Entergy

Entergy Nuclear Northeast
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
295 Broadway, Suite 1
P.O. Box 249
Buchanan, NY 10511-0249

January 11, 2002

Re: Indian Point Unit No. 2
Docket No. 50-247
NL-02-006

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, DC 20555-0001

SUBJECT: Response to Request for Additional Information Indian Point 2 License Amendment Request for Reactor Coolant System Heatup and Cooldown Limitation Curves (TAC No.: MB2419)

References: 1. Consolidated Edison letter (NL-01-092) to NRC, "Indian Point 2 License Amendment Request for Reactor Coolant System Heatup and Cooldown Limitation Curves and Request for Exemption from the Requirements of 10CFR50.60(a) and Appendix G," dated July 16, 2001

By letter dated July 16, 2001 (Ref. 1), Consolidated Edison (the former licensee) submitted an application for an amendment to the Technical Specifications (TS) for Indian Point Unit No. 2 (IP2). The proposed amendment requested revised Reactor Coolant System Heatup and Cooldown Limitation Curves, as well as new Overpressure Protection System (OPS) limits. The U.S. Nuclear Regulatory Commission (NRC) staff reviewed this submittal, determined that additional information was required to complete its review, and requested that additional information in telephone conferences on November 14, 2001 and December 18, 2001. As a result of the telephone conferences, Entergy Nuclear Operations, Inc. (ENO – the current licensee) initiated a revision to the original Ref. 1 Attachment 4, "WCAP-15629, Indian Point Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation." Revision 1 to WCAP-15629 (December 2001) is included as Enclosure 1 of this submittal.

In addition, excerpts from an existing IP2 operating procedure for plant heatup are being submitted as an example to demonstrate how IP2 applies instrument uncertainty to the values in the TS curves. Although the 10CFR50, Appendix G pressure/temperature limitations included in the Indian Point 2 Technical Specifications do not include explicit margins to account for instrument uncertainties, the limits in the operating procedures are decreased to account for pressure and temperature uncertainties, as well as system hydraulic losses and elevation corrections. Attachment 1 of this submittal contains excerpts from an IP2 operating procedure.

This letter contains no new commitments.

A001

Should you or your staff have any questions regarding this submittal, please contact Mr. John F. McCann, Manager, Nuclear Safety and Licensing at (914) 734-5074.

Sincerely,

Fred Dacimo
Vice President – Operations
Indian Point 2

Attachment

Enclosure

cc: See page 3

cc:

Mr. Hubert J. Miller
Regional Administrator-Region I
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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
ENTERGY NUCLEAR OPERATIONS, INC.	)	Docket No. 50-247
Indian Point Nuclear Generating Unit No. 2	)	

APPLICATION FOR AMENDMENT TO OPERATING LICENSE

Pursuant to Section 50.90 of the Regulations of the Nuclear Regulatory Commission (NRC), Entergy Nuclear Operations, Inc., as holder of Facility Operating License No. DPR-26, hereby submits additional information in support of the July 16, 2001 application for amendment of the Technical Specifications contained in Appendix A of this license. The specific additional information is set forth in Enclosure 1 and Attachment 1.

As required by 10CFR50.91(b)(1), a copy of this submittal has been provided to the appropriate New York State official designated to receive such amendments.

BY: _____
Fred Dacimo
Vice President – Operations
Indian Point 2

Subscribed and sworn to
before me this 10 day
January, 2002.

Ersilia A. Amanna (Borrero)
Notary Public

ERSILIA A. AMANNA
Notary Public, State of New York
No. 01AM6038389
Qualified in Westchester County
Commission Expires March 20, 2002

ATTACHMENT 1 TO NL-02-006

3 pages from an Indian Point Unit 2 operating procedure

2.3 The Reactor shall be maintained subcritical by at least 1 percent K/K UNTIL an ACTUAL water level of 33 - 40 percent is established in the Pressurizer (Technical Specification 3.1.C.4).

2.4 RCS pressure increases should be limited to 100 psig per hour when above 1700 psig to limit the potential for safety valve leakage.

NOTE

- IF any heatup OR cooldown rate is violated, a safety evaluation SHALL be performed. (Reference 6.2.15)
- The heatup, and cooldown rates in Technical Specification Figure 3.1.B-1, and 3.1.B-2 do NOT make allowance for instrument error. Compensation for pressure, and temperature instrumentation error is as follows:
  - o During Steady-State, use Figure 1 (D1), for RCS Pressure, and Temperature instrument error compensation.
  - o During Heatup, use Figure 2 (D3), for RCS Pressure, and Temperature instrument error compensation.
- The CCR instrumentation to be used is as follows:
  - o RCS Temperature, as indicated on RCS Cold Leg RTD TE-413 (TR-413J), TE-433 (TR-433J), or TE-443 (TR-443J).
  - o RCS pressure above 1500 psig, as indicated on Pressurizer Pressure (if on scale), OR Wide Range indicated pressure on PT-402, or PT-403.
  - o RCS pressure 0 - 1500 psig, as indicated on PT-413 (PI-413K), PT-433 (PI-433K), or PT-443 (PI-443K).

2.5 RCS Heatup Requirements:

2.5.1 RCS

- RCS temperature, AND Pressure SHALL be maintained within the limits of Technical Specification Figure 3.1.B-2, as compensated for, per Step Note, 2nd Bullet, as applicable, AND Graph RCS-12A, 50°F Subcooling and Saturation Curves

FIGURE 1 (D1), RCS TEMPERATURE VS. PRESSURE - STEADY STATE (CORRECTED FOR INSTRUMENT ERROR)

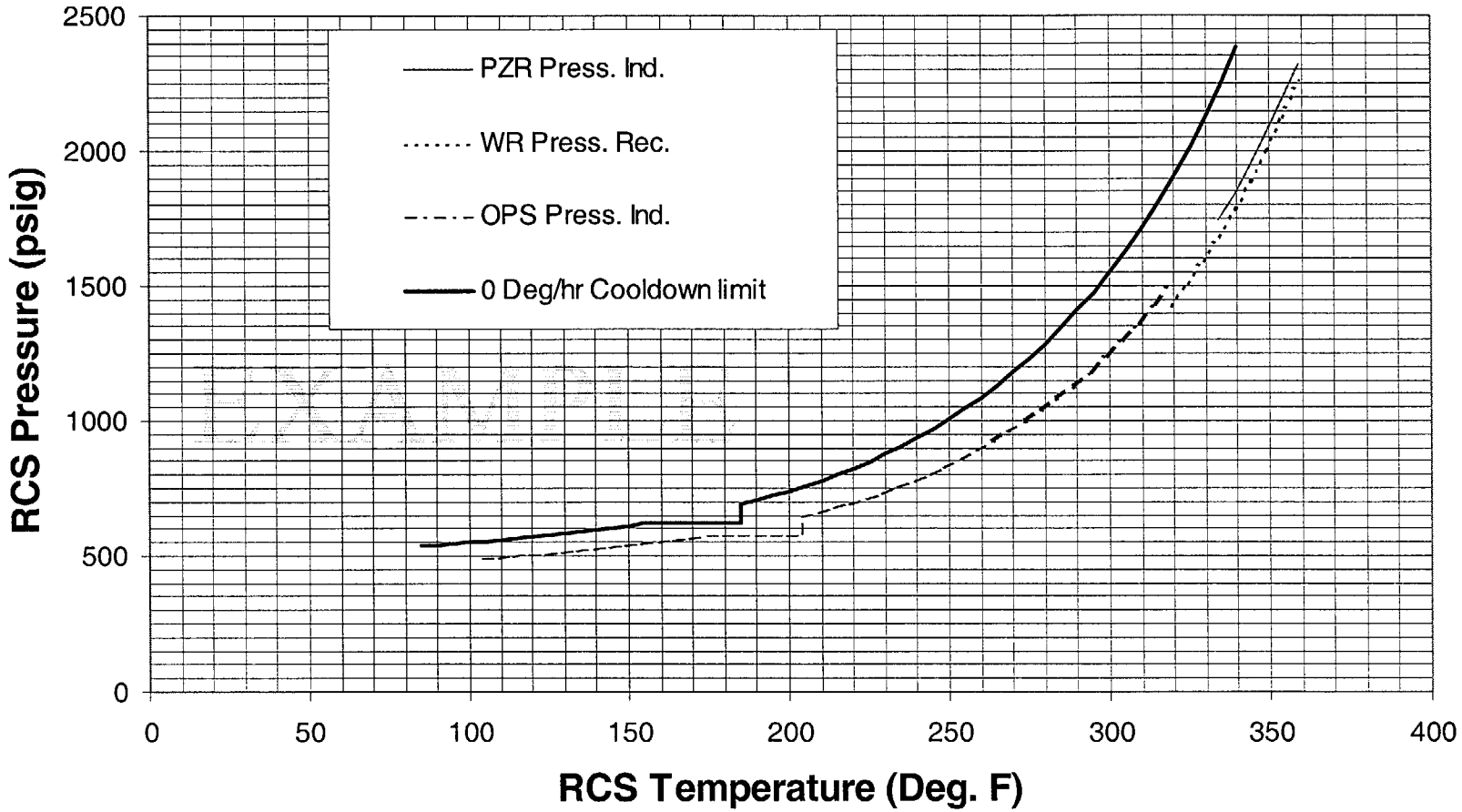
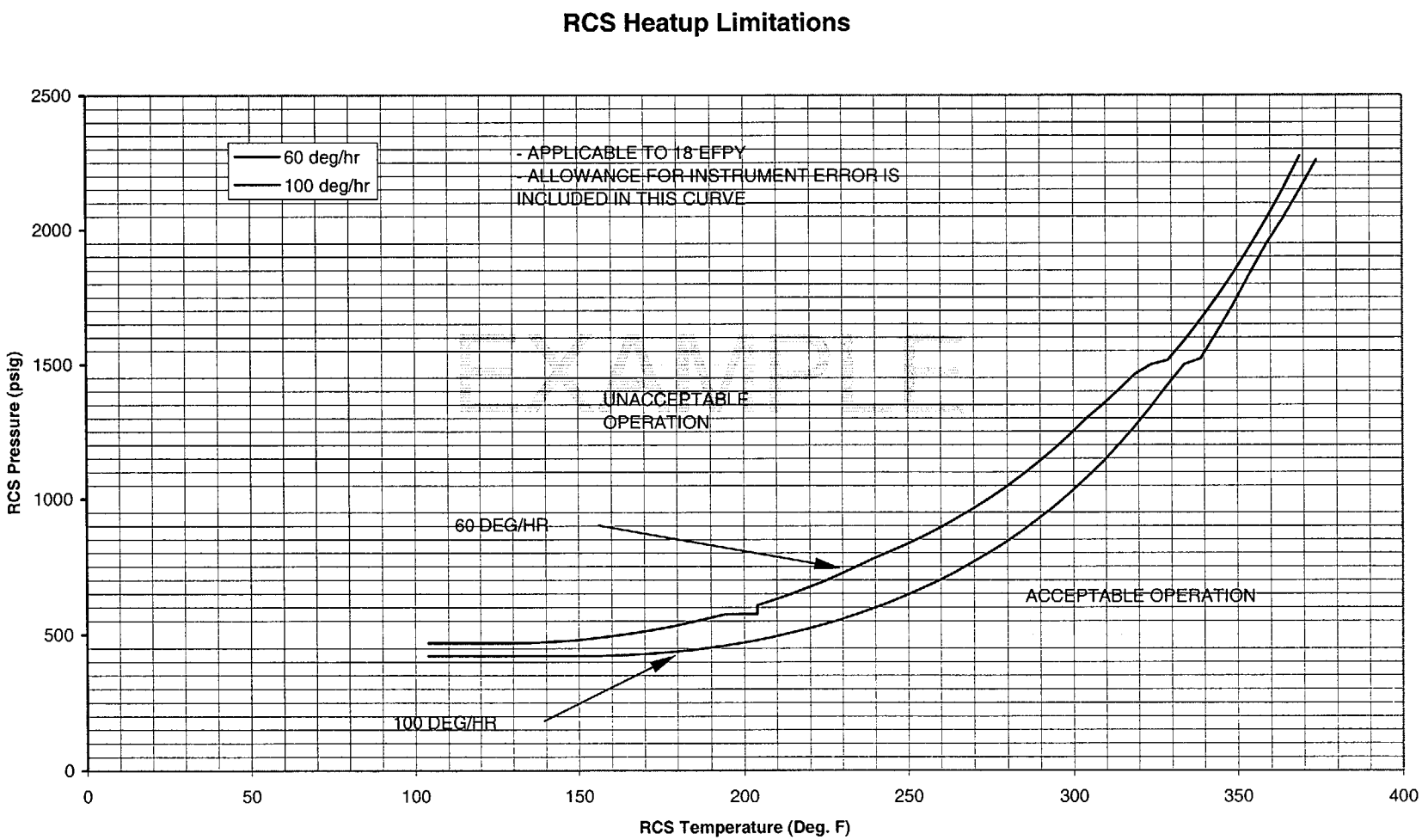


