A. Alan Blind
Vice President

Consolidated Edison Company of New York, Inc.
Indian Point Station
Broadway & Bleakley Avenue
Buchanan, NY 10511
Telephone (914) 734-5340
Fax: (914) 734-5718
blinda@coned.com

July 16, 2001

Re: Indian Point Unit No. 2
Docket No. 50-247
NL 01-092

US Nuclear Regulatory Commission
Document Control Desk
Mail Stop O-P1-17
Washington, DC 20555-0001

SUBJECT: 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

Transmitted herewith is an Application for Amendment to the Operating License and a Request for Exemption from the requirements of 10CFR50.60(a) and 10CFR50 Appendix G. This application requests an amendment to the Consolidated Edison Company of New York, Inc. (Con Edison), Indian Point Unit No. 2 (IP2) Technical Specifications (TS). The exemption request is needed to facilitate the TS amendment.

The purpose of this License Amendment Request is to propose changes to the IP2 TS Sections 3.1.A, "Reactor Coolant System Operational Components," 3.1.B, "Reactor Coolant System Heatup and Cooldown," 3.2, "Chemical and Volume Control System," 3.3.A, "Engineered Safety Feature Safety Injection and Residual Heat Removal Systems," and 4.3, "Reactor Coolant System Integrity Testing," to incorporate revised reactor vessel Pressure-Temperature (P-T) limits to allow operation up to 25 Effective Full Power Years (EFPY). Associated Bases changes are also provided. The current reactor vessel P-T limits were incorporated into the IP2 TS by License Amendment 195, effective February 28,1998. Operation up to 18 EFPY is currently authorized in accordance with the NRC staff's Safety Evaluation accompanying License Amendment 195.

The proposed TS changes rely on American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code methodology for determining allowable heatup and cooldown limits. This methodology includes the incorporation of ASME B&PV Code Case N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels, Section XI, Division 1," and N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for ASME Section XI, Division 1." Since these Code Cases have not yet received USNRC approval for generic usage, this License Amendment Request also includes a request for an exemption in accordance with 10CFR50.12 from the requirements of 10CFR50.60(a), "Acceptance criteria for fracture prevention measures for lightwater nuclear

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power reactors for normal operation," to comply with 10CFR50, Appendix G, "Fracture Toughness Requirements" to allow use of ASME Code Cases N-588 and N-640. Similar exemptions were granted to Duke Energy for the Oconee Nuclear Station, Commonwealth Edison for the Quad Cities Nuclear Power Station, and PECO Nuclear for the Limerick Generating Station, Unit 1.

ASME Code Case N-588 allows the use of alternative procedures for defining the orientation of postulated flaws in circumferential welds and for calculating the applied stress intensity factors of axial and circumferential flaws. Code Case N-588 was approved for use by the ASME on December 12,1997. ASME Code Case N-640 allows an alternate method for determining the fracture toughness of reactor pressure vessel materials for use in determining P-T Limits. Code Case N-640 was approved for use by the ASME on February 26, 1999. The use of these Code Cases results in a reduction in required temperatures, for a given pressure, than would have been required without the use of the Code Cases.

Attachment 1 to this letter provides the description and evaluation of the proposed changes. The revised TS pages are provided in Attachment 2 (strikeout/shaded format). Attachment 3 to this letter provides information supporting the request for exemption from the requirements of 10CFR50.60(a) and 10CFR50 Appendix G to allow the use of ASME B&PV Code Cases N-640 and N-588. Attachment 4 is Westinghouse Technical Report WCAP-15629, "Indian Point 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation," that explains and justifies the changes to the P-T limit curves. Attachment 5 is Northeast Technology Corporation (NETCO) Technical Report NET-177-01, "Indian Point Unit 2 Overpressure Protection System (OPS) Thermal Hydraulic Analysis, Setpoint Development and Technical Specification Revision for 25 EFPY," that explains and justifies the changes to the TS Overpressure Protection System (OPS) requirements.

Con Edison requests NRC approval of the proposed change by February 3, 2002 with an effective date within 30 days of approval. The current heatup and cooldown limitation curves are only applicable until 18 EFPY. Con Edison currently projects IP2 to exceed 18 EFPY on March 3, 2002. The requested date of approval is therefore necessary to ensure continued plant operation.

The Station Nuclear Safety Committee (SNSC) and the Nuclear Facilities Safety Committee (NFSC) have reviewed the proposed change. Both committees concur that the proposed change does not involve a significant hazards consideration as defined by 10CFR50.92(c).

In accordance with 10CFR50.91, a copy of this submittal and the associated attachments are being submitted to the designated New York State official.

There are no new commitments in this letter.

There are no new commitments in this letter.

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.

Very truly yours,

A Alan Blind

A. Alan Blind
Vice President - Nuclear Power

cc: Hubert J. Miller
Regional Administrator-Region I
US Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. Patrick Milano, Project Manager
Project Directorate I-1
Division of Reactor Projects I/II
US Nuclear Regulatory Commission
Mail Stop O-8-2C
Washington, DC 20555

NRC Senior Resident Inspector
US Nuclear Regulatory Commission
PO Box 38
Buchanan, NY 10511

Mayor, Village of Buchanan
236 Tate Avenue
Buchanan, NY 10511

Mr. Paul Eddy
NYS Department of Public Service
3 Empire Plaza
Albany, NY 12223

Mr. William F. Valentino, President
NYS ERDA
Corporate Plaza West
286 Washington Ave. Extension
Albany, NY 12223-6399

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of CONSOLIDATED EDISON COMPANY OF NEW YORK, INC. (Indian Point Station, Unit No. 2)	))))	Docket No. 50-247

APPLICATION FOR AMENDMENT
TO OPERATING LICENSE

Pursuant to Section 50.90 of the Regulations of the Nuclear Regulatory Commission (NRC), Consolidated Edison Company of New York, Inc., as holder of Facility Operating License No. DPR-26, hereby applies for amendment of the Technical Specifications contained in Appendix A of this license. The specific proposed Technical Specification revision is set forth in the attachment. The associated assessment demonstrates that the proposed change does not involve a significant hazards consideration as defined in 10CFR50.92(c). As required by 10CFR50.91(b)(1), a copy of this Application and our evaluation concluding that the proposed change does not involve a significant hazards consideration has been provided to the appropriate New York State official designated to receive such amendments.

Pursuant to Section 50.12 of the Regulations of the Nuclear Regulatory Commission (NRC), Consolidated Edison Company of New York, Inc., as holder of Facility Operating License No. DPR-26, hereby applies for an exemption to the requirements of 10CFR50.60(a) and Appendix G. The specific proposed exemption is set forth in the attachment. The associated assessment demonstrates that the proposed exemption meets the criteria of 10CFR50.12.

BY: A. Alan Blind
A. Alan Blind

Subscribed and sworn to
before me this 16 day
July, 2001.

