UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 17, 2009

Vice President, Operations
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
Buchanan, NY 10511-0249

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 - ISSUANCE OF AMENDMENT RE: REACTOR VESSEL UPDATED HEATUP AND COOLDOWN CURVES (TAC NO. ME0788)

Dear Sir or Madam:

The Commission has issued the enclosed Amendment No. 262 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated March 5, 2009, as supplemented by letters dated April 17 and June 22, 2009.

The amendment revises the TSs by updating the reactor vessel heatup and cooldown limit curves and the low-temperature over-pressure protection curves.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly *Federal Register* notice.

Sincerely,

John P. Boska

John P. Boska, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-247

Enclosures:
1. Amendment No. 262 to DPR-26
2. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENTERGY NUCLEAR INDIAN POINT 2, LLC

ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-247

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 262
License No. DPR-26

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Entergy Nuclear Operations, Inc. (the licensee) dated March 5, 2009, as supplemented on April 17 and June 22, 2009, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-26 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 262, are hereby incorporated in the license. ENO shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Douglas V Pickett for

Nancy L. Salgado, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the License and
Technical Specifications

Date of Issuance: August 17, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 262

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Replace the following page of the License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove Page	Insert Page
3	3

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
3.4.3-3	3.4.3-3
3.4.3-4	3.4.3-4
3.4.12-8	3.4.12-8
3.4.12-9	3.4.12.-9
3.4.12-10	3.4.12-10
3.4.12-11	3.4.12-11
3.4.12-12	3.4.12-12
3.4.12-13	3.4.12-13

instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) ENO pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; Amdt. 42 10-17-78

(5) ENO pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility. Amdt. 220 09-06-01

C. This amended license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

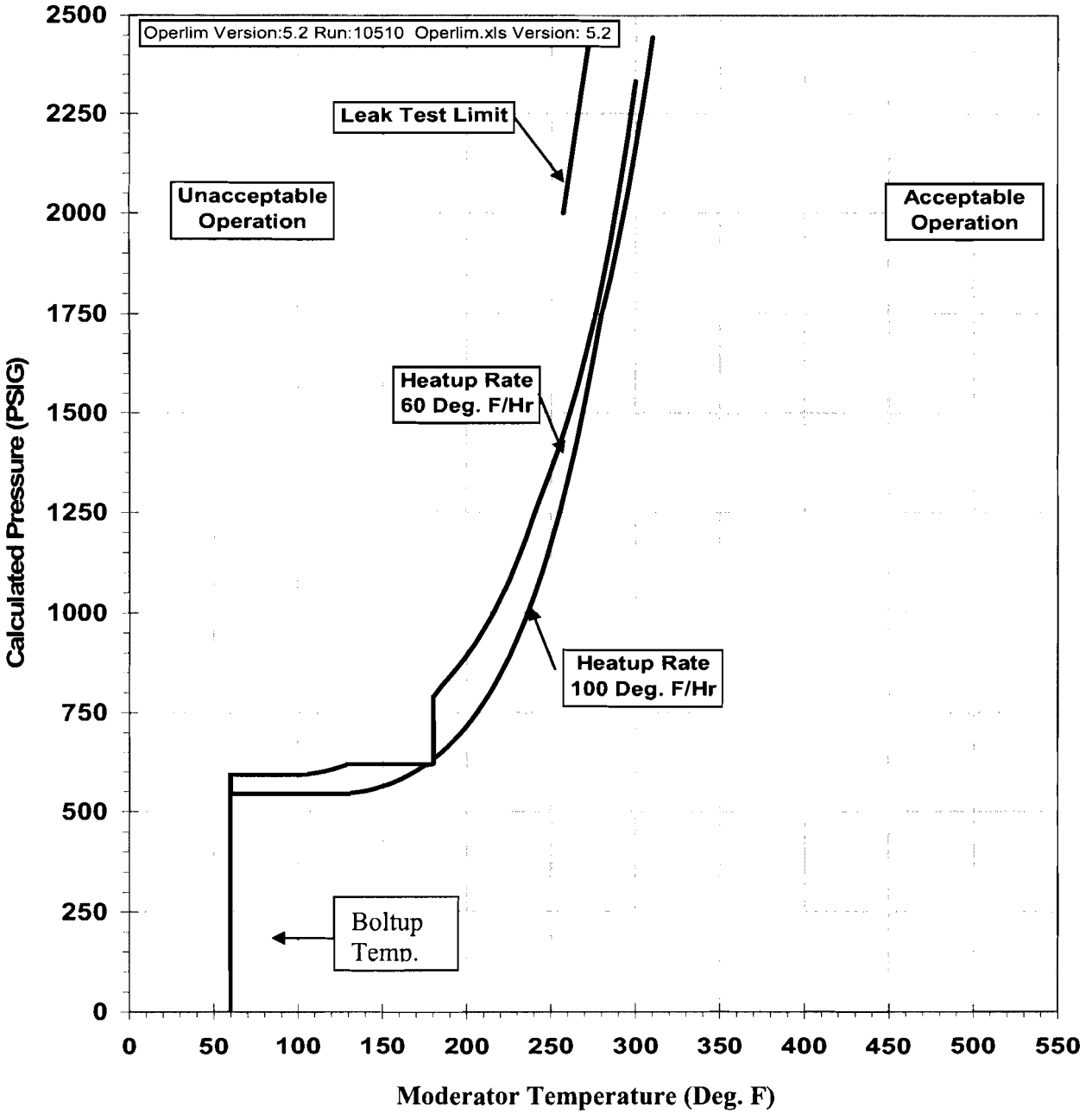
ENO is authorized to operate the facility at steady state reactor core power levels not in excess of 3216 megawatts thermal. Amdt. 241 10-27-04

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 262, are hereby incorporated in the license. ENO shall operate the facility in accordance with the Technical Specifications.