FIGURE 2 (D3), RCS TEMPERATURE VS. PRESSURE - HEATUP (CORRECTED FOR INSTRUMENT ERROR)



ENCLOSURE 1 TO NL-02-006

WCAP-15629, Revision 1, "Indian Point Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation"

ENTERGY NUCLEAR OPERATIONS, INC
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247

Westinghouse Non-Proprietary Class 3

WCAP-15629
Revision 1

Indian Point Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation

Westinghouse Electric Company LLC

WCAP-15629, Revision 1

Indian Point Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation

T. J. Laubham

December 2001

Prepared by the Westinghouse Electric Company LLC
for Entergy

Approved: C H Boyd
C. H. Boyd, Manager
Engineering and Materials Technology

Westinghouse Electric Company LLC
Energy Systems
P.O. Box 355
Pittsburgh, PA 15230-0355

PREFACE

This report has been technically reviewed and verified by:

J.H. Ledger _____ Section 1 through 6 and Appendices A, C through G

S.L. Anderson _____ Appendix B

Record of Revision

Revision 0: Original Issue

Revision 1: The following was revised in this revision:

- Updated text on pages 2, 22, C-1 and G2 to address typos.
- Added clarification to the plate chemistry values in Table 1 (Page 3), and revised the nickel value for the Lower Shell Plate B-2003-2. In turn the chemistry factor for the lower shell plate B-2003-2 was revised in Table 5 (Page 8). This chemistry factor changed resulted in changes to Tables 9 and 10 (Pages 16 & 17).
- Clarified the references for the unirradiated USE in Table D-1. This resulted in adding Reference 17.
- Changed note on page E-1 to read, "Withdrawal Schedule to be provided in PTLR only by Indian Point Unit 2".

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EXECUTIVE SUMMARY

This report provides the methodology and results of the generation of heatup and cooldown pressure temperature limit curves for normal operation of the Indian Point Unit 2 reactor vessel. In addition, Pressure Temperature Limits Report (PTLR) support information, such as Fluence, PTS , EOL USE and Withdrawal Schedule, are documented herein under the Appendices. The PT curves were generated based on the latest available reactor vessel information and updated fluences (Appendix B). The new Indian Point Unit 2 heatup and cooldown pressure-temperature limit curves were generated using ASME Code Case
N-640[3] (which allows the use of the K_{Ic} methodology) and the axial flaw methodology of the 1995 ASME Code, Section XI through the 1996 Addenda.
It should be noted that Indian Point was limited at the 1/4T location by the intermediate to lower shell circumferential weld and at the 3/4T location by the intermediate shell plate B-2002-3. The pressure-temperature (PT) limit curves presented in Section 5 are those developed using the axial flaw methodology with the most limiting axial flaw adjusted reference temperatures (ARTs). Theses PT curves bound the PT curves that used the ASME Code Case N-588[4] (Circ. Flaw Methodology) with the most limiting Circ Flaw ARTs. The circ. flaw PT curves are presented in Appendix G herein.

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."[5] 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 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 2[6], "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" with exception of the following: 1) **The fluence values used in this report are calculated fluence values, not the best estimate fluence values (See Appendix B).** 2) The K_{Ic} critical stress intensities are used in place of the K_{Ia} critical stress intensities. This methodology is taken from approved ASME Code Case N-640[3]. 3) The 1996 Version of Appendix G to Section XI[7] will be used rather than the 1989 version. 4) PT Curves were generated with the most limiting circumferential weld ART value in conjunction with Code Case N-588[4]. The curves, which are included in Appendix G, are bounded by the curves using the standard "axial" flaw methodology from ASME Code 1996 App. G with the ART from the limiting plate material B-2002-3.

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[8]. The beltline material properties of the Indian Point Unit 2 reactor vessel is presented in Table 1.

Best estimate copper (Cu) and nickel (Ni) weight percent values used to calculate chemistry factors (CF) in accordance with Regulatory Guide 1.99, Revision 2, are provided in Table 1. Additionally, surveillance capsule data is available for four capsules (Capsules V, Z, Y and T) already removed from the Indian Point Unit 2 reactor vessel. This surveillance capsule data was also used to calculate CF values per Position 2.1 of Regulatory Guide 1.99, Revision 2 in Table 4. These CF values are summarized in Table 5. It should be noted that in addition to Indian Point Unit 2, surveillance weld data from Indian Point Unit 3 and H.B. Robinson Unit 2 was used in the determination of CF. In addition, all the surveillance data has been determined to be credible, with exception to surveillance plate B-2002-2.

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 2.

TABLE 1
Summary of the Best Estimate Cu and Ni Weight Percent and Initial RT_{NDT} Values for the Indian Point Unit 2 Reactor Vessel Materials

Material Description	Cu (%)	Ni(%)	Initial RT_{NDT}[a]
Closure Head Flange	---	---	60°F
Vessel Flange	---	---	60°F
Intermediate Shell Plate B-2002-1[e]	0.19 (0.21)	0.65 (0.62)	34°F
Intermediate Shell Plate B-2002-2[e]	0.17 (0.15)	0.46 (0.44)	21°F
Intermediate Shell Plate B-2002-3[e]	0.25 (0.20)	0.60 (0.59)	21°F
Lower Shell Plate B-2003-1	0.20	0.66	20°F
Lower Shell Plate B-2003-2	0.19	0.48	-20°F
Intermediate & Lower Shell Longitudinal Weld Seams (Heat # W5214)[b, d]	0.21	1.01	-56°F
Intermediate to Lower Shell Girth Weld (Heat # 34B009)[c, d]	0.19	1.01	-56°F
Indian Point Unit 2 Surveillance Weld (Heat # W5214)[b, d]	0.20	0.94	---
Indian Point Unit 3 Surveillance Weld (Heat # W5214)[b, d]	0.16	1.12	---
H.B. Robinson Unit 2 Surveillance Weld (Heat # W5214)[b, d]	0.32	0.66	---

Notes:

(a) The Initial RT_{NDT} values are measured values, with exception to the weld materials.

(b) The weld material in the Indian Point Unit 2 surveillance program was made of the same wire and flux as the reactor vessel intermediate shell longitudinal weld seams (Wire Heat No. W5214 RACO3 + Ni200, Flux Type Linde 1092, Flux Lot No. 3600). The lower shell longitudinal weld seam also had the same heat and flux type but different flux lot. Indian Pt. Unit 3 and H.B. Robinson Unit 2 also contain surveillance material of this heat.

(c) The intermediate to lower shell circ. weld material was made of Wire Heat No. 34B009 RACO3 + Ni200, Flux Type Linde 1092, Flux Lot No. 3708).

(d) The weld best estimate copper and nickel weight percents were obtained from CE Reports NPSD-1039, Rev. 2[15] and/or NPSD-1119, Rev. 1[16]. The values from the CE Report NPSD-1119, Rev. 2 for the Indian Point 2 vessel axial and circ. welds matches that in the NRC database RVID2. The values were rounded to two decimal points.

(e) Copper and Nickel Values were obtained from WCAP-12796, which in turn used Southwest Research Report 17-2108 (Capsule V Analysis). This report calculated a best estimate copper/Nickel weight percent excluding values that appeared to be outliers. If all data was considered, then the best estimate would match RVID2. The data above for the intermediate shell plates are conservative with exception to plate B-2002-1. The chemistry for plate B-2002-1 produces a Table chemistry factor of 156.2°F as compared to the chemistry factor calculated using credible surveillance data (114°F, See Tables 4 & 5). Thus, this non-conservative difference versus RVID2 is negligible. Values from WCAP-12796 will be used herein. RVID2 Values are in Parenthesis.

The chemistry factors 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. Position 2.1 uses the surveillance capsule data from all capsules withdrawn to date, including those capsules from Indian Point Unit 3 and H.B. Robinson Unit 2. The fluence values used to determine the CFs in Table 4 are the calculated fluence values at the surveillance capsule locations. Hence, the calculated fluence values were used for all cases.