Ersilia A. Amanna (Bonin)
Notary Public

ERSILIA A. AMANNA
Notary Public, State of New York
No. 01AM6038689
Qualified in Westchester County
Commission Expires March 20, 2002

ATTACHMENT 1 TO NL 01-092

LICENSE AMENDMENT REQUEST

REACTOR COOLANT SYSTEM HEATUP AND COOLDOWN LIMITATIONS

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247

LICENSE AMENDMENT REQUEST

DESCRIPTION OF THE PROPOSED CHANGE

Consolidated Edison Company of New York, Inc. (Con Edison) is requesting a License Amendment to change to the Indian Point Unit No. 2 (IP2) Technical Specifications (TS) as described below.

The License Amendment Request (LAR) proposes revisions to the following TS sections to allow operation up to 25 effective full power years (EFPY).

1. List of Figures
2. Section 3.1.A.1 Reactor Coolant System, Operational Components, Coolant Pump
3. Section 3.1.A.4 Reactor Coolant System, Operational Components, Overpressure Protection System (OPS)
4. Table 3.1.A-2 OPS Operability Requirements
5. Figure 3.1.A-1 PORV Opening Pressure for Operation Less Than or Equal to 305°F
6. Figure 3.1.A-2 Maximum Pressurizer Level with PORVs Inoperable and One Charging Pump Energized
7. Figure 3.1.A-3 Maximum Reactor Coolant Pressure for Operation with PORVs Inoperable and One Safety Injection Pump And/Or Three Charging Pumps Energized
8. Section 3.1.B Reactor Coolant System, Heatup and Cooldown
9. Figure 3.1.B-1 Coolant System Heatup Limitations
10. Figure 3.1.B-2 Coolant System Cooldown Limitations
11. Section 3.2.D Chemical and Volume Control System
12. Section 3.3.A.2 Engineered Safety Features Safety Injection and Residual Heat Removal Systems
13. Table 4.1-1 Minimum Frequencies for Checks, Calibrations, and Tests of Instrument Channels
14. Section 4.3 Reactor Coolant System Integrity Testing,
15. Section 4.18 Overpressure Protection System

The LAR proposes to add the following new Figures:

1. Figure 3.1.A-4 Maximum RCS Pressure: OPS Inoperable and 3 Charging Pumps Capable of Injecting into the RCS
2. Figure 3.1.A-5 Maximum RCS Pressure: OPS Inoperable and Start of 1 RCP with SGs 40°F Hotter than RCS
3. Figure 3.1.A-6 Maximum RCS Pressure: OPS Inoperable and Start of 1 RCP with SGs 100°F Hotter than RCS

The LAR proposes to delete Figure 4.3-1, "Vessel Leak Test Limitations," and include the vessel leak test limitations in revised Figure 3.1.B-1, "Reactor Coolant System Heatup and Leak Test Limitations Applicable for the First 25 EFPY."

Associated changes are proposed for Bases Sections:
1. 3.1.A, Reactor Coolant System, Operational Components
2. 3.1.B, Reactor Coolant System, Heatup and Cooldown
3. 3.1.C, Minimum Conditions for Criticality
4. 3.2, Chemical and Volume Control System
5. 3.3, Engineered Safety Features
6. 4.3, Reactor Coolant System Integrity Testing

REASONS FOR THE CHANGE

As of April 23, 2001, the IP2 burnup was 17.138 EFPY. The Heatup and Cooldown Limitation Curves that are currently available only allow operation until 18 EFPY. It is anticipated that IP2 will have achieved 18 EFPY by March 3, 2002. These curves require revision prior to the plant reaching the current limit to ensure that continuity is maintained regarding the availability of heatup and cooldown limitations. The revised curves will reflect heatup and cooldown limitations valid until 25 EFPY.

EVALUATION OF THE PROPOSED CHANGE

General

The heatup and cooldown limitation curves are established to provide assurance of reactor pressure vessel (RPV) integrity during plant operation. All components of the reactor coolant system (RCS) are designed to withstand the effects of loads resulting from system pressure and temperature changes. These loads are introduced by heatup and cooldown operations, power transients, and reactor trips. In accordance with Appendix G to 10CFR50, the TS limit the pressure and temperature changes during heatup and cooldown to be within the fracture toughness requirements to preclude non-ductile failure of the carbon and low alloy RCS materials. These limits are defined by the Pressure-Temperature (P-T) curves for heatup and cooldown. Each curve defines an acceptable region for normal operation. These curves are used for operational guidance during heatup and cooldown maneuvering when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

Inservice leak and hydrostatic pressure testing required by Section XI of the ASME (American Society of Mechanical Engineers) Boiler and Pressure Vessel (B&PV) Code is performed as required to ensure system integrity. The minimum temperatures at the required pressures allowed for these tests are determined from the RPV pressure and temperature limits required by current TS Figure 4.3-1 (proposed Figure 3.1.B-1).

The cold Overpressure Protection System (OPS) is designed to relieve the RCS pressure for certain unlikely overpressure transients to prevent these incidents from causing the peak RCS pressure from exceeding 10 CFR 50 Appendix G limits. The OPS is set to cause the PORVs to open at a pressure sufficiently low to prevent exceeding the Appendix G isothermal limits for the following events: .

1. Startup of a reactor coolant pump with no other reactor coolant pumps running and the steam generator secondary side water temperature hotter than the RCS water temperature.
2. Letdown isolation with three charging pumps operating.
3. Startup of three charging pumps or one safety injection pump with 2 charging pumps.
4. Loss of residual heat removal causing pressure rise from heat additions from core decay heat or reactor coolant pump heat.
5. Inadvertent activation of the pressurizer heaters.

Summary of the Methodology and Results for the Calculation of Updated Heatup and Cooldown Limitations

Attachment 4, WCAP-15629, "Indian Point Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation," provides a detailed description of methodology used to calculate the heatup and cooldown limitations and the results of those calculations.

The proposed changes were developed in accordance with the following NRC regulations and guidance:

- 10CFR50 Appendix G
- Regulatory Guide (RG) 1.99, Rev. 2
- ASME B&PV Code Section XI Appendix G, 1995 Edition with 1996 Addenda
- ASME Code Cases N-588 and N-640

10CFR50 Appendix G, by reference to ASME B&PV Code Section XI Appendix G, specifies fracture toughness and testing requirements for reactor vessel materials . 10CFR50 Appendix G also requires prediction of the effects of neutron irradiation on vessel embrittlement by calculating the Adjusted Reference Temperature (ART) and the Charpy Upper Shelf Energy (USE). Generic Letter 88-11 requested that the methods provided in RG 1.99, Rev. 2, be used to predict the effect of neutron irradiation on reactor vessel materials. RG 1.99, Rev.2, defines the ART as the sum of unirradiated reference temperature, the increase of reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Effect of Use of the Code Cases

ASME Code Case N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels, Section XI, Division 1," allows the use of alternative procedures for defining the orientation of postulated flaws in circumferential welds (i.e., only circumferential flaws need to be considered for circumferential welds) and for calculating the applied stress intensity factors of axial and circumferential flaws.

ASME Code Case N-640, "Alternative Fracture Toughness for Development of P-T Curves for ASME Section XI, Division 1," provides an alternate method for determining the fracture toughness of reactor vessel materials for use in determining P-T limits. This Code Case allows the use of the critical stress intensity factor (K_{Ic}) rather than the more restrictive arrest stress intensity factor (K_{Ia}/K_{Ir}) required by ASME B&PV Code Section XI Appendix G.