(3) The following conditions relate to the amendment approving the conversion to Improved Standard Technical Specifications:

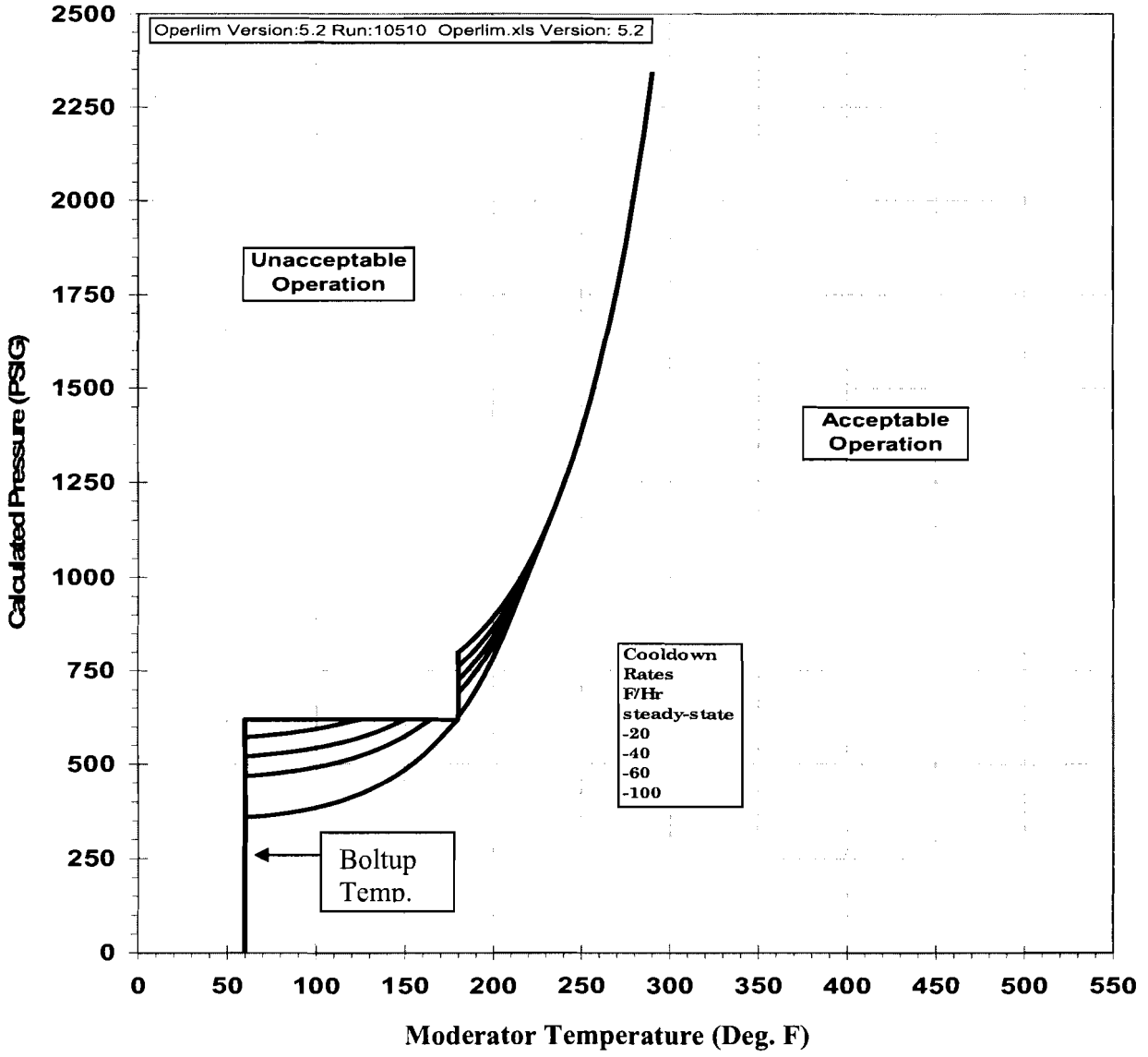
1. This amendment authorizes the relocation of certain Technical Specification requirements and detailed information to licensee-controlled documents as described in Table R, "Relocated Technical Specifications from the CTS," and Table LA, "Removed Details and Less Restrictive Administrative Changes to the CTS" attached to the NRC staff's Safety Evaluation enclosed with this amendment. The relocation of requirements and detailed information shall be completed on or before the implementation of this amendment.



-Notes-

1. Acceptable operation is to the right of or below the applicable curve. Unacceptable operation is to the left of or above the applicable curve.
2. Figure 3.4.3-1 is effective until 29.2 effective full power years (EFPYs)
3. Figure 3.4.3-1 does not include any allowance for instrument uncertainty.

Figure 3.4.3-1:
Heatup Limitations for the Reactor Coolant System (RCS) and
Hydrostatic and Inservice Leak Testing Limitations for the RCS.



-Notes-

1. Acceptable operation is to the right of or below the applicable curve. Unacceptable operation is to the left of or above the applicable curve.
2. Figure 3.4.3-2 is effective until 29.2 effective full power years (EFPYs).
3. Figure 3.4.3-2 does not include any allowance for instrument uncertainty.

Figure 3.4.3-2:
Cooldown Limitations for the RCS (including RCS cooldown following RCS inservice leak and hydrostatic testing)