The measured ΔRT_{NDT} values for the weld data were adjusted for temperature difference between differing plants and for chemistry using the ratio procedure given in Position 2.1 of Regulatory Guide 1.99, Revision 2. See Table 2 for the Tcold operating temperatures at Indian Point Units 2 and 3 and H.B. Robinson Unit 2.

TABLE 2
Inlet (Tcold) Operating Temperatures

Indian Point Unit 2[(a)]	Indian Point Unit 3[(b)]	H.B. Robinson Unit 2[(c)]
543°F (Cycle 1)	540°F (Capsule T)	547°F (Capsule S)
543°F (Cycle 2)	540°F (Capsule Y)	547°F (Capsule T)
522.5°F (Cycle 3)	540°F (Capsule Z)	- - -
522.5°F (Cycle 4)		
522.8°F (Cycle 5)		
522.8°F (Cycle 6)		
522.8°F (Cycle 7)		
522.5°F (Cycle 8)	- - -	- - -
528°F (Average)	540°F (Average)	547°F (Average)

Notes:

(a) Confirmed by Indian Point Unit 2. Average over eight matches E900 Database. Note that cycle 8 is when the last capsule was withdrawn, IP2 is currently in cycle 15.

(b) Per E900 Database. Confirmed by Indian Point Unit 3.

(c) Per E900 Database the value for all Capsules at H.B. Robinson Unit 2 was 546°F, however Ted Huminski at Robinson indicated that the Inlet Operating Temperatures was documented as being between 546°F and 547°F. Thus, for conservatism (i.e. larger delta versus IP2) 547°F will be assumed.

All calculated fluence values (capsule and projections) for Indian Point Unit 2 were updated and documented in Appendix B. These fluences were calculated using the ENDF/B-VI scattering cross-section data set. In addition, capsule fluences from Indian Point Unit 3 and H.B. Robinson Unit 2 are included since they share the same surveillance weld material and can be used in the calculation of chemistry factor. The Indian Point Unit 3 fluences are taken from Letter INT-00-211[9], and the H.B. Robinson fluences were taken from WCAP-14044[10]. The Indian Point Unit 3 fluences are calculated fluences using ENDF/B-VI cross-sections. The best available fluence data for H.B. Robinson are the fluences from WCAP-14044. Calculated fluences exist in WCAP-14044, however they were determined using ENDF/B-IV & V cross-sections and would increase if ENDF/B-VI cross-sections were used. Thus, for conservatism the calculated fluences were increased 15% to account for going to ENDF/B-VI and used herein for the calculation of chemistry factor. It should be noted that the measured fluences would not increase under ENDF/B-VI. Table 3 is a summary of the capsule fluences from Indian Point Unit 2 and 3 and H.B Robinson.

TABLE 3

Calculated Integrated Neutron Exposure of the Surveillance Capsules @ Indian Point Unit 2, Indian Point Unit 3 and H.B. Robinson Unit 2

Capsule	Fluence
Indian Point Unit 2[(a)]	
T	2.53×10^{18} n/cm^2, (E > 1.0 MeV)
Y	4.55×10^{18} n/cm^2, (E > 1.0 MeV)
Z	1.02×10^{19} n/cm^2, (E > 1.0 MeV)
V	4.92×10^{18} n/cm^2, (E > 1.0 MeV)
Indian Point Unit 3[(b)]	
T	2.88×10^{18} n/cm^2, (E > 1.0 MeV)
Y	7.52×10^{18} n/cm^2, (E > 1.0 MeV)
Z	1.12×10^{19} n/cm^2, (E > 1.0 MeV)
H.B. Robinson Unit 2[(c)]	
S	5.80×10^{18} n/cm^2, (E > 1.0 MeV)
V	6.20×10^{18} n/cm^2, (E > 1.0 MeV)
T	4.66×10^{19} n/cm^2, (E > 1.0 MeV)

NOTES:

(a) Per Appendix B.

(b) The fluences are calculated fluences per Letter INT-00-211 using ENDF/B-VI.

(c) The fluences are Calculated values per WCAP-14044 plus 15%.

TABLE 4
Calculation of Chemistry Factors using Indian Point Unit 2 Surveillance Capsule Data

Material	Capsule	Capsule f[a]	FF[b]	ΔRT_{NDT}[c]	$FF*\Delta RT_{NDT}$	FF^2
Intermediate Shell Plate B-2002-1	T	0.253	0.627	55.0	34.49	0.393
	Z	1.02	1.006	125.0	125.75	1.012
				SUM:	160.24	1.405
	$CF_{B\text{-}2002\text{-}1} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (160.24) \div (1.405) =$ **114.0°F**					
Intermediate Shell Plate B-2002-2	T	0.253	0.627	95.0	59.57	0.393
	Z	1.02	1.006	120.0	120.72	1.012
	V	0.492	0.802	77.0	61.75	0.643
				SUM:	242.04	2.048
	$CF_{B\text{-}2002\text{-}2} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (242.04) \div (2.048) =$ **118.2°F**					
Intermediate Shell Plate B-2002-3	T	0.253	0.627	115.0	72.11	0.393
	Y	0.455	0.781	145.0	113.25	0.610
	Z	1.02	1.006	180.0	181.08	1.012
				SUM:	366.44	2.015
	$CF_{B\text{-}2002\text{-}2} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (366.44) \div (2.015) =$ **181.9°F**					
Surveillance Weld Material[d]	Y (IP2)	0.455	0.781	208.65 (195)	162.96	0.610
	V (IP2)	0.492	0.802	218.28 (204)	175.06	0.643
	T (IP3)	0.288	0.660	173.6 (143)	114.58	0.436
	Y (IP3)	0.752	0.920	215.04 (180)	197.84	0.846
	Z (IP3)	1.12	1.03	259.84 (220)	267.64	1.061
	V(HBR2)	0.620	0.866	248.87 (209.32)	215.52	0.750
	T(HBR2)	4.66	1.39	334.72 (288.08)	465.26	1.932
				SUM:	1598.86	6.278
	$CF_{Surv.\ Weld} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (1598.86°F) \div (6.278) =$ **254.7°F**					

See Next Page for Notes

Notes:

(a) f = fluence. See Table 3, (x 10^{19} n/cm^2, E > 1.0 MeV).

(b) FF = fluence factor = $f^{(0.28 - 0.1 * \log f)}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values taken from the following documents:
- Indian Point Unit 2 Plate and Weld...WCAP-12796 (Which Refers back to the Original Southwest Research Institute Report for each Capsule.)
- Indian Point Unit 3 Weld...WCAP-11815[11].
- H.B.Robinson Unit 2...Letter Report CPL-96-203[12]

(d) Per Table 2 Indian Point Unit 3 operates with an inlet temperature of approximately 540°F, H.B. Robinson Unit 2 operates with an inlet temperature of approximately 547°F, and Indian Point Unit 2 operates with an inlet temperature of approximately 528°F. The measured ΔRT_{NDT} values from the Indian Point Unit 3 surveillance program were adjusted by adding 12°F to each measured ΔRT_{NDT} and the H.B. Robinson Unit 2 surveillance program were adjusted by adding 19°F to each measured ΔRT_{NDT} value before applying the ratio procedure. The surveillance weld metal ΔRT_{NDT} values have been adjusted by a ratio factor of:
Ratio IP2 = 230.2 ÷ 215.8 = 1.07 for the Indian Point Unit 2 data.
Ratio IP3 = 230.2 ÷ 206.2 = 1.12 for the Indian Point Unit 3 data.
Ratio HBR2 = 230.2 ÷ 210.7 = 1.09 for the H.B. Robinson Unit 2 data.
(The pre-adjusted values are in parenthesis.)

TABLE 5
Summary of the Indian Point Unit 2 Reactor Vessel Beltline Material Chemistry Factors

Material	Reg. Guide 1.99, Rev. 2 Position 1.1 CF's	Reg. Guide 1.99, Rev. 2 Position 2.1 CF's
Intermediate Shell Plate B-2002-1	144°F	114
Intermediate Shell Plate B-2002-2	115.1°F	118.2
Intermediate Shell Plate B-2002-3	176°F	181.9
Lower Shell Plate B-2003-1	152°F	---
Lower Shell Plate B-2003-2	128.8°F	---
Intermediate & Lower Shell Longitudinal Weld Seams (Heat # W5214)	230.2°F	254.7
Intermediate to Lower Shell Girth Weld Seam (Heat # 34B009)	220.9°F	---
Indian Point Unit 2 Surveillance Weld (Heat # W5214)	214.3°F	---
Indian Point Unit 3 Surveillance Weld (Heat # W5214)	206.2°F	---
H.B. Robinson Unit 2 Surveillance Weld (Heat # W5214)	210.7°F	---

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 Code Case N-640, "Alternative Reference Fracture Toughness for Development of PT Limit Curves for Section XI"[3 & 7] of the ASME Appendix G to Section XI. The K_{Ic} curve is given by the following equation:

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

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 Class1, 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:

$$C^* K_{Im} + K_{It} < K_{Ic} \quad (2)$$

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:

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

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:

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

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 (P-T) limit curves. No other changes were made to the OPERLIM computer code with regard to P-T calculation methodology. Therefore, the P-T curve methodology is unchanged from that described in WCAP-14040, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldwon Limit Curves"[6] 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[13] 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 (3106 psi), which is 621 psig for Indian Point Unit 2. The limiting unirradiated RT_{NDT} of 60°F occurs in both the closure head and vessel flanges of the Indian Point Unit 2 reactor vessel, so the minimum allowable temperature of this region is 180°F at pressures greater than 621 psig. This limit is shown in Figures 5-1 and 5-2 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:

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

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[14]. 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 8.625 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 in Appendix B and are also presented in a condensed version in Table 6 of this report. The evaluation used the ENDF/B-VI scattering cross-section data set. This is consistent with methods presented in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves"[2]. Table 6 contains the **calculated vessel surface fluences values** at various azimuthal locations. Tables 7 and 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 ART values for all beltline materials in the Indian Point Unit 2 reactor vessel.

TABLE 6

Calculated Neutron Fluence Projections at Key Locations on the Reactor Vessel Clad/Base Metal Interface (10^{19} n/cm^2, E > 1.0 MeV)

EFPY	Azimuthal Location			
	0°	15°	30°	45°
8.62[a]	0.145	0.231	0.275	0.416
16.87[b]	0.256	0.415	0.498	0.744
25	0.350	0.553	0.677	1.016
32	0.446	0.690	0.855	1.283
48	0.666	1.004	1.263	1.894

Notes:

(a) Date of last capsule removal.