The changes to the calculation methodology for the heatup and cooldown limitation curves based on Code Case N-640 and Code Case N-588 provide sufficient margin in the prevention of non-ductile type fracture of the reactor pressure vessel while maximizing operator flexibility during plant heatup and cooldown. The code cases were developed using knowledge gained through years of industry experience. However, the experience gained in the areas of fracture toughness of materials and pre-existing undetected defects show that some of the previous assumptions used for the calculation of the heatup and cooldown limitations were overly conservative. Therefore, using the methods of the subject Code Cases in developing the heatup and cooldown limitation curves will continue to provide protection against non-ductile failures of the carbon and low alloy steel components of the RCS.

ASME Code Cases N-588 and N-640 have not been approved in Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1." Con Edison requests an exemption in accordance with 10CFR50.12 from the requirement of 10CFR50.60(a), "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation," to comply with 10CFR50 Appendix G, "Fracture Toughness Requirements" to allow use of ASME Code Cases N-588 and N-640 in the calculation of heatup and cooldown limitations. IP2 has evaluated the use of these ASME Code Cases and has concluded that the use of the code cases will not present an undue risk to the public health and safety, are consistent with the common defense and security, and special circumstances are present. The detailed evaluation of the exemption to the criteria of 10CFR50.12 is contained in Attachment 3.

Con Edison evaluated the effect of neutron irradiation embrittlement on each beltline material in the IP2 RPV. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. Con Edison determined that the material with the highest Adjusted Reference Temperature (ART) at the ¾T (T= reactor vessel beltline thickness) location at 25 EFPY is the Intermediate Shell Plate B-2002-3 with 0.25 % copper, 0.60 % nickel, and an ART of 145°F. The limiting material for the ¼T location is the intermediate to lower shell girth weld with an ART of 200°F. However, the use of Code Case N-588 results in Intermediate Shell Plate B-2002-3 controlling the ¼T location with an ART of 195°F.

Con Edison has removed surveillance samples from the IP2 RPV at the exposures shown on the following Table. The test results from these samples were published in the reports listed in the Table and were transmitted to the NRC as shown. The increase in RT_{NDT} values for the limiting materials for 25 EFPY were calculated based on these IP2 surveillance capsule results supplemented by surveillance capsule data from Indian Point Unit 3 and H.B. Robinson Unit 2.

Surveillance Sample	Exposure (EFPY)	Report Number and Title	NRC Transmittal Letter No. and Date
T	1.08	Final Report - SWRI Project No. 02-4531 - "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule T," E.B. Norris, June 30, 1977	NL 79-A04, January 9, 1979
Y	2.34	Final Report - SWRI Project No. 02-5212 - "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule Y," E.B. Norris, November 1980	NL 82-A40, May 5, 1982
Z	5.17	Final Report - SWRI Project No. 06-7379 - "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule Z," E.B. Norris, April 1984	NL 84-A43, May 7, 1984
V	8.6	Final Report - SWRI Project No. 17-2108 (Revised)- "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule V," F.A. Iddings - SWRI, March, 1990	NL 90-044, March 30, 1990. (originally transmitted in NL 88-161, Oct. 12, 1988)

The Chemistry Factors were calculated using both Positions 1.1 and 2.1 of RG 1.99, Rev. 2.

The Heatup and Cooldown curves were calculated using the methodology of WCAP-14040-NP-A, Rev. 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," with the following exceptions as explained in Attachment 3:

1. The fluence values are calculated, not best estimate, values.
2. Using the ASME B&PV Code Case N-640 methodology, the K_{Ic} critical stress intensities were used in place of the K_{Ia} critical stress intensities.
3. The 1996 version of the ASME B&PV Code Section XI Appendix G was used rather than the 1989 version.
4. P-T curves for the limiting circumferential weld ART were generated in conjunction with ASME B&PV Code Case N-588.

The calculation inputs used historical data for IP2 operation for cycles 1 through 15. For cycles 16 through 25, the calculations assumed an increase in core power to 3216 MW_t and operation with T_c of 545°F and T_{avg} of 579°F. IP2 currently operates at 3071.4 MW_t with T_c of 529°F and T_{avg} of 559°F.

The intermediate to lower shell girth weld was determined to be the limiting beltline material for the

¼T location and the intermediate shell plate B-2002-3 was determined to the most limiting beltline material for the ¾T location. Heatup and Cooldown Limitation curves using the methodology from ASME Code Case N-588 were calculated for the intermediate to lower shell girth weld as the limiting component with a postulated circumferential flaw. Heatup and Cooldown Limitation curves using the methodology from the 1996 ASME B&PV Code Section XI Appendix G were calculated for the intermediate shell plate B-2002-3 as the limiting component with a postulated axial flaw. The Heatup and Cooldown Limitation curves calculated for the intermediate shell plate B-2002-3 were found to be more restrictive than those calculated for the intermediate to lower shell girth weld. Thus the intermediate shell plate B-2002-3 is the limiting beltline material. The Heatup and Cooldown Limitation curves for the intermediate shell plate B-2002-3 are presented in Attachment 2 as the updated TS figures.

For the limiting beltline material, the intermediate shell plate B-2002-3, Con Edison calculated the ART at 25 EFPY to be 145°F for the ¾T location and 195°F for the ¼T location. The ART was determined in accordance with Position 2.1 of RG 1.99, Rev 2. A margin of 17°F was applied to the ART calculation.

The analysis confirmed that the proposed heatup and cooldown limitation curves meet the beltline material requirements in 10CFR50 Appendix G for the most limiting component, intermediate shell plate B-2002-3.

In addition to beltline materials, 10CFR50 Appendix G imposes heatup and cooldown limitations based on the reference temperature for the RPV closure flange region materials. Section IV.A.2 of Appendix G states that when the pressure exceeds 20% of the pre-service system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by bolt preload must exceed the reference temperature of the materials in those regions by at least 120°F for normal operation and 90°F for hydrostatic and leak tests. The pre-service system hydrostatic test pressure for the IP2 RPV was 3106 psi. Based on the limiting unirradiated flange region RT_{NDT} of 60°F, Con Edison has determined that imposing a minimum allowable temperature limit of 180°F when pressure exceeds 621 psig satisfies Section IV.A.2 of 10CFR50 Appendix G.

The analysis confirms the conservatism and continued applicability of the minimum temperature condition for criticality in TS 3.1.C to meet the requirements of 10CFR50 Appendix G.

Based on the highest calculated ¼T ART of 200°F, the analysis determined that a minimum low temperature Overpressure Protection System enable (arming) temperature of 280°F satisfies the requirements of ASME B&PV Code Section XI Appendix G to ensure that overpressurization of the RCS at low temperatures will not result in component stresses in excess of those allowed by the ASME B&PV Code Section XI Appendix G.

