034	5.434
155	141.447	-7.229	126.846	5.477
160	146.370	-7.278	131.675	5.516
165	151.297	-7.327	136.518	5.553
170	156.230	-7.370	141.372	5.587
175	161.167	-7.413	146.235	5.619
180	166.106	-7.452	151.106	5.650
185	171.049	-7.491	155.983	5.679
190	175.993	-7.527	160.865	5.707
195	180.939	-7.564	165.751	5.735
200	185.886	-7.599	170.639	5.762
205	190.835	-7.634	175.531	5.788
210	195.784	-7.667	180.424	5.814
215	200.735	-7.702	185.319	5.839
220	205.685	-7.734	190.216	5.865
225	210.637	-7.768	195.113	5.890
230	215.587	-7.800	200.012	5.915

TABLE A-2
K_{It} Values for 100°F/hr Cooldown Curve (w/o Margins for Instrument Errors)

Water Temp. (°F)	Vessel Temperature @ 1/4T Location for 100°F/hr Cooldown (°F)	100°F/hr Cooldown 1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)
200.000	215.032	8.131
195.000	209.981	8.097
190.000	204.931	8.063
185.000	199.881	8.029
180.000	194.831	7.995
175.000	189.781	7.961
170.000	184.732	7.927
165.000	179.682	7.893
160.000	174.633	7.860
155.000	169.583	7.826
150.000	164.534	7.793
145.000	159.484	7.759
140.000	154.435	7.726
135.000	149.386	7.693
130.000	144.337	7.660
125.000	139.288	7.627
120.000	134.240	7.594
115.000	129.191	7.561
110.000	124.142	7.528
105.000	119.094	7.495
100.000	114.045	7.463
95.000	108.997	7.430
90.000	103.949	7.397
85.000	98.901	7.365
80.000	93.852	7.333
75.000	88.804	7.300
70.000	83.757	7.268
65.000	78.709	7.236
60.000	73.663	7.203