(b) Current EFPY.

TABLE 7

Summary of the Vessel Surface, 1/4T and 3/4T Fluence Values used for the Generation of the 25 EFPY Heatup/Cooldown Curves

Material	Surface	¼ T[a]	¾ T[a]
Intermediate Shell Plate B-2002-1	1.02×10^{19}	6.08×10^{18}	2.16×10^{18}
Intermediate Shell Plate B-2002-2	1.02×10^{19}	6.08×10^{18}	2.16×10^{18}
Intermediate Shell Plate B-2002-3	1.02×10^{19}	6.08×10^{18}	2.16×10^{18}
Lower Shell Plate B-2003-1	1.02×10^{19}	6.08×10^{18}	2.16×10^{18}
Lower Shell Plate B-2003-2	1.02×10^{19}	6.08×10^{18}	2.16×10^{18}
Intermediate & Lower Shell Longitudinal Welds (Heat # W5214) - 0°, 15° & 30°	6.77×10^{18}	4.03×10^{18}	1.43×10^{18}
Intermediate to Lower Shell Girth Weld (Heat # 34B009)	1.02×10^{19}	6.08×10^{18}	2.16×10^{18}

Note:

(a) 1/4T and 3/4T = $F_{(Surface)} * e^{(-0.24*x)}$, where x is the depth into the vessel wall (i.e. 8.625*0.25 or 0.75)

TABLE 8
Summary of the Calculated Fluence Factors used for the Generation of the 25 EFPY Heatup and Cooldown Curves

Material	1/4T F (n/cm^2 ,E > 1.0 MeV)	1/4T FF	3/4T f (n/cm^2 ,E > 1.0 MeV)	3/4T FF
Intermediate Shell Plate B-2002-1	6.08×10^{18}	0.861	2.16×10^{18}	0.588
Intermediate Shell Plate B-2002-2	6.08×10^{18}	0.861	2.16×10^{18}	0.588
Intermediate Shell Plate B-2002-3	6.08×10^{18}	0.861	2.16×10^{18}	0.588
Lower Shell Plate B-2003-1	6.08×10^{18}	0.861	2.16×10^{18}	0.588
Lower Shell Plate B-2003-2	6.08×10^{18}	0.861	2.16×10^{18}	0.588
Intermediate & Lower Shell Longitudinal Welds (Heat # W5214) - 0°, 15° & 30°	4.03×10^{18}	0.748	1.43×10^{18}	0.492
Intermediate to Lower Shell Girth Weld (Heat # 34B009)	6.08×10^{18}	0.861	2.16×10^{18}	0.588

Margin is calculated as, $M = 2\sqrt{\sigma_i^2 + \sigma_\Delta^2}$. The standard deviation for the initial RT_{NDT} margin term, is σ_i 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 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 9 and 10 are the calculations of the 25 EFPY ART values used for generation of the heatup and cooldown curves.

TABLE 9
Calculation of the ART Values for the 1/4T Location @ 25 EFPY

Material	RG 1.99 R2 Method	CF (°F)	FF	IRT_{NDT}[a] (°F)	ΔRT_{NDT}[b] (°F)	Margin[c] (°F)	ART[d] (°F)
Intermediate Shell Plate B-2002-1	Position 1.1	144	0.861	34	124.0	34	192
	Position 2.1	114.0	0.861	34	98.2	17[e]	149
Intermediate Shell Plate B-2002-2	Position 1.1	115.1	0.861	21	99.1	34	154
	Position 2.1	118.2	0.861	21	101.8	34[e]	157
Intermediate Shell Plate B-2002-3	Position 1.1	176	0.861	21	151.5	34	207
	Position 2.1	181.9	0.861	21	156.6	17[e]	**195**
Lower Shell Plate B-2003-1	Position 1.1	152	0.861	20	130.9	34	185
Lower Shell Plate B-2003-2	Position 1.1	128.8	0.861	-20	110.9	34	125
Intermediate & Lower Shell Long. Welds (Heat # W5214)[c]	Position 1.1	230.2	0.748	-56	172.2	65.5	182
	Position 2.1	254.7	0.748	-56	191.0	44.0[e]	179
Intermediate to Lower Shell Girth Weld (Heat # 34B009)	Position 1.1	220.9	0.861	-56	190.2	65.5	**200**

Notes:
(a) Initial RT_{NDT} values are measured values except for the welds.
(b) $\Delta RT_{NDT} = CF * FF$
(c) $M = 2 *(\sigma_i^2 + \sigma_\Delta^2)^{1/2}$
(d) ART = Initial $RT_{NDT} + \Delta RT_{NDT}$ + Margin (°F)
(e) All surveillance data is credible except for the lower shell plate B-2002-2. For this case a full σ_Δ was used.

TABLE 10
Calculation of the ART Values for the 3/4T Location @ 25 EFPY

Material	RG 1.99 R2 Method	CF (°F)	FF	IRT_{NDT}[a] (°F)	ΔRT_{NDT}[b] (°F)	Margin[c] (°F)	ART[d] (°F)
Intermediate Shell Plate B-2002-1	Position 1.1	144	0.588	34	84.7	34	153
	Position 2.1	114.0	0.588	34	67.0	17[e]	118
Intermediate Shell Plate B-2002-2	Position 1.1	115.1	0.588	21	67.7	34	123
	Position 2.1	118.2	0.588	21	69.5	34[e]	125
Intermediate Shell Plate B-2002-3	Position 1.1	176	0.588	21	103.5	34	159
	Position 2.1	181.9	0.588	21	107.0	17[e]	**145**
Lower Shell Plate B-2003-1	Position 1.1	152	0.588	20	89.4	34	143
Lower Shell Plate B-2003-2	Position 1.1	128.8	0.588	-20	75.7	34	89
Intermediate & Lower Shell Long. Welds (Heat # W5214)[c]	Position 1.1	230.2	0.492	-56	113.3	65.5	123
	Position 2.1	254.7	0.492	-56	125.3	44.0[e]	113
Intermediate to Lower Shell Girth Weld (Heat # 34B009)	Position 1.1	220.9	0.588	-56	130.0	65.5	**140**

Notes:
(a) Initial RT_{NDT} values are measured values except for the welds..
(b) $\Delta RT_{NDT} = CF * FF$
(c) $M = 2 *(\sigma_i^2 + \sigma_\Delta^2)^{1/2}$
(d) ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin (°F)
(e) All surveillance data is credible except for the lower shell plate B-2002-2. For this case a full σ_Δ was used.

The intermediate to lower shell girth weld is the limiting beltline material for the 1/4T location (See Table 9) and the intermediate shell plate B-2002-3 is the limiting beltline material for the 3/4T location (See Table 10). Contained in Table 11 is a summary of the limiting ARTs to be used in the generation of the Indian Point Unit 2 reactor vessel heatup and cooldown curves. Since there are different limiting materials and one of which is a circumferential weld, then two sets of curves will be generated. One set will use the methodology from ASME Code Case N-588 with the limiting circ weld ARTs, while the other will use the methodology from the 1996 ASME Code Section XI, Appendix G with the limiting plate ARTs. The most limiting curves will be presented in Section 5, while the other set will be documented in Appendix G.

TABLE 11
Summary of the Limiting ART Values Used in the
Generation of the Indian Point Unit 2 Heatup/Cooldown Curves

¼ T Limiting ART	**¾ T Limiting ART**
Circ Weld ART	
200	140
Intermediate Shell Plate B-2002-3	
195	145

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 3.0 and 4.0 of this report. This approved methodology is also presented in WCAP-14040-NP-A, Revision 2 with exception to those items discussed in Section 1 of this report.

Figure 1 presents the limiting heatup curves without margins for possible instrumentation errors using heatup rates of 60 and 100°F/hr applicable for the first 25 EFPY. This curve was generated using the1996 ASME Code Section XI, Appendix G with the limiting plate ARTs. It bounds the heatup curves (found in Appendix G) generated using ASME Code Case N-588 with the limiting circ weld ARTs. Figure 2 presents the limiting cooldown curves without margins for possible instrumentation errors using cooldown rates of 0, 20, 40, 60 and 100°F/hr applicable for 25 EFPY. Again, this curve was generated using the1996 ASME Code Section XI, Appendix G with the limiting plate ARTs. It bounds the cooldown curves (found in Appendix G) generated using ASME Code Case N-588 with the limiting circ weld ARTs. Allowable combination of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown in Figures 1 and 2. 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 Figure 1. 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 Code Case N-640[3] (approved in February 1999) 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 13. The pressure-temperature limits for core operation (except for low power physics tests) are that the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and 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 3.0 of this report. For the heatup and cooldown curves without margins for instrumentation errors, the minimum temperature for the in service hydrostatic leak tests for the Indian Point Unit 2 reactor vessel at 25 EFPY is 255°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 1 and 2 define all of the above limits for ensuring prevention of nonductile failure for the Indian Point Unit 2 reactor vessel. The data points used for the heatup and cooldown pressure-temperature limit curves shown in Figures 1 and 2 are presented in Tables 12 and 13. By comparison to the curves and data points in Appendix G, it can be seen that the curves in Figures 1 and 2 bound the curves using code case N-588 with a slightly higher 1/4T ART.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE SHELL PLATE
LIMITING ART VALUES AT 25 EFPY: 1/4T, 195°F
3/4T, 145°F