Summary of the Methodology and Results for the Calculation of Updated Overpressure Protection System Limits

Attachment 5, Northeast Technology Corp (NETCO) Technical Report NET-177-01 titled "Indian Point Unit 2 Overpressure Protection System (OPS) Thermal Hydraulic Analysis, Setpoint Development and Technical Specification Revision for 25 EFPY," provides a detailed description of the methodology used to calculate the Overpressure Protection System (OPS) Limits and the results of those calculations. The Appendices of NET-177-01 have not been included since the report and this submittal stand alone. The appendices provide the recommended TS changes, an analysis of instrument uncertainties that was used to validate that the proposed limits could be effectively implemented, and supporting information.

The OPS limits were developed using a thermal hydraulic analysis based on the heatup and cooldown limits described in WCAP-15629, "Indian Point Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation."

The analysis determined that a revised arming temperature of 280°F (reduced from a previous value of 305°F) along with the other proposed limits provides the required protection. By implementing the proposed OPS setpoints and limits, the OPS system will prevent the peak pressure from the analyzed events from exceeding the 10CFR50 Appendix G limits. Thus, the IP2 RPV continues to be protected against the analyzed overpressure transients through the 25 EFPY analysis limit.

Implementation of Heatup, Cooldown, and OPS Curves and Limitations

The proposed TS limits are the deterministic limits based on the requirements of 10CFR50 Appendix G. Plant processes will be used to apply appropriate instrument correction and uncertainty adjustments for the implementation of the approved limits in operating and setpoint adjustment procedures.

The use of 25 EFPY curves will conservatively bound IP2 operation between its exposure at implementation and the limit of the current curves.

Other Changes

Three additional changes have been made:

- Clarifying Notes have been added to Table 3.1.A-2, "OPS Operability Requirements – Safety Injection and Charging Pumps."
- Footnote 6 of Table 4.1-1, "Minimum Frequencies for Checks, Calibrations, and Tests of Instrument Channels" applicable to Item 37, "Overpressure Protection System Test," has been changed.
- The current TS in numerous places describe pumps as either being "energized" or "de-energized." This has been changed to "Capable (or not capable) of Injecting into the RCS."

A review of the NRC staff's Safety Evaluation (Ref. 1) for the IP2 OPS indicated that the staff's approval for the TS relied upon the fact that SI pumps are de-energized below 300 psi but that:

> "There are certain plant procedures that may require a single HPSI pump to have power reinstated or to be run while the plant is in a cold shutdown condition. The plant procedures contain numerous administrative precautions during these necessary operations."

Con Ed has decided that a more conservative implementation of the allowance to energize a single SI pump during cold shutdown would be to apply the logic used by the NUREG-1431, "Standard Technical Specifications Westinghouse Plants," as implemented in the TS for our sister plant IP3. That logic is to add NOTES to allow the energizing of a single SI pump only for the following conditions:

- When necessary to respond to abnormal conditions such as loss of RHR cooling, emergency boration, etc.
- When required for surveillance testing for periods not to exceed 8 hours in conjunction with operation of charging pumps for normal makeup.

In addition, the first proposed Note allows the use of more than one SI pump to be capable of injection to respond to loss of RCS inventory events. This Note reinforces the clear priority that core cooling takes precedent over the administrative limitations of Table 3.1.A-2. This Note is consistent with the NUREG-1431 B 3.4.12 Background statement:

> "With minimum coolant input capability [due to implementation of the OPS limitations], the ability to provide core coolant addition is restricted. ... If conditions require the use of more than one [HPI or] charging pump for makeup in the event of a loss of inventory, the pumps can be made available through manual actions."

The change to the surveillance frequency is also consistent with the NUREG-1431 surveillance frequency. In approving the Standard TS with this 12 hour delay, the NRC considered the unlikelihood of a low temperature overpressure event during this time as well as the reliability of the PORV actuation circuits. Without the flexibility allowed by this proposed change, plant operations following a forced outage when a cooldown may be required are unnecessarily:

- Delayed by the need to perform the test before decreasing temperature less than the OPS arming temperature, or
- Complicated by the need to impose the pressurizer level and RCS pressure and temperature limits required when OPS is inoperable.

Con Edison has evaluated the option of performing the required testing on the once-per-31-day frequency when the plant is operating so that OPS is always available to support a forced outage rapid cooldown. This option is undesirable due to the increased risk of PORV actuation at power and a decrease in PORV availability since there would be an increase in the time when PORV control circuits are inoperable while in the abnormal test lineup.

The change from "energized" to "capable of injecting into the RCS" is a clarification that is consistent with the NRC Staff's Safety Evaluation (Ref. 1) regarding administrative controls, the logic of current TS 4.8.D, and NUREG-1431. The Safety Evaluation describes the use of pull-to-lock switches as a means of de-energizing Safety Injection pumps. TS 4.8.D allows “other means” of de-energizing the safety injection and/or charging pumps. This clarification makes it clear that “other means” such as valve lineups to prevent injection and placement of control switches into pull-to-lock will prevent the inadvertent pressurization of the RCS.

NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

Con Edison has determined that this proposed Technical Specification change does not involve a significant hazards consideration as defined by 10CFR50.92(c).

1. **Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated.**

   The proposed TS changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. There are no physical changes to the plant being introduced by the proposed changes to the heatup and cooldown limitation curves. The proposed changes do not modify the RCS pressure boundary. That is, there are no changes in operating pressure, materials, or seismic loading. The proposed changes do not adversely affect the integrity of the RCS pressure boundary such that its function in the control of radiological consequences is affected. The proposed heatup and cooldown limitation curves were generated in accordance with the fracture toughness requirements of 10CFR50 Appendix G, and ASME B&PV Code, Section XI, Appendix G in conjunction with ASME Code Cases N-640 and N-588. The proposed heatup and cooldown limitation curves were established in compliance with the methodology used to calculate and predict effects of radiation on embrittlement of RPV beltline materials. Use of this methodology provides compliance with the intent of 10CFR50 Appendix G and provides margins of safety that ensure non-ductile failure of the RPV will not occur.

   The proposed heatup and cooldown limitation curves prohibit operation in regions where it is possible for non-ductile failure of carbon and low alloy RCS materials to occur. Hence, the primary coolant pressure boundary integrity will be maintained throughout the limit of applicability of the curves, 25 EFPY. Operation within the proposed OPS limits ensures that overpressurization of the RCS at low temperatures will not result in component stresses in excess of those allowed by the ASME B&PV Code Section XI Appendix G.

   Consequently, the proposed changes do not involve a significant increase in the probability or the consequences of an accident previously evaluated.

2. **Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.**

The proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes to the heatup and cooldown limitation curves were generated in accordance with the fracture toughness requirements of 10CFR50 Appendix G and ASME B&PV Code, Section XI, Appendix G in conjunction with ASME Code Cases N-588 and N-640. Compliance with the heatup and cooldown limitation curves will ensure that conditions in which non-ductile failure of the RCS pressure boundary materials is possible will be avoided. Compliance with the proposed OPS limits will ensure that the RCS will be physically protected against overpressurization events during low temperature operation when the fracture toughness properties of the carbon and low alloy components are at their lowest.