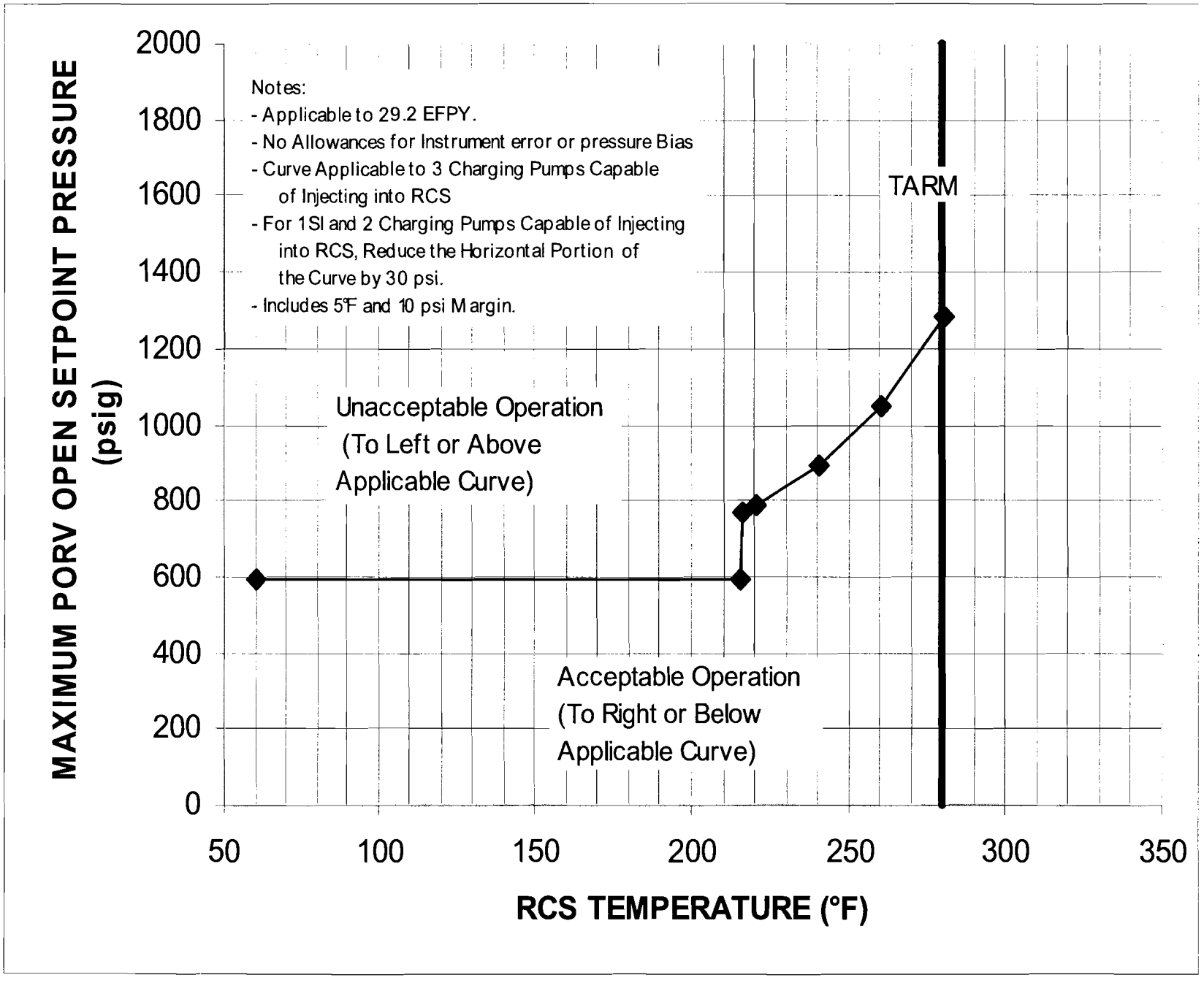


Figure 3.4.12-1: Maximum PORV Opening Setpoint as a Function of Temperature

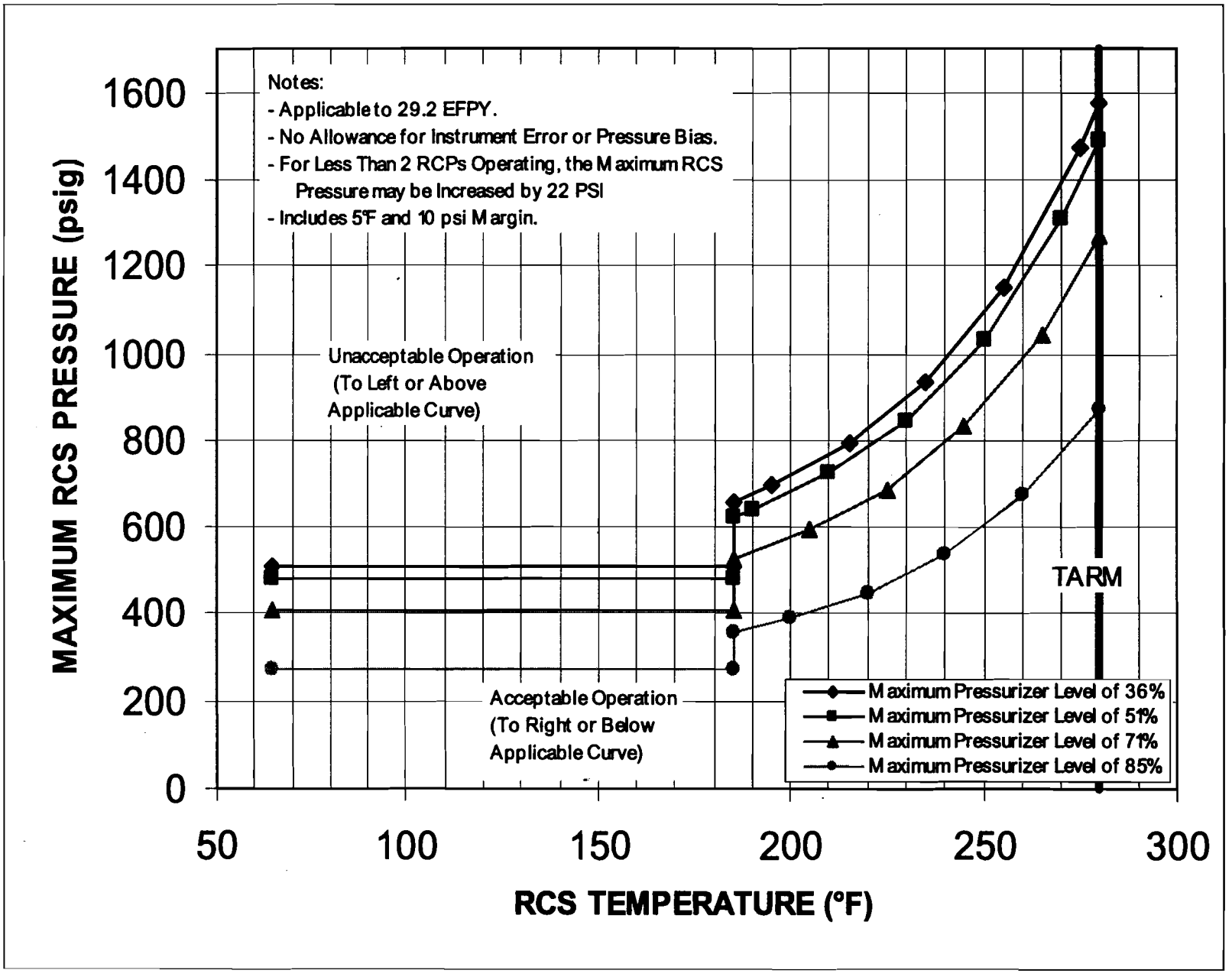


Figure 3.4.12-2: Maximum RCS Pressure - PORVs Inoperable