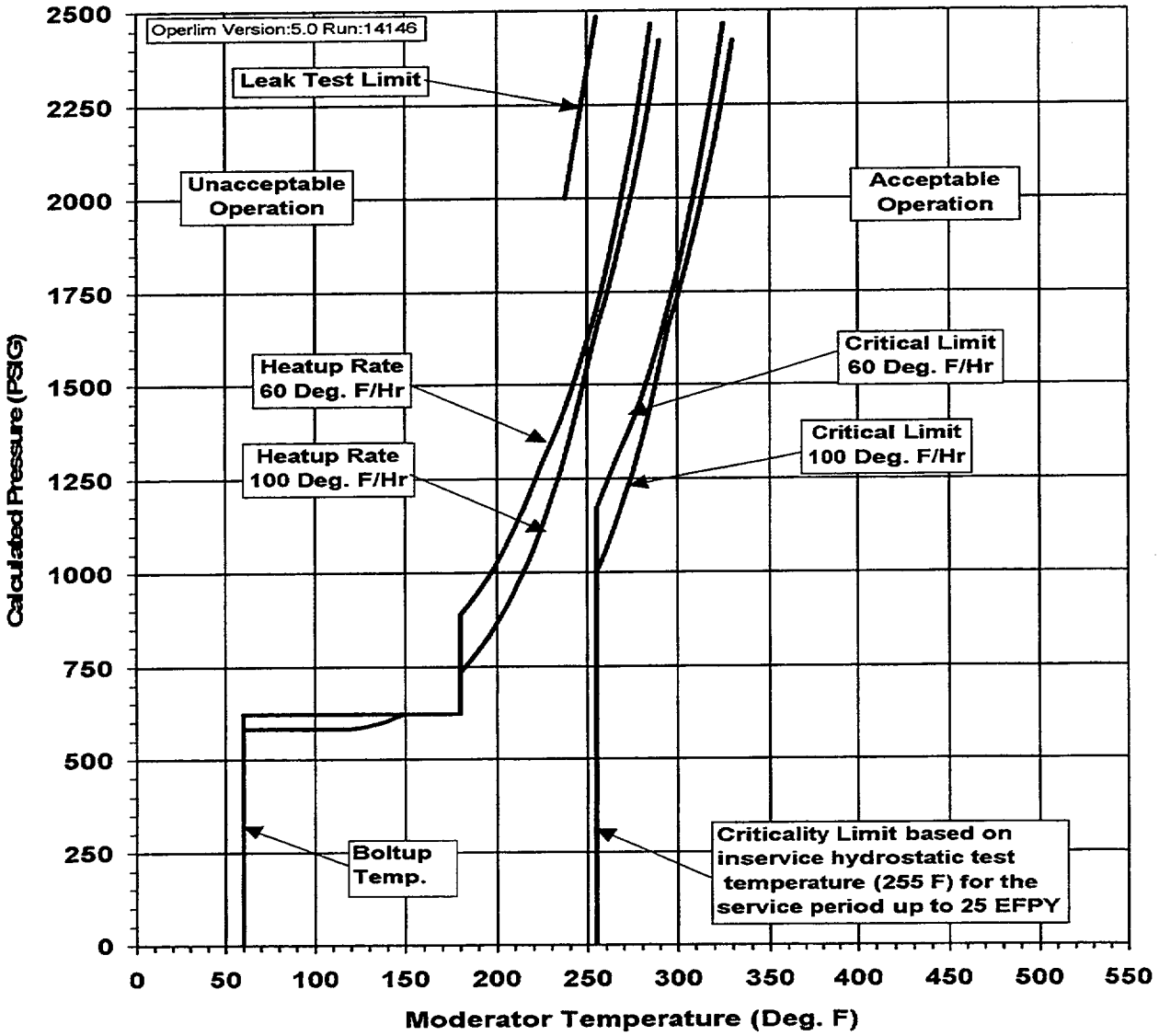

Figure 1 Indian Point Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 60 & 100°F/hr) Applicable for the First 25 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE SHELL PLATE
LIMITING ART VALUES AT 25 EFPY: 1/4T, 195°F
3/4T, 145°F

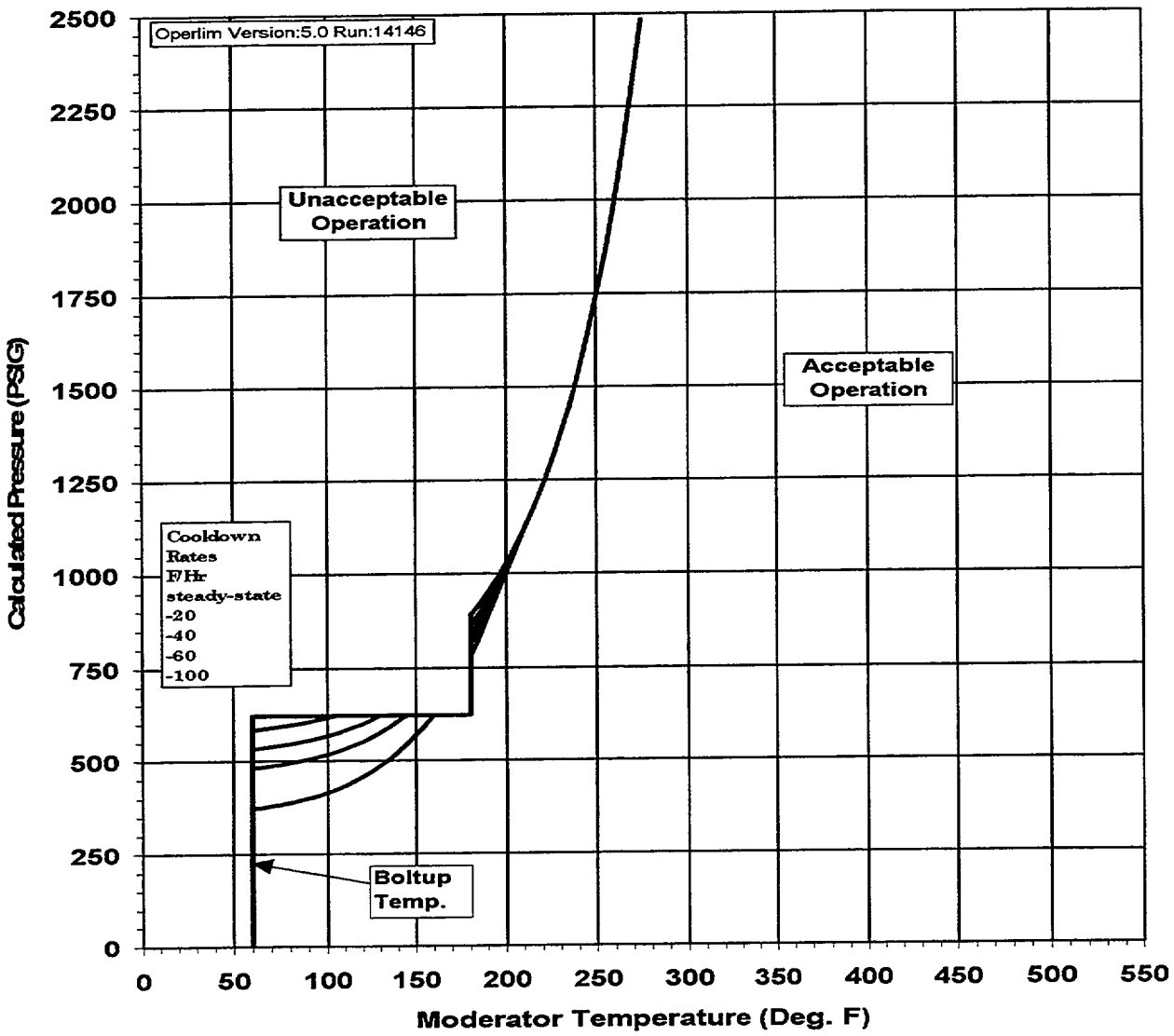

Figure 2 Indian Point Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 25 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology

TABLE 12
25 EFPY Heatup Curve Data Points Using 1996 App. G
(without Uncertainties for Instrumentation Errors)

Heatup		Curves							
60 Heatup		60 Critical	Limit	100 Heatup		100 Critical	Limit	Leak Test	Limit
T	P	T	P	T	P	T	P	T	P
60	0	255	0	60	0	255	0	238	2000
60	621	255	621	60	581	255	581	255	2485
65	621	255	621	65	581	255	581		
70	621	255	621	70	581	255	582		
75	621	255	621	75	581	255	583		
80	621	255	621	80	581	255	585		
85	621	255	621	85	581	255	587		
90	621	255	621	90	581	255	590		
95	621	255	621	95	581	255	592		
100	621	255	621	100	581	255	596		
105	621	255	621	105	581	255	599		
110	621	255	621	110	581	255	603		
115	621	255	621	115	581	255	608		
120	621	255	621	120	582	255	613		
125	621	255	621	125	585	255	620		
130	621	255	621	130	590	255	621		
135	621	255	621	135	596	255	621		
140	621	255	621	140	603	255	621		
145	621	255	621	145	613	255	621		
150	621	255	621	150	621	255	621		
155	621	255	621	155	621	255	621		
160	621	255	621	160	621	255	621		
165	621	255	621	165	621	255	621		
170	621	255	621	170	621	255	621		
175	621	255	621	175	621	255	621		
180	621	255	888	180	621	255	734		
180	621	255	917	180	621	255	762		
180	888	255	950	180	734	255	792		
185	917	255	986	185	762	255	826		
190	950	255	1026	190	792	255	864		
195	986	255	1070	195	826	255	906		
200	1026	255	1119	200	864	255	952		
205	1070	255	1173	205	906	255	1003		

TABLE 12 - (Continued)
25 EFPY Heatup Curve Data Points Using 1996 App. G
(without Uncertainties for Instrumentation Errors)

Heatup		Curves					
60 Heatup		60 Critical Limit		100 Heatup		100 Critical Limit	
T	P	T	P	T	P	T	P
210	1119	260	1232	210	952	260	1060
215	1173	265	1291	215	1003	265	1123
220	1232	270	1345	220	1060	270	1192
225	1291	275	1405	225	1123	275	1269
230	1345	280	1470	230	1192	280	1353
235	1405	285	1543	235	1269	285	1447
240	1470	290	1622	240	1353	290	1550
245	1543	295	1711	245	1447	295	1657
250	1622	300	1808	250	1550	300	1737
255	1711	305	1915	255	1657	305	1826
260	1808	310	2033	260	1737	310	1923
265	1915	315	2163	265	1826	315	2030
270	2033	320	2307	270	1923	320	2148
275	2163	325	2466	275	2030	325	2278
280	2307			280	2148	330	2422
285	2466			285	2278		
				290	2422		

TABLE 13
25 EFPY Cooldown Curve Data Points Using 1996 App. G
(without Uncertainties for Instrumentation Errors)