No new modes of operation are introduced by the proposed changes. The proposed changes will not create any failure mode not bounded by previously evaluated accidents. Further, the proposed changes to the heatup and cooldown limitation curves and the OPS limits do not affect any activities or equipment other than the RCS pressure boundary and are not assumed in any analysis to initiate or mitigate any accident sequence. Therefore the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in the margin of safety.**

The proposed TS changes do not involve a significant reduction in the margin of safety.

The revised heatup and cooldown limitation curves and OPS limits provide more operating flexibility than the current heatup and cooldown limitation curves. Industry experience since the inception of pressure-temperature limits in the 1970s confirms that some of the original methodologies used to develop the heatup and cooldown limitation curves are overly conservative. Accordingly, ASME Code Cases N-588 and N-640 take advantage of the acquired knowledge by establishing more realistic methodologies for development of the heatup and cooldown limitation curves. Therefore, operational flexibility is gained and an acceptable margin of safety to reactor pressure vessel non-ductile type fracture is maintained.

The revised heatup and cooldown limitation curves and OPS limits are established in accordance with current regulations and the ASME B&PV Code 1996 version. These proposed changes are acceptable because the ASME B&PV Code maintains the margin of safety required by 10CFR50.55(a). Because operation will be within these limits, the RCS materials will continue to behave in a ductile manner consistent with the original design bases.

The proposed changes to the allowable operation of charging and safety injection pumps when OPS is required to be operable is consistent with the IP2 licensing bases but implements the licensing bases in a more conservative manner than the current TS. The change in OPS surveillance frequency has been previously evaluated by the NRC to involve an insignificant increase in risk. That insignificant increase in risk is offset by the adverse effects of the alternatives of either 1.) delaying forced cooldowns until OPS testing is complete; 2.) complicating cooldown operations by imposition of limits required when OPS is inoperable; or 3.) conducting OPS testing periodically while at power.

Therefore, Con Edison has concluded that the proposed changes do not involve a significant reduction in a margin of safety.

CONCLUSIONS

Based on the above evaluation, Con Edison has concluded that the proposed change will not result in a significant increase in the probability or consequences of any accident previously analyzed; will not result in a new or different kind of accident from any accident previously analyzed; and does not result in a reduction in any margin of safety. Therefore, operation of IP2 in accordance with the proposed amendment does not involve a significant hazards consideration. The Station Nuclear Safety Committee (SNSC) and the Nuclear Facilities Safety Committee (NFSC) have reviewed the proposed change. Both committees concur that the proposed change does not involve a significant hazards consideration as defined by 10CFR50.92(c).

ENVIRONMENTAL ASSESSMENT

An environmental assessment is not required for the above proposed change because the requested change to the Indian Point Generating Station Unit 2 Technical Specifications conform to the criteria for "actions eligible for categorical exclusion" as specified in 10CFR 51.22(c)(9). The requested change will have no impact on the environment. The proposed change does not involve a significant hazards consideration as discussed in the preceding section. The proposed change does not involve a significant change in the types or significant increase in the amounts of any effluents that may be released offsite. In addition, the proposed change does not involve a significant increase in individual or cumulative occupational radiation exposure.

REFERENCES

1. NRC letter (RA 84-A38) to Con Edison, "Indian Point 2 – Low Temperature Overpressure Protection System," dated April 24, 1984

ATTACHMENT 2 TO NL 01-092

TECHNICAL SPECIFICATION PAGES IN

STRIKEOUT/SHADOW FORMAT

Deleted text is shown as ~~strikeout~~.

Added text is shown as shaded.
New Figures are not shaded but are shown with no Amendment number.

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247

LIST OF FIGURES

A. OPERATIONAL COMPONENTS

1. Coolant Pump

a. Except as noted in 3.1.A.1.b below, four reactor coolant pumps shall be in operation during power operation.

b. During power operation, one reactor coolant pump may be out of service for testing or repair purposes for a period not to exceed four hours.

c. During shutdown conditions with fuel in the reactor, the operability requirements for reactor coolant and/or residual heat removal pumps specified in Table 3.1.A-1 shall be met.

d. When RCS temperature is less than or equal to ~~305~~ 280°F, the requirements of Specification 3.1.A.4 regarding startup of a reactor coolant pump with no other reactor coolant pumps operating shall be adhered to.

2. Steam Generator

Two steam generators shall be capable of performing their heat transfer function whenever the reactor coolant system is above 350°F.

3. Safety Valves

a. At least one pressurizer code safety valve shall be operable, or an opening greater than or equal to the size of one code safety valve flange shall be provided to allow for pressure relief, whenever the reactor head is on the vessel except for hydrostatically testing the RCS in accordance with Section XI of the ASME Boiler and Pressure Vessel Code.

b. All pressurizer code safety valves shall be operable whenever the reactor is critical.

c. The pressurizer code safety valve lift settings shall be set at 2485 psig with ±1% allowance for error.

4. Overpressure Protection System (OPS)

a. Except as permitted by Table 3.1.A-2, the OPS shall be armed and operable when the RCS temperature is ≤ ~~305~~ 280°F. When OPS is required to be operable, the PORV will have settings within the limits shown in Figure 3.1.A-1.

b. The requirements of 3.1.A.4.a may be modified to permit one PORV and/or its associated motor operated valve to be inoperable for a maximum of seven (7) consecutive days. If the PORV and/or its series motor operated valve is not restored to operable status within this seven (7) day period, or if both PORVs or their associated block valves are inoperable, action shall be initiated immediately to place the reactor in a condition where OPS operability is not required.

c. In the event either a PORV(s) or a RCS vent(s) is used to mitigate an RCS pressure transient, a special report shall be prepared and submitted to the Nuclear Regulatory Commission within 30 days pursuant to Specification 6.9.2.i. The report shall describe the circumstances initiating the transient, the effect of the PORV(s) or vent(s) on the transient, and any corrective action necessary to prevent recurrence.

The OPS will be set to cause the PORVs to open at a pressure sufficiently low to prevent exceeding the Appendix G limits for the following events:

1. Startup of a reactor coolant pump with no other reactor coolant pumps running and the steam generator secondary side water temperature hotter than the RCS water temperature.

2. Letdown isolation with three charging pumps operating.

3. Injection into the RCS from the ~~S~~startup of:
   - Three charging pumps, or
   - One safety injection pump and 2 charging pumps

4. Loss of residual heat removal causing pressure rise from heat additions from core decay heat or reactor coolant pump heat.

5. Inadvertent activation of the pressurizer heaters.

Consideration of the above events provides bounding PORV setpoints for other potential overpressure conditions caused by heat or mass additions at low temperature.

The RCS is protected against overpressure transients when RCS temperature is less than or equal to ~~305~~ 280°F by: (1) restricting the number of charging and safety injection pumps that ~~can be energized~~ are capable of injecting into the RCS to that which can be accommodated by the PORVs or the gas space in the pressurizer, (2) providing administrative controls on starting of a reactor coolant pump when the primary water temperature is less than the secondary water temperature, or (3) providing vent area from the RCS to containment for those situations where neither the PORVs nor the available pressurizer gas space are sufficient to preclude the pressure resulting from postulated transients from exceeding the limits of 10 CFR 50, Appendix G.