and 1 Charging Pump Capable of Injecting into the RCS.

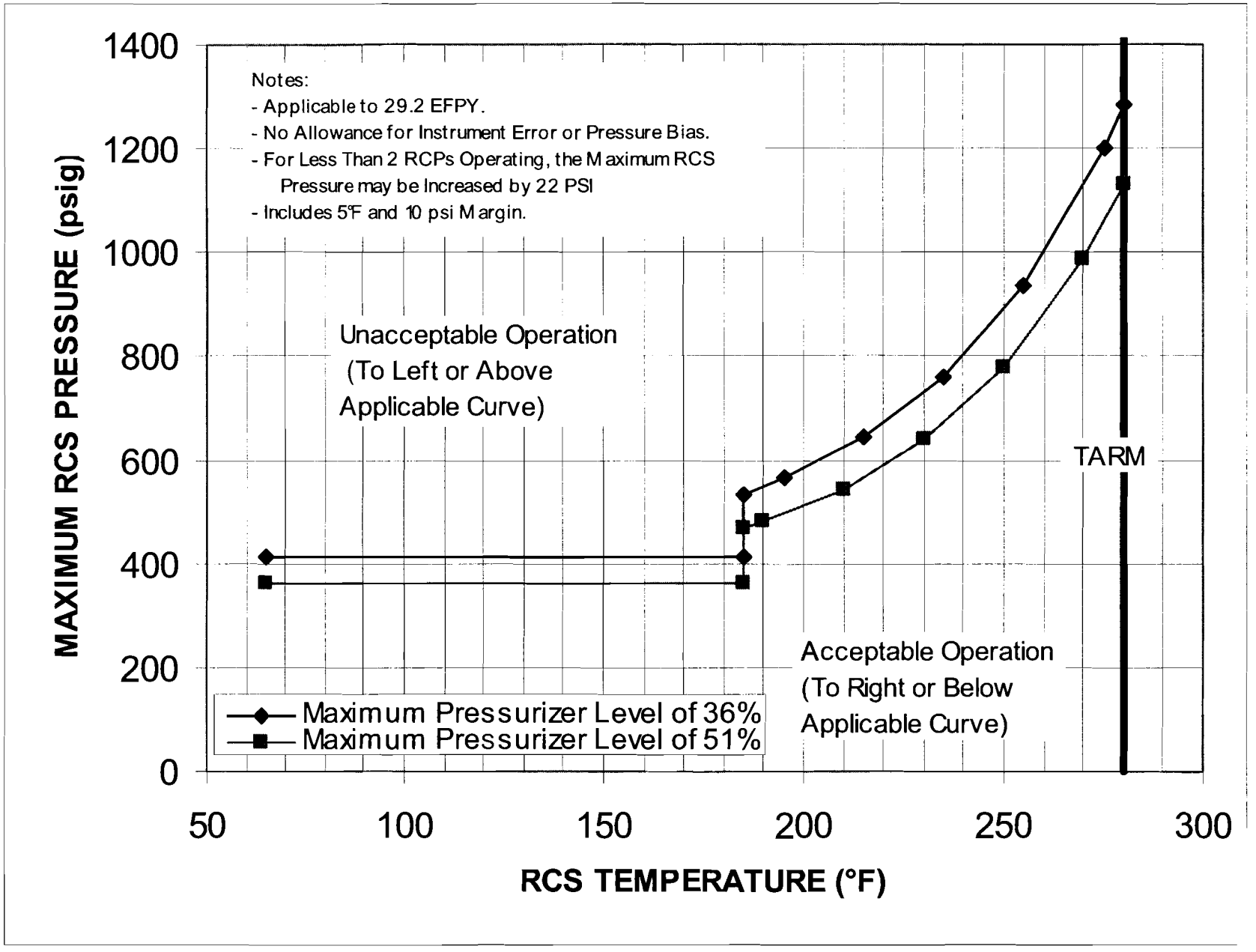


Figure 3.4.12-3: Maximum RCS Pressure - PORVs Inoperable and 2 Charging Pumps Capable of Injecting into the RCS.