Cooldown Curves									
Steady State		20F		40F		60F		100F	
T	P	T	P	T	P	T	P	T	P
60	0	60	0	60	0	60	0	60	0
60	621	60	583	60	532	60	480	60	373
65	621	65	586	65	535	65	483	65	376
70	621	70	589	70	538	70	486	70	380
75	621	75	592	75	542	75	490	75	384
80	621	80	596	80	546	80	494	80	389
85	621	85	600	85	550	85	499	85	394
90	621	90	605	90	555	90	504	90	400
95	621	95	610	95	560	95	510	95	407
100	621	100	615	100	566	100	517	100	415
105	621	105	621	105	573	105	524	105	424
110	621	110	621	110	581	110	532	110	434
115	621	115	621	115	589	115	541	115	445
120	621	120	621	120	599	120	552	120	457
125	621	125	621	125	609	125	563	125	471
130	621	130	621	130	621	130	576	130	486
135	621	135	621	135	621	135	590	135	504
140	621	140	621	140	621	140	606	140	523
145	621	145	621	145	621	145	621	145	544
150	621	150	621	150	621	150	621	150	568
155	621	155	621	155	621	155	621	155	595
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	888	180	860	180	835	180	812	180	779
185	917	185	893	185	871	185	852	185	828
190	950	190	929	190	911	190	897	190	884
195	986	195	969	195	955	195	946	195	945
200	1026	200	1013	200	1004	200	1000		
205	1070	205	1062	205	1058				
210	1119	210	1115						

TABLE 13 – (Continued)
25 EFPY Cooldown Curve Data Points Using 1996 App. G
(without Uncertainties for Instrumentation Errors)

Cooldown Curves									
Steady State		20F		40F		60F		100F	
T	P	T	P	T	P	T	P	T	P
215	1173								
220	1232								
225	1298								
230	1370								
235	1451								
240	1540								
245	1638								
250	1746								
255	1866								
260	1998								
265	2144								
270	2306								
275	2485								

6 REFERENCES

1. Southwest Research Final Report, SwRI Project 17-2108, "Reactor Vessel Material Surveillance Program for Indian Point Unit 2: Analysis of Capsule V", March 1990.

2. WCAP-12796, "Heatup and Cooldown Limit Curves for the Consolidated Edison Company Indian Point Unit 2 Reactor Vessel", N. K. Ray, January 1991.

3. ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1", February 26, 1999.

4. ASME Boiler and Pressure Vessel Code, Case N-588, "Attenuation to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels", Section XI, Division 1, Approved December 12, 1997.

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

6. WCAP-14040-NP-A, Revision 2, "Methodology used to Develop Cold Overpressure Mitigating system Setpoints and RCS Heatup and Cooldown Limit Curves", J.D. Andrachek, et. al., January 1996.

7. Section XI of the ASME Boiler and Pressure Vessel Code, Appendix G, "Fracture Toughness Criteria for Protection Against Failure." Dated December 1995, through 1996 Addendum.

8. "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.

9. INT-00-211, "Evaluation of Reactor Vessel Flux and Fluence Calculations", R.R. Laubham, April 25, 2000.

10. WCAP-14044, "Westinghouse surveillance Capsule Neutron Fluence Re-evaluation", E.P Lippencott, April 1994.

11. WCAP-11815, "Analysis of Capsule Z from the New York Power Authority Indian Point Unit 3 Reactor Vessel Radiation Surveillance Program", S. E. Yanichko, et. al., March, 1988.

12. CPL-96-203, "Robinson Unit 2 Surveillance Capsule Charpy Test Results", P. A. Grendys, March 6, 1996.

13. 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.

14. 1989 Section III, Division 1 of the ASME Boiler and Pressure Vessel Code, Paragraph NB-2331, "Material for Vessels."

15. CE Report NPSD-1039, Revision 2, "Best Estimate Copper and Nickel Values in CE Fabricated Reactor Vessel Welds", CEOG Task 902, By the CE Owners Group. June 1997.

16. CE Report NPSD-1119, Revision 1, "Updated Analysis for Combustion Engineering Fabricated Reactor Vessel Welds Best Estimate Copper and Nickel Content", CEOG Task 1054, By the CE Owners Group. July 1998.

17. WOG CalcNote 92-016 (Westinghouse File # WOG-108/4-18), "WOG USE Program – Onset of Upper Shelf Energy Calculations", J.M. Chicots, 3/8/93. [Note: This calcnote used the original Combustion Engineering CMTRs]

APPENDIX A

PRESSURIZED THERMAL SHOCK (PTS) RESULTS

PTS Calculations:

The PTS Rule requires that for each pressurized water nuclear power reactor for which an operating license has been issued, the licensee shall have projected values of RT_{PTS} , accepted by the NRC, for each reactor vessel beltline material for the EOL fluence of the material. This assessment must specify the basis for the projected value of RT_{PTS} for each vessel beltline material, including the assumptions regarding core loading patterns, and must specify the copper and nickel contents and the fluence value used in the calculation. This assessment must be updated whenever there is a significant change in projected values of RT_{PTS}, or upon request for a change in the expiration date for operation of the facility. (Changes to RT_{PTS} values are considered significant if either the previous value or the current value, or both values, exceed the screening criterion prior to the expiration of the operating license, including any renewed term, if applicable, for the plant.

To verify that RT_{NDT} , for each vessel beltline material is a bounding value for the specific reactor vessel, licensees shall consider plant-specific information that could affect the level of embrittlement. This information includes but is not limited to the reactor vessel operating temperature and any related surveillance program results. (Surveillance program results mean any data that demonstrates the embrittlement trends for the limiting beltline material, including but not limited to data from test reactors or from surveillance programs at other plants with or without surveillance program integrated per 10 CFR Part 50, Appendix H.)

Calculations:

Table A-1 contains the results of the calculations for each of the beltline region materials in the Indian Point Unit 2 Reactor Vessel. Per ConEd, the actual EOL is less than 32 EFPY, however for conservatism EOL will be assumed to be 32 EFPY.

TABLE A-1

RT_{PTS} Calculations for Indian Point Unit 2 Beltline Region Materials at 32 EFPY

Material	Fluence (n/cm^2, E>1.0 MeV)	FF	CF (°F)	ΔRT_{PTS} [c] (°F)	Margin (°F)	$RT_{NDT(U)}$ [a] (°F)	RT_{PTS} [b] (°F)
Inter. Shell Plate B-2002-1	1.28×10^{19}	1.07	144	154.1	34	34	222
- Using S/C Data	1.28×10^{19}	1.07	114	122.0	17	34	173
Inter. Shell Plate B-2002-2	1.28×10^{19}	1.07	115.1	123.2	34	21	178
- Using S/C Data	1.28×10^{19}	1.07	118.2	126.5	34	21	182
Inter. Shell Plate B-2002-3	1.28×10^{19}	1.07	176	188.3	34	21	243
- Using S/C Data	1.28×10^{19}	1.07	181.9	194.6	17	21	233
Lower Shell Plate B-2003-1	1.28×10^{19}	1.07	152	162.6	34	20	217
Lower Shell Plate B-2003-2	1.28×10^{19}	1.07	142	151.9	34	-20	166
Intermediate & Lower Shell Long. Welds (Heat # W5214)	8.55×10^{18}	0.956	230.2	220.1	65.5	-56	230
- Using S/C Data	8.55×10^{18}	0.956	254.7	243.5	44.0	-56	232
Intermediate to Lower Shell Girth Weld (Heat # 34B009)	1.28×10^{19}	1.07	220.9	236.4	65.5	-56	**246**

Notes:

(a) Initial RT_{NDT} values are measured values

(b) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS}$ + Margin (°F)

(c) ΔRT_{PTS} = CF * FF

All of the beltline materials in the Indian Point Unit 2 reactor vessel are below the screening criteria values of 270°F and 300°F at 32 EFPY.

APPENDIX B

CALCULATED FLUENCE DATA

Neutron Fluence Calculations

Discrete ordinates transport calculations were performed on a fuel cycle specific basis to determine the neutron environment within the reactor geometry of Indian Point Unit 2. The specific calculational methods applied are consistent with those described in WCAP-15557, "Qualification of the Westinghouse Pressure Vessel Neutron Fluence Evaluation Methodology"[1] and in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," January 1996.[2]

In the application of this methodology to the fast neutron exposure evaluations for the Indian Point Unit 2 surveillance capsules and reactor vessel, plant specific forward transport calculations were carried out using the following three-dimensional flux synthesis technique:

$$\phi(r,\theta,z) = [\phi(r,\theta)] * [\phi(r,z)]/[\phi(r)]$$

where $\phi(r,\theta,z)$ is the synthesized three-dimensional neutron flux distribution, $\phi(r,\theta)$ is the transport solution in r,θ geometry, $\phi(r,z)$ is the two-dimensional solution for a cylindrical reactor model using the actual axial core power distribution, and $\phi(r)$ is the one-dimensional solution for a cylindrical reactor model using the same source per unit height as that used in the r,θ two-dimensional calculation.

For this analysis, all of the transport calculations were carried out using the DORT discrete ordinates code Version 3.1[3] and the BUGLE-96 cross-section library[4]. The BUGLE-96 library provides a 67 group coupled neutron-gamma ray cross-section data set produced specifically for light water reactor application. In these analyses, anisotropic scattering was treated with a P_5 legendre expansion and the angular discretization was modeled with an S_{16} order of angular quadrature. Energy and space dependent core power distributions as well as system operating temperatures were treated on a fuel cycle specific basis.

Results of the discrete ordinates calculations performed for Indian Point Unit 2 are provided in Tables 1 through 3. In Table 1, the calculated neutron exposures for the four surveillance capsules withdrawn to date are given in terms of both fast neutron (E > 1.0 MeV) fluence and iron atom displacements (dpa). The maximum neutron exposure of the pressure vessel at the clad/base metal interface is provided for several azimuthal angles in Table 2. Again, calculated exposure data are listed for both fluence (E > 1.0 MeV) and dpa. Calculated lead factors associated with each of the Indian Point Unit 2 surveillance capsules are listed in Table 3.

Following the completion of the plant specific transport analyses, the calculated results were compared with available measurements in order to demonstrate that the differences between calculations and measurements support the 20% (1σ) uncertainty required by Draft Regulatory Guide DG-1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence".[5] Two levels of comparison of calculation with measurement were made to demonstrate compliance with the requirements of DG-1053. In the first instance, ratios of measured and calculated sensor reaction rates (M/C) were compared for all fast neutron sensors contained in the surveillance capsules withdrawn to date. In the second case, comparisons of calculated and least squares adjusted best estimate values of neutron fluence (E > 1.0 MeV) and dpa were examined.