The restrictions on starting a reactor coolant pump with the secondary side water temperature higher than the primary side will prevent RCS overpressurizations from the resultant volumetric swell into the pressurizer that is caused by potential heat additions from the startup of a reactor coolant pump without any other reactor coolant pumps operating. When pressurizer level is between 30 and 85% of span, protection is provided through the use of the PORVs. When pressurizer level is less than 30% of span, additional restrictions on pressurizer pressure make reliance on the PORVs unnecessary since the gas compression resulting from the insurge of liquid from the RCS pump start is insufficient to cause RCS pressure to exceed the Appendix G limits. The same method, i.e., control of pressurizer pressure and level, is used to accommodate the mass insurge into the pressurizer from safety injection and charging pump starts when the PORVs are not operational.

An additional restriction is put on the reactor coolant pump start when the secondary system water temperature is less than or equal to ~~30~~ 40°F higher than the primary system water temperature and the pressurizer level is greater than 30%. This restriction is to prohibit starting the first reactor coolant pump when the RCS temperature is between ~~275~~ 249°F and ~~305~~ 280°F. The purpose of the restriction is to assure that the temperature rise resulting from the transient will not be outside the temperature limits for OPS actuation.

When comparison to the Appendix G limits is made, the comparison is to the isothermal Appendix G curve. ~~Other than the delay time associated with opening the PORVs and the error caused by non-uniform RCS metal and water temperatures during heat addition transients, the analysis does not make any allowance for instrument error.~~ The analysis for Figure 3.1.A-1 includes the time delay associated with the opening of the PORVs, the difference in elevation between the PORVs and the RCS pressure sensors, a 5°F temperature margin, a 10 psi pressure margin, and the error caused by non-uniform RCS metal and water temperatures during heat addition transients. Instrument error and bias will be taken into account when the OPS is set; i.e., the instrumentation will be set so that the PORVs will open at less than the required setpoint, including allowance for instrument error~~s~~ and bias.

The determination of reactor coolant temperature may be made from the Control Room instrumentation. The determination of the steam generator water temperature may be made in the following ways:

(a) assuming that the secondary side water temperature is at the saturation temperature corresponding to the secondary side steam pressure indicated on the Control Room instrumentation, or

(b) conservatively assuming that the secondary side water temperature is at the reactor coolant temperature at which the last RCP was stopped during cooldown, or

(c) actual or inferred measurement of the secondary side steam generator water temperature at those times it can be measured (such as return from a refueling outage).

(d) Similarly, when OPS is not available, the limitations on RCS pressure and level, and secondary-to-primary water temperature difference, include the effects of differences in elevation between the pressurizer liquid level and the RCS pressure sensors, a 5° F temperature margin, a 10 psi pressure margin, and the error caused by non-uniform RCS metal and water temperature. Allowances for other instrument errors and bias are not included in the Tech Spec values; but are included in the curves and procedures that implement the Tech Spec limits on operation.

Each of the pressurizer code safety valves is designed to relieve 408,000 lbs. per hr. of the saturated steam at the valve set point. Below approximately 350°F and 450 psig in the Reactor Coolant System, the Residual Heat Removal System can remove decay heat and thereby control system temperatures and pressure(2).

If no residual heat were removed by the Residual Heat Removal System, the amount of steam which could be generated at safety valve relief pressure would be less than half the capacity of a single valve. One valve therefore provides adequate protection for overpressurization.

The combined capacity of the three pressurizer safety valves is greater than the maximum surge rate resulting from complete loss of load(3) without a direct trip or any other control.

Two steam generators capable of performing their heat transfer function will provide sufficient heat removal capability to remove decay heat after a reactor shutdown.

All pressurizer heaters are supplied electrical power from an emergency bus. The requirement that 150kW of pressurizer heaters and their associated controls be operable when the reactor coolant system is above 350°F provides assurance that these heaters will be available and can be energized during a loss of offsite power condition to assist in maintaining natural circulation at hot shutdown

The power-operated relief valves (PORVs) can operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to provide a relief path when desirable and to ensure the ability to seal off possible RCS leakage paths. Both the PORVs and the PORV block valves are subject to periodic valve testing for operability in accordance with the ASME Code Section XI as specified in the Indian Point Unit No. 2 Inservice Inspection and Testing Program.

<u>Reference</u>

(1) UFSAR Section 14.1.12
(2) UFSAR Section 9.3.1
(3) UFSAR Section 14.1.8
(4) ~~Revised OPS Setpoints For Indian Point Unit 2, D.M. Speyer and A.P. Ginsberg, Feburary 14, 1991~~. NET-177-01, Indian Point Unit 2 Overpressure Protection System (OPS) Thermal Hydraulic Analysis, Setpoint Development And Technical Specification Revision for 25 EFPY

Table 3.1.A-2

OPS Operability Requirements

Reactor Coolant Pumps

With OPS operable at or below ~~305~~280°F, a reactor coolant pump can be started (or jogged) with no other reactor coolant pumps operating if:

(1) The temperature of all steam generators is less than or equal to the RCS temperature, or

(2) The temperature of all steam generators is less than or equal to ~~30~~ 40°F higher than the RCS temperature and:

- o RCS temperature is less than or equal to ~~275~~ 249°F,
- o Pressurizer level is between 30 - 85% of span; or

~~(3) The temperature of all steam generators is less than or equal to 100°F higher than RCS temperature and:~~

- ~~o RCS pressure is less than or equal to 450 psig,~~
- ~~o RCS temperature is greater than or equal to 145°F,~~
- ~~o Pressurizer level is less than or equal to 30% of span.~~

With OPS inoperable at or below ~~305~~280°F, a reactor coolant pump can be started (or jogged) with no other reactor coolant pumps operating if:

(1) The temperature of all steam generators is less than or equal to the RCS temperature, or

(2) The temperature of all steam generators is less than or equal to 40°F higher than the RCS temperature and:
- o RCS pressure operating restrictions are as specified in figure 3.1.A-5

(3) The temperature of all steam generators is less than or equal to 100°F higher than RCS temperature and:

- ~~RCS pressure is less than or equal to 450 psig,~~
- ~~RCS temperature is greater than or equal to 145°F,~~
- ~~Pressurizer level is less than or equal to 30% of span.~~
- RCS pressure operating restrictions are as specified in Figure 3.1.A-6

Table 3.1.A-2

OPS Operability Requirements

Safety Injection and Charging Pumps

NOTE:
1. If conditions require the use of Safety Injection pumps for makeup in the event of a loss of RCS inventory, the pumps can be made capable of injecting into the RCS through manual actions.
2. With charging pumps operating for normal RCS makeup, one SI pump may be made capable of injecting into the RCS as needed to support abnormal operations such as emergency boration or response to a loss of RHR cooling.
3. With charging pumps operating for normal makeup, one SI pump may be made capable of injecting into the RCS for pump testing for a period not to exceed eight hours.