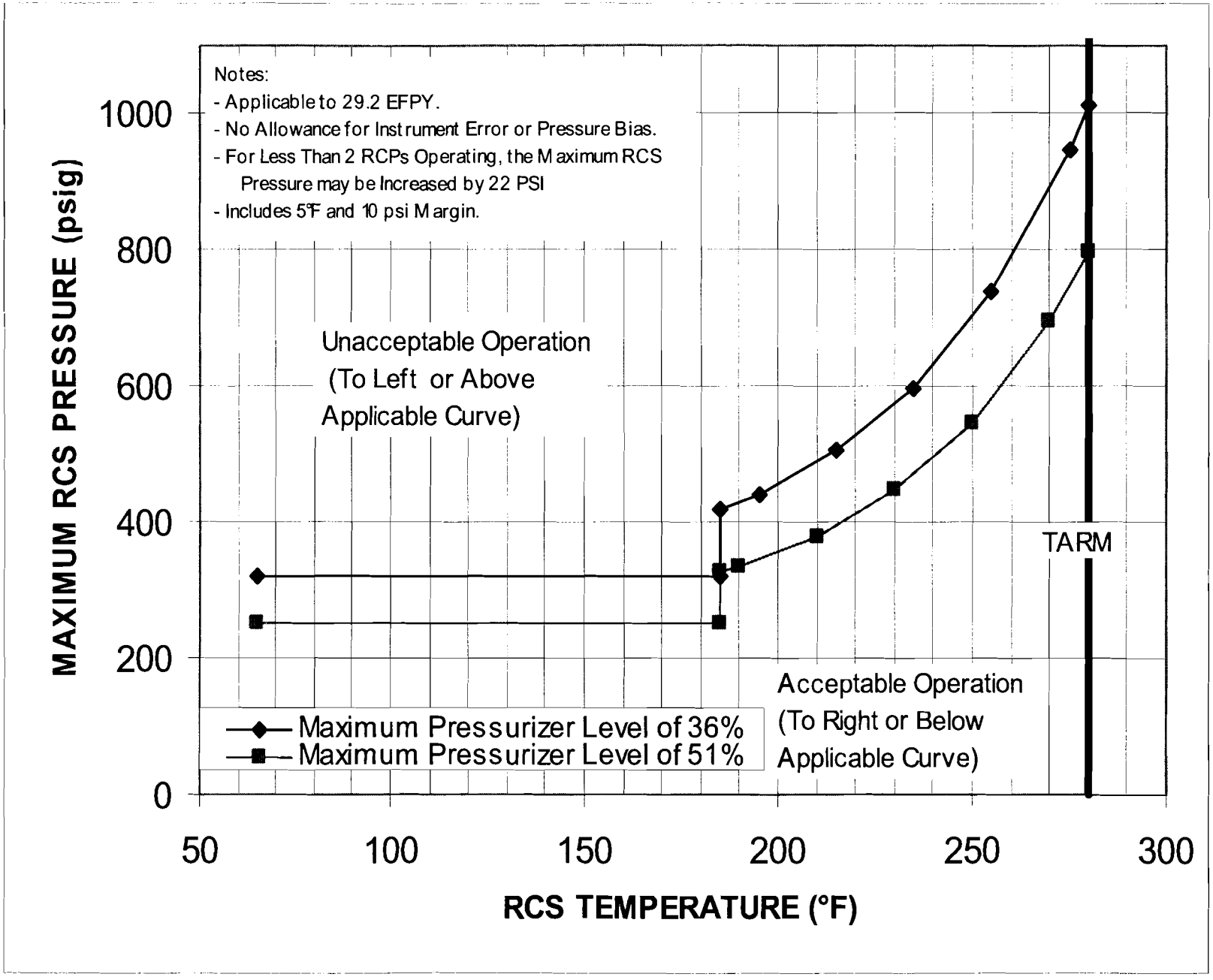


Figure 3.4.12-4: Maximum RCS Pressure - PORVs Inoperable and 3 Charging Pumps Capable of Injecting into the RCS.