The M/C comparisons of individual sensor reaction rates showed consistent behavior for all reactions at all capsule locations within the constraint of an allowable 20% (1σ) uncertainty in the final calculated results. The overall average M/C ratio for the entire 13 sample data set was 1.07 with an associated standard deviation of 9.2%. The observed M/C ratios for twelve of the 13 samples ranged from 0.87 to 1.16 with the remaining sample [$^{63}Cu(n,\alpha)^{60}Co$ reaction] exhibiting an M/C ratio of 1.22. This data set of M/C ratios from the Indian Point Unit 2 surveillance capsules indicates that the ± 20% acceptance criterion specified in DG-1053[3] has been met by the current neutron transport calculations.

The corresponding best estimate to calculation (BE/C) comparisons for neutron fluence (E > 1.0 MeV) spanned a range of 0.948 to 1.056 with an average BE/C ratio of 1.017 ± 1.4% (1σ). Likewise, in the case of iron atom displacements, the BE/C ratios spanned a range of 0.947 to 1.043 with an average BE/C of 1.008 ± 4.2% (1σ). These comparisons also fall well within the ± 20% criterion specified in DG-1053, thus supporting the validation of the current calculations for applicability for the Indian Point Unit 2 reactor.

Appendix B References:

1. S. L. Anderson, "Qualification of the Westinghouse Pressure Vessel Neutron Fluence Evaluation Methodology," WCAP-15557-R0, August 2000.
2. J. D. Andrachek, et al., "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," WCAP-14040-NP-A, Revision 2, January 1996.
3. RSICC Computer Code Collection CCC-650, "DOORS3.1 One-, Two-, and Three- Dimensional Discrete Ordinates Neutron/Photon Transport Code System," Radiation Shielding Information Center, Oak Ridge National Laboratory, August 1996.
4. RSIC Data Library Collection DLC-185, "BUGLE-96 Coupled 47 Neutron, 20 Gamma-Ray Group Cross Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications," Radiation Shielding Information Center, Oak Ridge National Laboratory, March 1996.
5. Draft Regulatory Guide DG-1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," U. S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, September 1999.

Table B-1
Summary of Calculated Surveillance Capsule Exposure Evaluations

Capsule	Irradiation Time [efpy]	Fluence (E > 1.0 MeV) [n/cm^2]	Iron Displacements [dpa]
T	1.42	2.53e+18	4.26e-03
Y	2.34	4.55e+18	7.68e-03
Z	5.17	1.02e+19	1.72e-02
V	8.62	4.92e+18	7.91e-03

Table 2
Summary of Calculated Maximum Pressure Vessel Exposure
Clad/Base Metal Interface

Irradiation Time [efpy]	Neutron Fluence (E > 1.0 MeV) [n/cm^2]			
	0.0 Degrees	15.0 Degrees	30.0 degrees	45.0 Degrees
16.87 (EOC 14)	2.556e+18	4.152e+18	4.975e+18	7.443e+18
18.66 (EOC 15)	2.764e+18	4.453e+18	5.368e+18	8.038e+18
25.00	3.505e+18	5.526e+18	6.766e+18	1.016e+19
32.00	4.464e+18	6.900e+18	8.551e+18	1.283e+19
48.00	6.657e+18	1.004e+19	1.263e+19	1.894e+19

Irradiation Time [efpy]	Iron Atom Displacements [dpa]			
	0.0 Degrees	15.0 Degrees	30.0 degrees	45.0 Degrees
16.87 (EOC 14)	4.140e-03	6.635e-03	8.011e-03	1.200e-02
18.66 (EOC 15)	4.476e-03	7.117e-03	8.643e-03	1.295e-02
25.00	5.884e-03	9.115e-03	1.125e-02	1.687e-02
32.00	7.438e-03	1.132e-02	1.413e-02	2.118e-02
48.00	1.099e-02	1.636e-02	2.070e-02	3.105e-02

Table 3
Calculated Surveillance Capsule Lead Factors

Capsule ID And Location	Status	Lead Factor
T(40°)	Withdrawn EOC 1	3.43
Y(40°)	Withdrawn EOC 2	3.48
Z(40°)	Withdrawn EOC 5	3.53
V(4°)	Withdrawn EOC 8	1.18
S(40°)	In Reactor	3.5
U(4°)	In Reactor	1.2
W(4°)	In Reactor	1.2
X(4°)	In Reactor	1.2

APPENDIX C

UPDATED SURVEILLANCE MATERIAL 30 FT-LB TRANSITION TEMPERATURE SHIFTS AND UPPER SHELF ENERGY DECREASES

TABLE C-1

Measured 30 ft-lb Transition Temperature Shifts of all Available Surveillance Data

Material	Capsule	Fluence ($x 10^{19}$ n/cm^2)	30 ft-lb Transition Temperature Shift		Upper Shelf Energy Decrease	
			Predicted (°F) [a]	Measured (°F) [b]	Predicted (%) [a]	Measured (%) [c]
Intermediate Shell Plate B-2002-1	T	2.53×10^{18}	90.29	55.0	21	16
	Z	1.02×10^{19}	144.86	125.0	29	21
Intermediate Shell Plate B-2002-2	T	2.53×10^{18}	72.17	95.0	19	17
	Z	1.02×10^{19}	115.79	120.0	26	23
	V	4.92×10^{18}	92.31	77.0	22	4
Intermediate Shell Plate B-2002-3	T	2.53×10^{18}	110.35	115.0	25	20
	Y	4.55×10^{18}	137.46	145.0	28	28
	Z	1.02×10^{19}	177.06	180.0	34	28
Surv. Program Weld Metal	Y	4.55×10^{18}	167.37	195.0	28	45
	V	4.92×10^{18}	171.87	204.0	29	38
Heat Affected Zone Material	Y	4.55×10^{18}	--	165	---	13
	V	4.92×10^{18}	--	150	---	0
Correlation Monitor Material	T	2.53×10^{18}	--	75	---	0
	Y	4.55×10^{18}	--	70	---	6
	Z	1.02×10^{19}	--	102	---	15
	V	4.92×10^{18}	--	100	---	0

Notes:

(a) Based on Regulatory Guide 1.99, Revision 2, methodology using the mean weight percent values of copper and nickel of the surveillance material.

(b) Calculated using measured Charpy data.

(c) Values are based on the definition of upper shelf energy given in ASTM E185-82.

APPENDIX D

REACTOR VESSEL BELTLINE MATERIAL PROJECTED END OF LICENSE UPPER SHELF ENERGY VALUES

TABLE D-1
Predicted End-of-License (32 EFPY) USE Calculations for all the Beltline Region Materials

Material	Weight % of Cu	1/4T EOL Fluence (10^{19} n/cm^2)	Unirradiated USE[(a)] (ft-lb)	Projected USE Decrease (%)	Projected EOL USE (ft-lb)
Intermediate Shell Plate B-2002-1	0.19	0.763	70	20	56
Intermediate Shell Plate B-2002-2	0.17	0.763	73	21	58
Intermediate Shell Plate B-2002-3	0.25	0.763	74	32	50.3
Lower Shell Plate B-2003-1	0.20	0.763	71	27	52
Lower Shell Plate B-2003-2	0.19	0.763	88	27	64
Intermediate & Lower Shell Longitudinal Welds (Heat # W5214)	0.21	0.510	121	43	69
Intermediate to Lower Shell Girth Weld (Heat # 34B009)	0.19	0.763	82[(b)]	32	56

Notes:

(a) These values were obtained from Reference 17. Values reported in the NRC Database RVID2 are identical with exception to Intermediate Shell Plates B-2002-1, 2. RVID2 reported the initial USE as 76 and 75. This evaluation conservatively used the lower values of 70 and 73.

(b) Value was obtained from the average of three impacts tests (71, 84, 90) at 10°F performed for the original material certification.

APPENDIX E

UPDATED SURVEILLANCE CAPSULE REMOVAL SCHEDULE

Withdrawal Schedule To Be Provided in PTLR Only by Indian Point Unit 2

APPENDIX F

ENABLE TEMPERATURE CALCULATIONS AND RESULTS

Enable Temperature Calculation:

ASME Section XI, Appendix G requires the low temperature overpressure (LTOP or COMS) system to be in operation at coolant temperatures less than 200°F or at coolant temperatures less than a temperature corresponding to a reactor vessel metal temperature less than RT_{NDT} + 50°F, whichever is greater. RT_{NDT} is the highest adjusted reference temperature (ART) for the limiting beltline material at a distance one fourth of the vessel section thickness from the vessel inside surface (ie. clad/base metal interface), as determined by Regulatory Guide 1.99, Revision 2.

32 EFPY

The highest calculated 1/4T ART for the Indian Point Unit 2 reactor vessel beltline region at 25 EFPY is 200°F.

From the OPERLIM computer code output for the Indian Point Unit 2 25 EFPY P-T limit curves without margins (Configuration # 14146 & 22915) the maximum ΔT_{metal} is:

Cooldown Rate (Steady-State Cooldown):
max (ΔT_{metal}) at 1/4T = 0°F

Heatup Rate of 100°F/Hr:
max (ΔT_{metal}) at 1/4T = 30.084°F

Enable Temperature (ENBT) = RT_{NDT} + 50 + max (ΔT_{metal}), °F
= (200 + 50 + 30.084) °F
= 280.084°F

The minimum required enable temperature for the Indian Point Unit 2 Reactor Vessel is **280°F** at 25 EFPY of operation.