With OPS operable at or below ~~305~~ 280°F, no more than ~~one (1) safety injection (SI) and or three (3) charging pumps may be energized~~ three charging pumps may be capable of injecting into the RCS; OR, for the reduced PORV actuation curve (See Figure 3.1.A-1), one safety injection and two charging pumps may be capable of injecting into the RCS..

OPS is not required to be operable at or below ~~305~~ 280°F if either the conditions of Column II or the conditions of Column III below are met for the ~~specified conditions~~ maximum number of SI and Charging pumps capable of injecting into the RCS specified in Column I:

Column I Maximum Number of ~~Energized~~ SI and Charging Pumps ~~(SI and/or charging)~~ Capable of Injecting into the RCS		Column II ~~I~~ Operating Restrictions (pressurizer pressure, pressurizer level, and RCS temperature)	Column III ~~II~~ Vent Area to Containment Atmosphere (square inches)
SI	Charging		
0	1	See Figure 3.1.A-2.	2.00 (or 1 PORV fully open)
0	2	See Figure 3.1.A-3.	2.00 (or 1 PORV fully open)
0	3	See Figure 3.1.A-4	2.00 (or 1 PORV fully open)
1	0, 1, 2 or 3	~~See Figure 3.1.A-3.~~ Use Column III only	2.00 (or 1 PORV fully open)
2	0, 1, 2 or 3	Use Column III only.	5.00 (or 2 PORVs fully open)
3	0, 1, or 2	Use Column III only	5.00 (or 2 PORVs fully open)
3	3	Use Column III only.	5.00