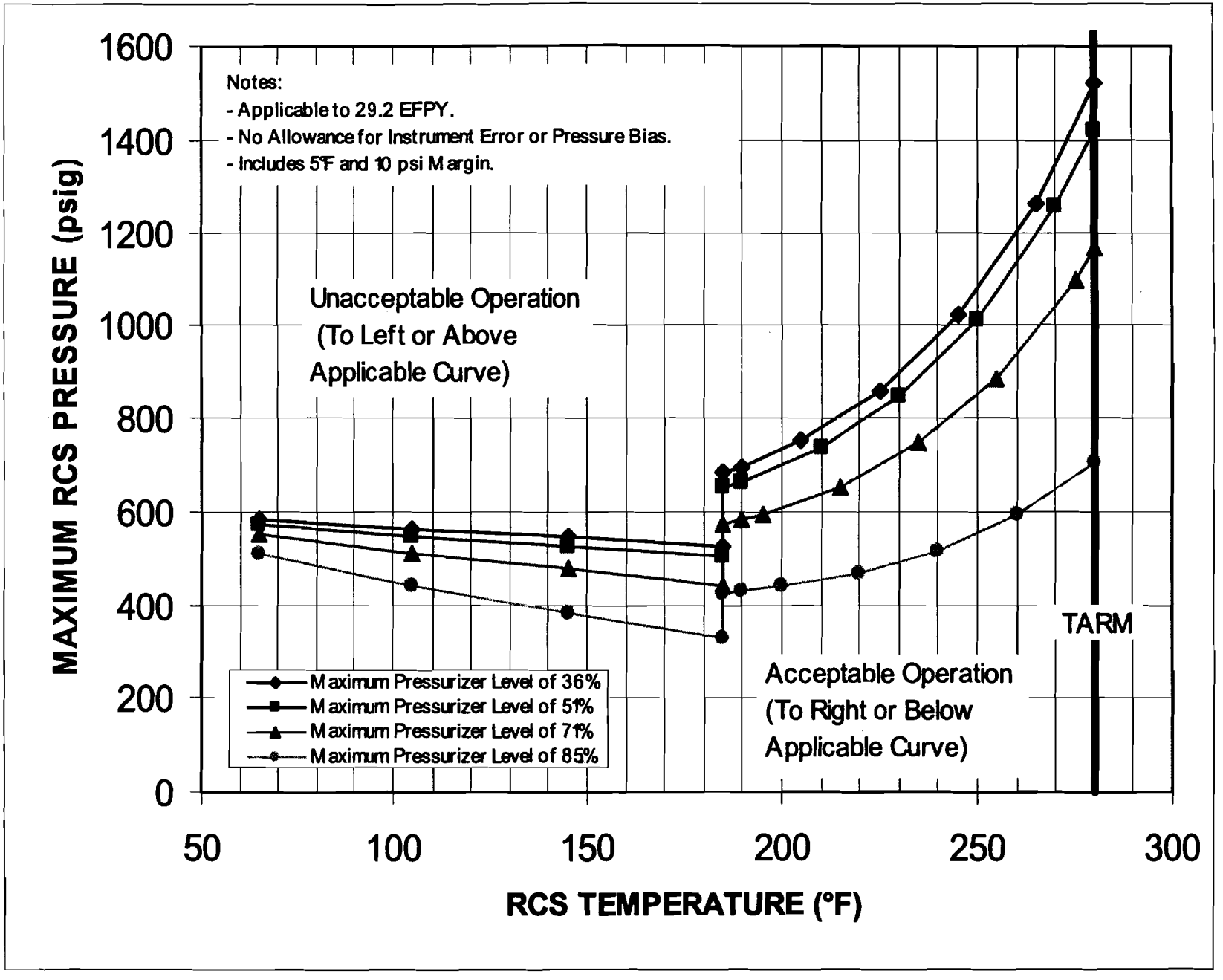


Figure 3.4.12-5: Maximum RCS Pressure and Pressurizer Level during Reactor Coolant Pump Start with PORVs Inoperable and SGs ≤ 40°F Hotter than RCS.

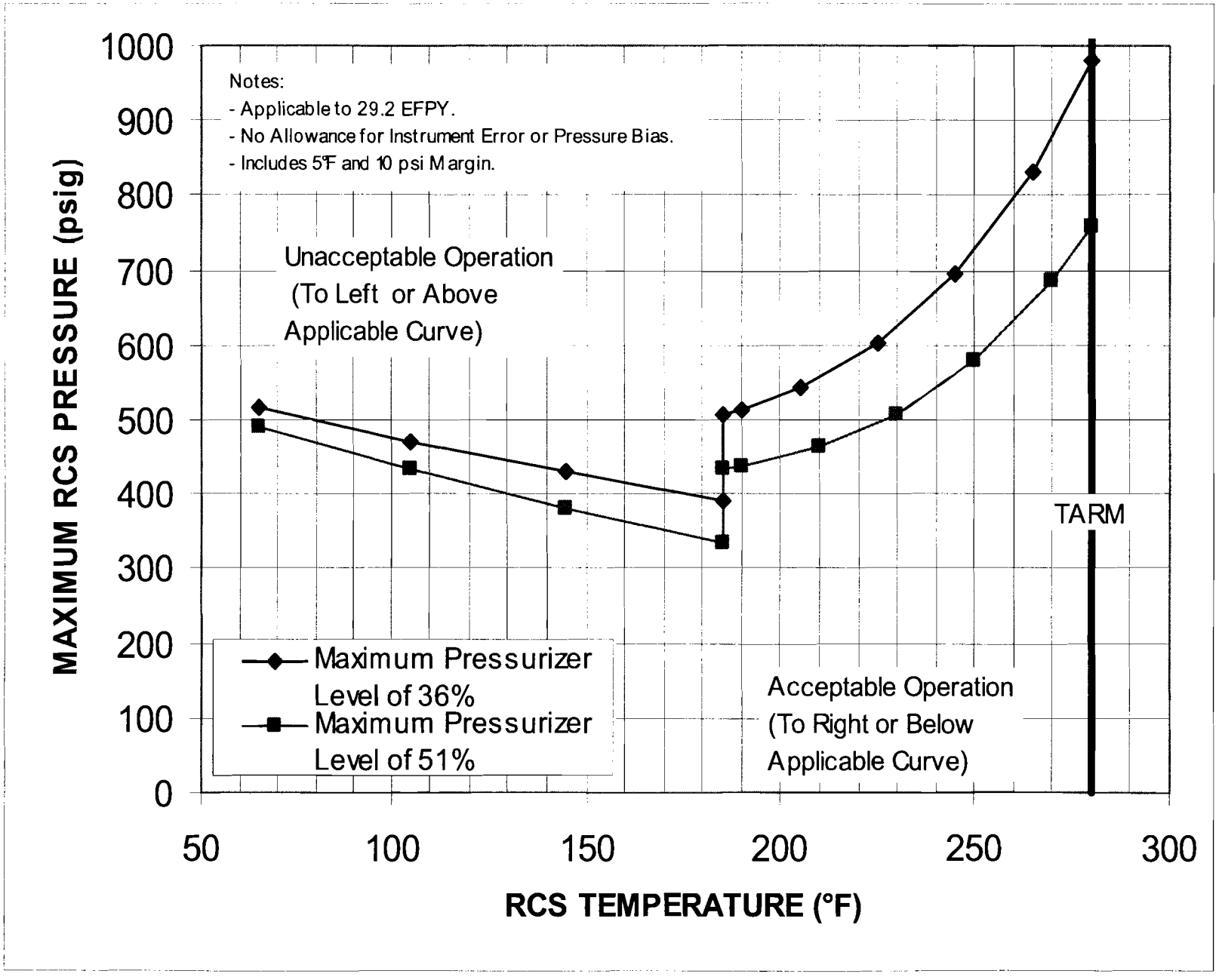


Figure 3.4.12-6: Maximum RCS Pressure and Pressurizer Level during Reactor Coolant Pump Start with PORVs Inoperable and SGs ≤ 100°F Hotter than RCS.

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 262 TO FACILITY OPERATING LICENSE NO. DPR-26

ENTERGY NUCLEAR OPERATIONS, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

1.0 INTRODUCTION

By letter dated March 5, 2009, Agencywide Documents Access and Management System (ADAMS) Accession No. ML090760649, as supplemented by letters dated April 17 and June 22, 2009, ADAMS Accession Nos. ML091180475 and ML091820326, Entergy Nuclear Operations, Inc. (Entergy or the licensee) submitted a request for changes to the Indian Point Nuclear Generating Unit No. 2 (IP2) Technical Specifications (TSs). The supplements dated April 17 and June 22 provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration. The proposed changes would revise the TSs by updating the reactor pressure vessel (RPV) heatup and cooldown limit curves and the low-temperature over-pressure protection (LTOP) curves. The current curves are valid to 25 effective full-power years (EFPY) of reactor operation. IP2 will exceed 25 EFPY later in 2009. The revised curves are effective to 29.2 EFPY.