APPENDIX G

PRESSURE TEMPERATURE LIMIT CURVES USING CODE CASE N-588

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE TO LOWER SHELL GIRTH WELD
LIMITING ART VALUES AT 25 EFPY: 1/4T, 200°F
3/4T, 140°F

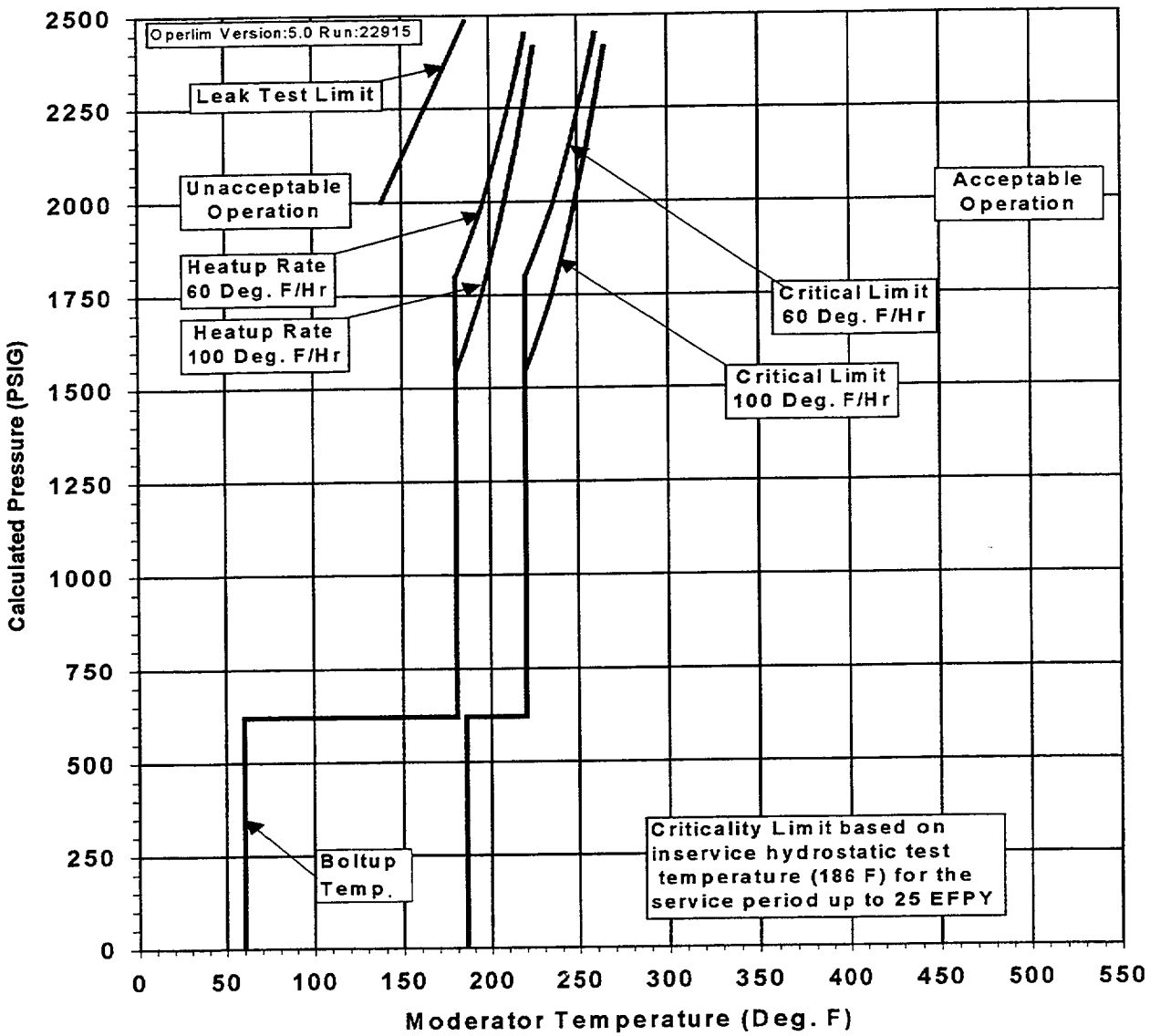

Figure G-1 Indian Point Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 60°F/hr) Applicable for the First 25 EFPY (Without Margins for Instrumentation Errors) Using Code Case N-588

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE TO LOWER SHELL GIRTH WELD
LIMITING ART VALUES AT 25 EFPY: 1/4T, 200°F
3/4T, 140°F

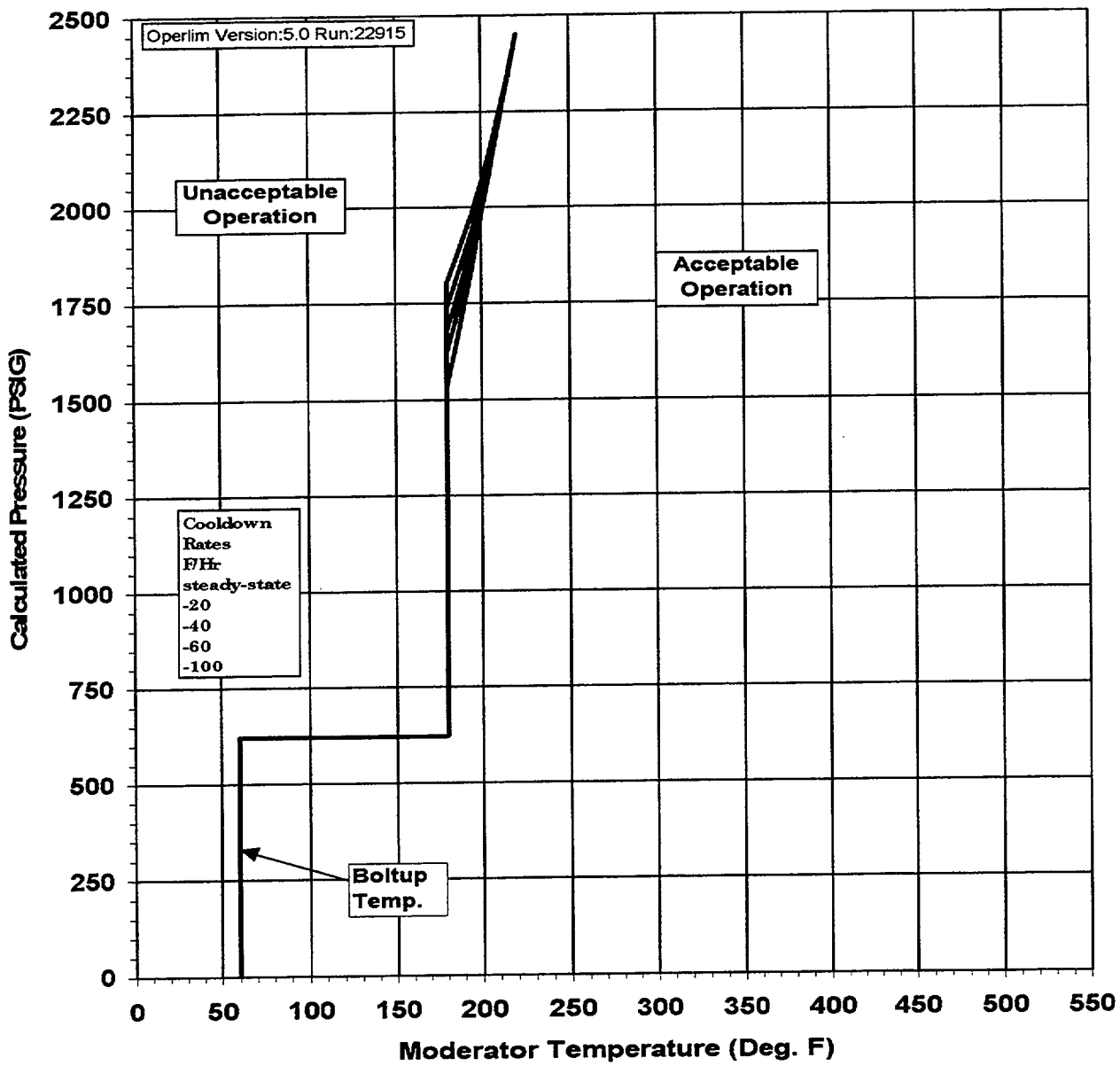

Figure G-2 Indian Point Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 25 EFPY (Without Margins for Instrumentation Errors) Using Code Case N-588

TABLE G-1
25 EFPY Heatup Curve Data Points Using Code Case N-588
(without Uncertainties for Instrumentation Errors)

Heatup		Curves							
60 Heatup		60 Critical	Limit	100 Heatup		100 Critical	Limit	Leak Test	Limit
T	P	T	P	T	P	T	P	T	P
60	0	186	0	60	0	186	0	138	2000
60	621	186	620	60	621	186	620	186	2485
65	621	186	620	65	621	186	620		
70	621	186	620	70	621	186	620		
75	621	186	620	75	621	186	620		
80	621	186	620	80	621	186	620		
85	621	186	620	85	621	186	620		
90	621	186	620	90	621	186	620		
95	621	186	620	95	621	186	620		
100	621	186	620	100	621	186	620		
105	621	186	620	105	621	186	620		
110	621	186	620	110	621	186	620		
115	621	186	620	115	621	186	620		
120	621	186	620	120	621	186	620		
125	621	186	620	125	621	186	620		
130	621	186	620	130	621	186	620		
135	621	186	620	135	621	186	620		
140	621	186	620	140	621	186	620		
145	621	190	620	145	621	190	620		
150	621	195	620	150	621	195	620		
155	621	200	620	155	621	200	620		
160	621	205	620	160	621	205	620		
165	621	210	620	165	621	210	620		
170	621	215	620	170	621	215	620		
175	621	220	620	175	621	220	620		
180	621	220	1800	180	621	220	1545		
180	621	225	1856	180	621	225	1606		
180	1800	230	1918	180	1545	230	1675		
185	1856	235	1986	185	1606	235	1751		
190	1918	240	2061	190	1675	240	1835		
195	1986	245	2145	195	1751	245	1929		
200	2061	250	2237	200	1835	250	2032		
205	2145	255	2339	205	1929	255	2147		
210	2237	260	2451	210	2032	260	2274		
215	2339			215	2147	265	2414		
220	2451			220	2274				
				225	2414				

TABLE G-2
25 EFPY Cooldown Curve Data Points Using Code Case N-588
(without Uncertainties for Instrumentation Errors)

Cooldown Curves									
Steady State		20F		40F		60F		100F	
T	P	T	P	T	P	T	P	T	P
60	0	60	0	60	0	60	0	60	0
60	621	60	621	60	621	60	621	60	621
65	621	65	621	65	621	65	621	65	621
70	621	70	621	70	621	70	621	70	621
75	621	75	621	75	621	75	621	75	621
80	621	80	621	80	621	80	621	80	621
85	621	85	621	85	621	85	621	85	621
90	621	90	621	90	621	90	621	90	621
95	621	95	621	95	621	95	621	95	621
100	621	100	621	100	621	100	621	100	621
105	621	105	621	105	621	105	621	105	621
110	621	110	621	110	621	110	621	110	621
115	621	115	621	115	621	115	621	115	621
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	621	180	621	180	621	180	621	180	621
180	1800	180	1736	180	1676	180	1621	180	1532
185	1856	185	1798	185	1744	185	1697	185	1627
190	1918	190	1866	190	1820	190	1781	190	1731
195	1986	195	1941	195	1903	195	1874	195	1847
200	2061	200	2025	200	1996	200	1977	200	1975
205	2145	205	2117	205	2098	205	2091		
210	2237	210	2219	210	2211				
215	2339	215	2332						
220	2451								