2.0 REGULATORY EVALUATION

The NRC has established requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The NRC staff evaluates the RPV pressure-temperature (P-T) limit curves for RPV heatup and cooldown based on the following NRC regulations and guidance: 10 CFR Part 50, Appendix G; Generic Letter (GL) 88-11; GL 92-01, Revision 1; GL 92-01, Revision 1, Supplement 1; Regulatory Guide (RG) 1.99, Revision 2 (Rev. 2); and NUREG-0800 Standard Review Plan (SRP) Section 5.3.2. Appendix G to 10 CFR Part 50 requires that P-T limit curves be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). Appendix G to 10 CFR Part 50 also provides minimum temperature requirements that must be considered in the development of the P-T limit curves. GL 88-11 advised licensees that the NRC staff would use RG 1.99, Rev. 2 to review P-T limit curves. RG 1.99, Rev. 2 contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy (USE) resulting from neutron radiation. GL 92-01, Rev. 1 requested that licensees submit their RPV materials property data for their plants to the staff for review. GL 92-01, Rev. 1, Supplement 1 requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations.

SRP Section 5.3.2 provides an acceptable method of determining the P-T limit curves for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics methodology of Appendix G to Section XI of the ASME Code. The basic parameter of this methodology is the stress intensity factor K_I, which is a function of the stress state and flaw configuration. ASME Code, Section XI, Appendix G requires a safety factor of 2.0 on stress intensities resulting from reactor pressure during normal and transient operating conditions, and a safety factor of 1.5 on these stress intensities for hydrostatic testing curves. The flaw postulated in the ASME Code, Section XI, Appendix G has a depth that is equal to 1/4 of the RPV beltline thickness and a length equal to 1.5 times the RPV beltline thickness. The critical locations in the RPV beltline region for calculating heatup and cooldown P-T limit curves are the 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively. The methodology found in Appendix G to Section XI of the ASME Code requires that licensees determine the adjusted reference temperature (ART or adjusted RT_{NDT}) by evaluating material property changes due to neutron radiation. The ART is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}) and a margin (M) term. The ΔRT_{NDT} is a product of a chemistry factor (CF) and a fluence factor. The CF is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Rev. 2, or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the CF was determined using the tables in RG 1.99, Rev. 2, or surveillance data. The margin term is used to account for uncertainties in the values of the initial RT_{NDT}, the copper and nickel contents, the neutron fluence and the calculational procedures. RG 1.99, Rev. 2, describes the methodology to be used in calculating the margin term.

3.0 TECHNICAL EVALUATION

3.1 Licensee's Evaluation

The proposed P-T limit curves in the licensee's letter dated March 5, 2009, are based on the current P-T limit curves which were approved for 25 EFPY in an NRC letter dated February 15, 2002 (ADAMS Accession No. ML020420477). These curves are based on the latest available RPV information and updated neutron fluence calculations, which include the impact of a stretch power uprate (SPU). In a safety evaluation, dated October 27, 2004 (ADAMS Accession No. ML045020362), the NRC staff approved an SPU to increase the licensed rated power by 3.26 percent.

The licensee generated new heatup and cooldown curves using the "axial-flaw" methodology of the 1998 ASME Code, Section XI, Section G, through the 2000 Addenda (which allows the use of the K_{Ic} methodology). The licensee determined that the limiting ART that bounds the IP2 RPV is from the intermediate shell plate B-2002-3 for the axial-flaw methodology and circumferential weld wire heat number 34B009 for the circ-flaw methodology. The critical parameters for the licensee's ART determination for these materials are shown in the table below at the 1/4T and 3/4T locations.

Material	Location	Initial RT_{NDT} (°F)	Fluence at Inside Surface (n/cm^2) (E>1 MeV)	Fluence at Location (n/cm^2) (E>1 MeV)	Chemistry Factor [1] (°F)	ΔRT_{NDT} (°F)	Margin [2] (°F)	ART (°F)
Intermediate Shell Plate B-2002-3	1/4T	21	1.189×10^{19}	7.087×10^{18}	176 (181.9)	159 (164.3)	34 (17)	214 (202)
Intermediate to Lower Shell Girth Weld Heat 34B0009	1/4T	-56	1.189×10^{19}	7.087×10^{18}	220.9	73	65.5	209
Intermediate Shell Plate B-2002-3	3/4T	21	1.189×10^{19}	2.517×10^{18}	110 (113.8)	159 (164.3)	34 (17)	165 (152)
Intermediate to Lower Shell Girth Weld Heat 34B0009	3/4T	-56	1.189×10^{19}	2.517×10^{18}	220.9	73	65.5	148

[1] Chemistry factors were determined using Regulatory Guide 1.99, Rev. 2, Position 1.1.
[2] The margin term for each ART calculation was based on the establishment of initial material property uncertainty (σ_I) and shift in material property uncertainty (σ_Δ) consistent with the guidance in Regulatory Guide 1.99, Rev. 2.
NOTE: Values in parentheses reflect calculations using surveillance capsule data.

The TS changes submitted by the March 5, 2009, letter include:

- Modified P-T limit curves, extended to 29.2 EFPY.
- LTOP system setpoints and LTOP system T_{enable} values to reflect the extended period to 29.2 EFPY.

3.2 NRC Staff's Evaluation

3.2.1 ART Value and P-T Limit Curves

To assess the validity of the licensee's proposed curves, the NRC staff performed an independent assessment of the licensee's submittal. The NRC staff first performed an independent calculation of the ART values for the limiting material using the methodology in RG 1.99, Revision 2. Based on these calculations, the NRC staff verified that the licensee's limiting material in the axial-flaw case is Intermediate Shell Plate B-2002-3, and Circumferential Weld 34B009 is the limiting material in the circ-flaw case. The NRC staff's calculated ART values were in good agreement with the licensee's calculated ART values of 214 °F and 165 °F for the

1/4T location and the 3/4T location, respectively for Intermediate Shell Plate B-2002-3, and ART values of 209 °F and 148 °F for the 1/4T location and the 3/4T location, respectively for Circumferential Weld 34B009.

The NRC staff then evaluated the licensee's P-T limit curves for acceptability by performing independent calculations using the methodologies of Appendix G of Section XI of the ASME Code and 10 CFR Part 50, Appendix G. The licensee stated that the proposed P-T limit curves were based on the methodologies of Appendix G of Section XI of the ASME Code, 1998 Edition with the 2000 Addenda, which utilizes an alternative reference fracture toughness (K_{IC}) curve instead of the K_{Ia} fracture toughness curve for RPV materials in determining the P-T limit curves. NRC Regulatory Issue Summary (RIS) 2004-04, "Use of Code Cases N-588, N-640, and N-641 in Developing Pressure-Temperature Operating Limits," dated April 5, 2004 (ADAMS Accession No. ML040920323), states that the ASME Code, Section XI, Appendix G, 1998 Edition with the 2000 Addenda, may be used without the need for exemption. The use of the K_{IC} fracture toughness curve is appropriate for evaluating the potential for crack initiation without imposing unnecessary conservatism. The K_{IC} curve appropriately implements the use of static initiation fracture toughness behavior to evaluate the controlled heatup and cooldown process of an RPV.

The proposed P-T limit curves in the March 5, 2009, letter modified the P-T limit curves using the approved methodology presented in WCAP-14040-NP-A, Revision 4. Limiting heatup curves used heatup rates of 60 °F/hr and 100 °F/hr applicable for 29.2 EFPY. The limiting cooldown curve used cooldown rates of 0, 20, 40, 60 and 100 °F/hr applicable for 29.2 EFPY. Limiting heatup and cooldown curves were generated using the 1998 ASME Code, Section XI, Appendix G. The NRC staff determined that the minimum temperature requirements of Table 1 of Appendix G to 10 CFR Part 50 were properly implemented in the P-T limit curves. Therefore, the NRC staff verified that the licensee's proposed P-T limits are in accordance with Appendix G to Section XI of the ASME Code and satisfy the requirements of Appendix G to 10 CFR Part 50.

3.2.2 Neutron Fluence

Issues related to RPV neutron fluence calculations were discussed in a safety evaluation (SE) dated May 18, 2009 (ADAMS Accession No. ML0913501511), by the Office of Nuclear Reactor Regulation's Reactor Systems Branch (SRXB), Division of Safety Systems (DSS). In the SE, SRXB determined that since the neutron fluence methodology adheres to the guidance of RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," dated March 2001, the neutron fluence values are acceptable for the purpose of developing P-T limit curves for IP2.

3.2.3 LTOP

The LTOP system, provided by the pressurizer power operated relief valves (PORVs), ensures reactor coolant system (RCS) overpressurization below certain temperatures would be prevented, thus maintaining reactor coolant pressure boundary integrity. The LTOP analysis yields Limiting Conditions for Operation (LCOs) that constitute LTOP system alignments for the period of applicability.

The NRC staff reviewed the LTOP analysis using the guidance contained in Branch Technical Position (BTP) 5-2, "Overpressurization Protection of Pressurized Water Reactors While Operating at Low Temperatures," of NUREG-0800, "Standard Review Plan for the Review of

Safety Analysis Reports for Nuclear Power Plants." Among other things unaffected by this license amendment request, BTP 5-2 recommends assuming single equipment failures and the analysis of multiple transients to establish the limiting transient. The licensee has followed the methodologies utilized in the development of the current P-T limits and LTOP requirements (for 25 EFPY). These methodologies were reviewed and accepted by NRC staff in the SE for TS Amendment No. 224, dated February 15, 2002 (ADAMS Accession No. ML020420477). The revised LTOP system setpoint, and associated LTOP system curves, reflect the increase in neutron fluence for a service life increase from 25 EFPY to 29.2 EFPY. The PORV opening setpoint is implemented as a variable setpoint for the LTOP system instrumentation.

The NRC staff confirmed that in accordance with ASME Code, Section XI, Appendix G, the IP2 LTOP system arming temperature (T_{enable}) can be as low as 264 °F. In TS Amendment No. 224, the NRC staff accepted the use of ASME Code Case N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels," and N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves," as allowed by ASME Code, Section XI, Appendix G, for determination of the LTOP system setpoint. In the IP2 TS Amendment No. 224, the LTOP system arming temperature is 280 °F. Since the LTOP system arming temperature remains below the current TS value, the licensee is retaining the conservative TS value of 280 °F. Because the analyses demonstrate acceptable results and employ assumptions recommended in BTP 5-2, the NRC staff accepts the licensee's analyses and concludes that the requested LTOP system setpoint change and LTOP system arming temperature are acceptable.

3.3 NRC Staff's Conclusions

The NRC staff concludes that the proposed P-T limits curves and LTOP system requirements for IP2 satisfy the requirements in Appendix G to 10 CFR Part 50 and Appendix G to Section XI of the ASME Code. The proposed P-T limit curves also satisfy GL 88-11, because the methodology in RG 1.99, Rev. 2 was used to calculate the ART. Hence, the proposed P-T limit curves and LTOP system requirements may be incorporated into the IP2 TS and are valid through 29.2 EFPY.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (74 FR 23443). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the

issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Carolyn Fairbanks

Date: August 17, 2009

August 17, 2009

Vice President, Operations
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
Buchanan, NY 10511-0249

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 - ISSUANCE OF AMENDMENT RE: REACTOR VESSEL UPDATED HEATUP AND COOLDOWN CURVES (TAC NO. ME0788)

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Sincerely,
/ra/
John P. Boska, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-247

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2. Safety Evaluation

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