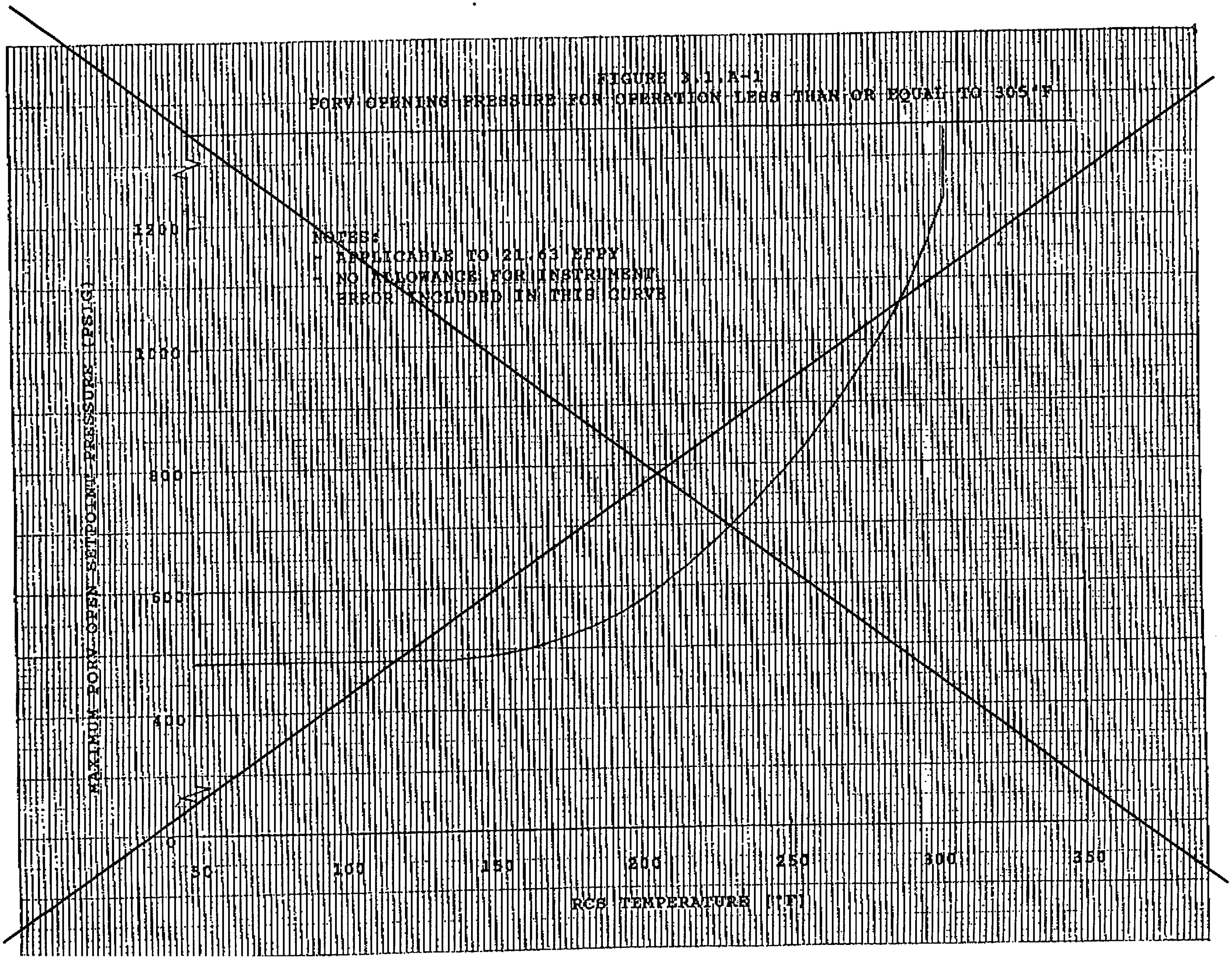

FIGURE 3.1.A-1
PORV OPENING PRESSURE FOR OPERATION LESS THAN OR EQUAL TO 305°F

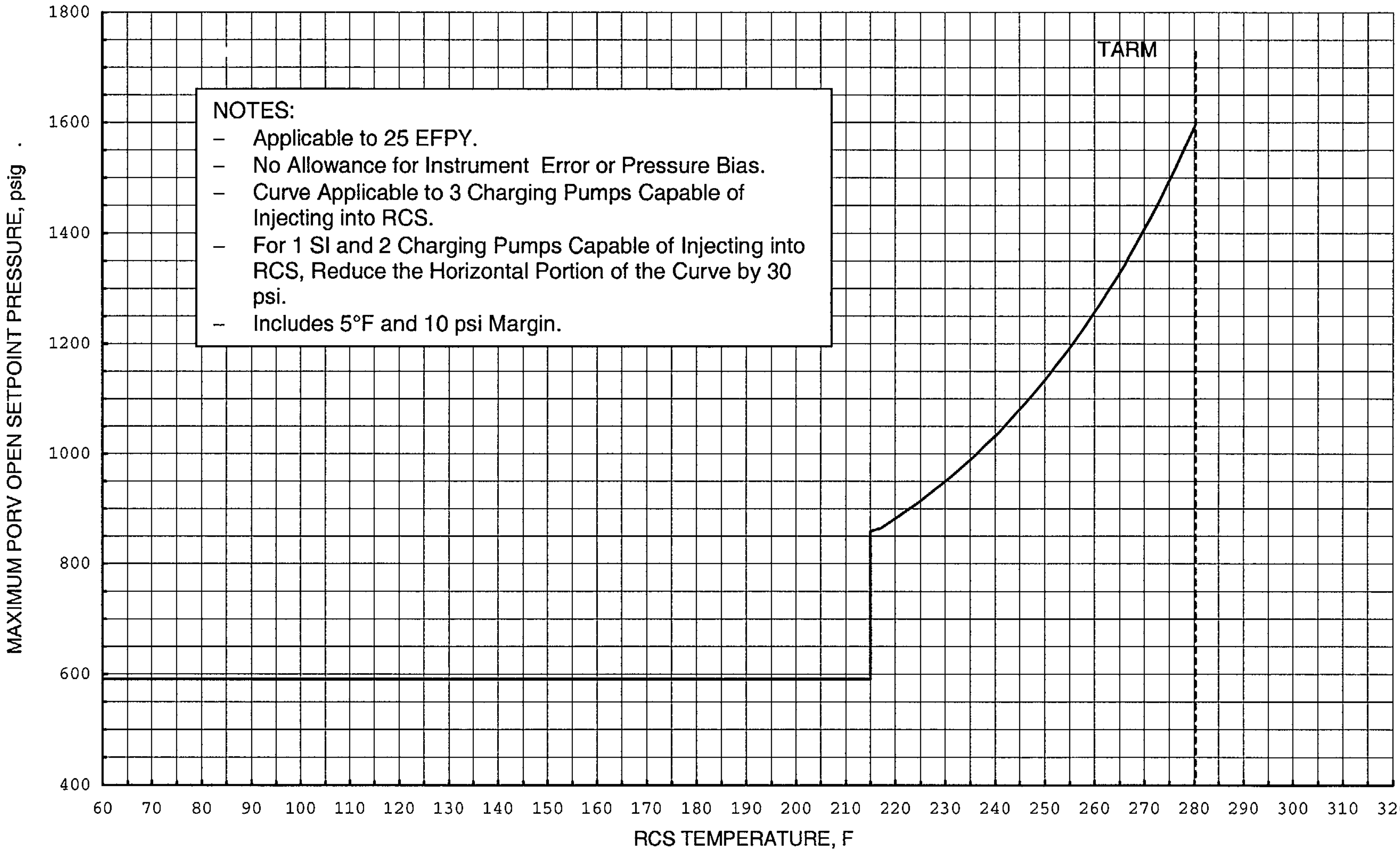

Figure 3.1.A-1 PORV Open Pressure

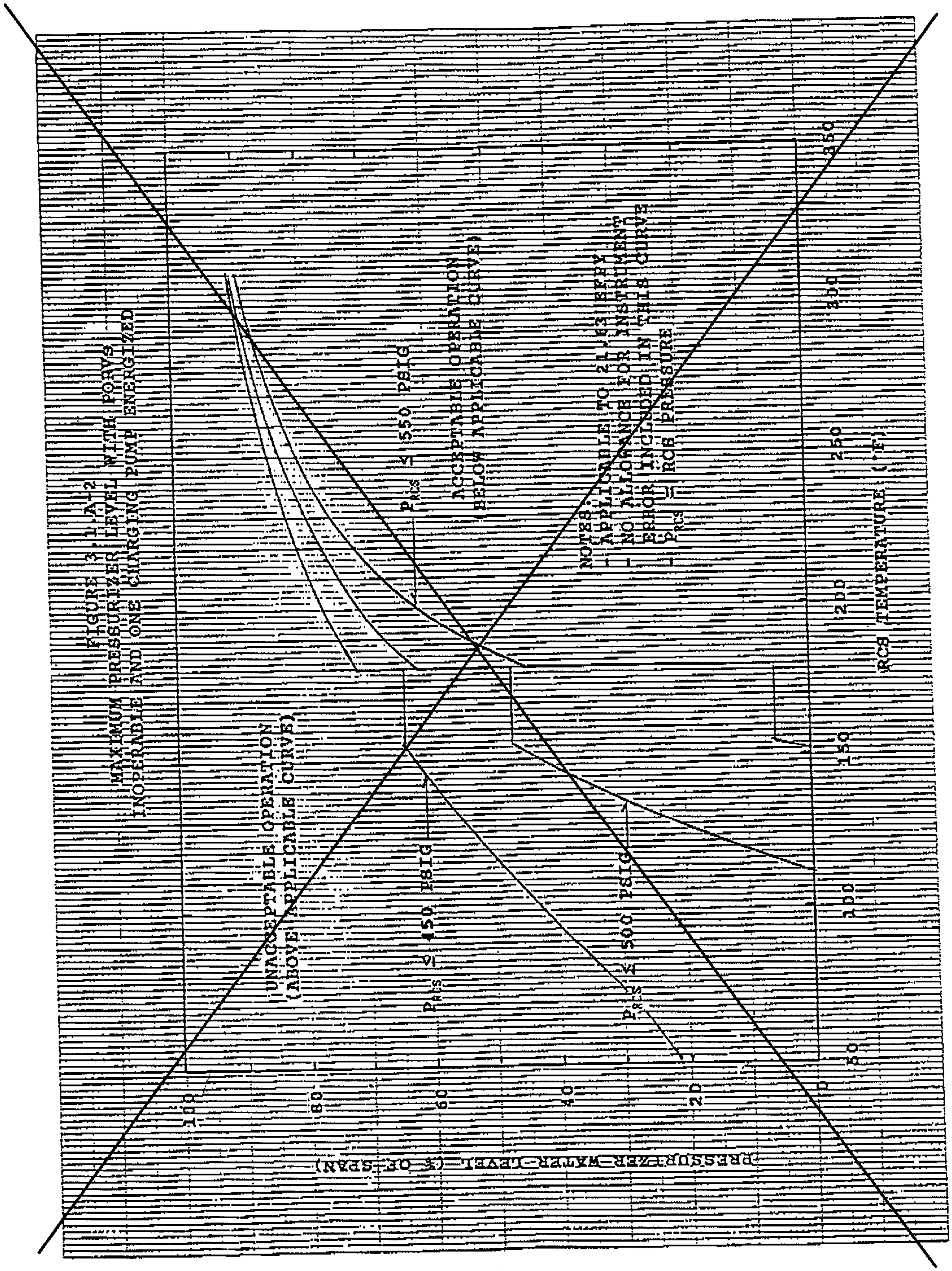
FIGURE 3.1.A-2
MAXIMUM PRESSURIZER LEVEL WITH PORVS
INOPERABLE AND ONE CHARGING PUMP ENERGIZED
UNACCEPTABLE OPERATION
(ABOVE APPLICABLE CURVE)
ACCEPTABLE OPERATION
(BELOW APPLICABLE CURVE)
P RCS ≤ 550 PSIG
P RCS ≤ 450 PSIG
P RCS ≤ 500 PSIG
NOTES:
- APPLICABLE TO 21.63 EFPY
- NO ALLOWANCE FOR INSTRUMENT
ERROR INCLUDED IN THIS CURVE
- P RCS = RCS PRESSURE
PRESSURIZER WATER LEVEL (% OF SPAN)
100
80
60
40
20
0
50
100
150
200
250
300
350
RCS TEMPERATURE (°F)

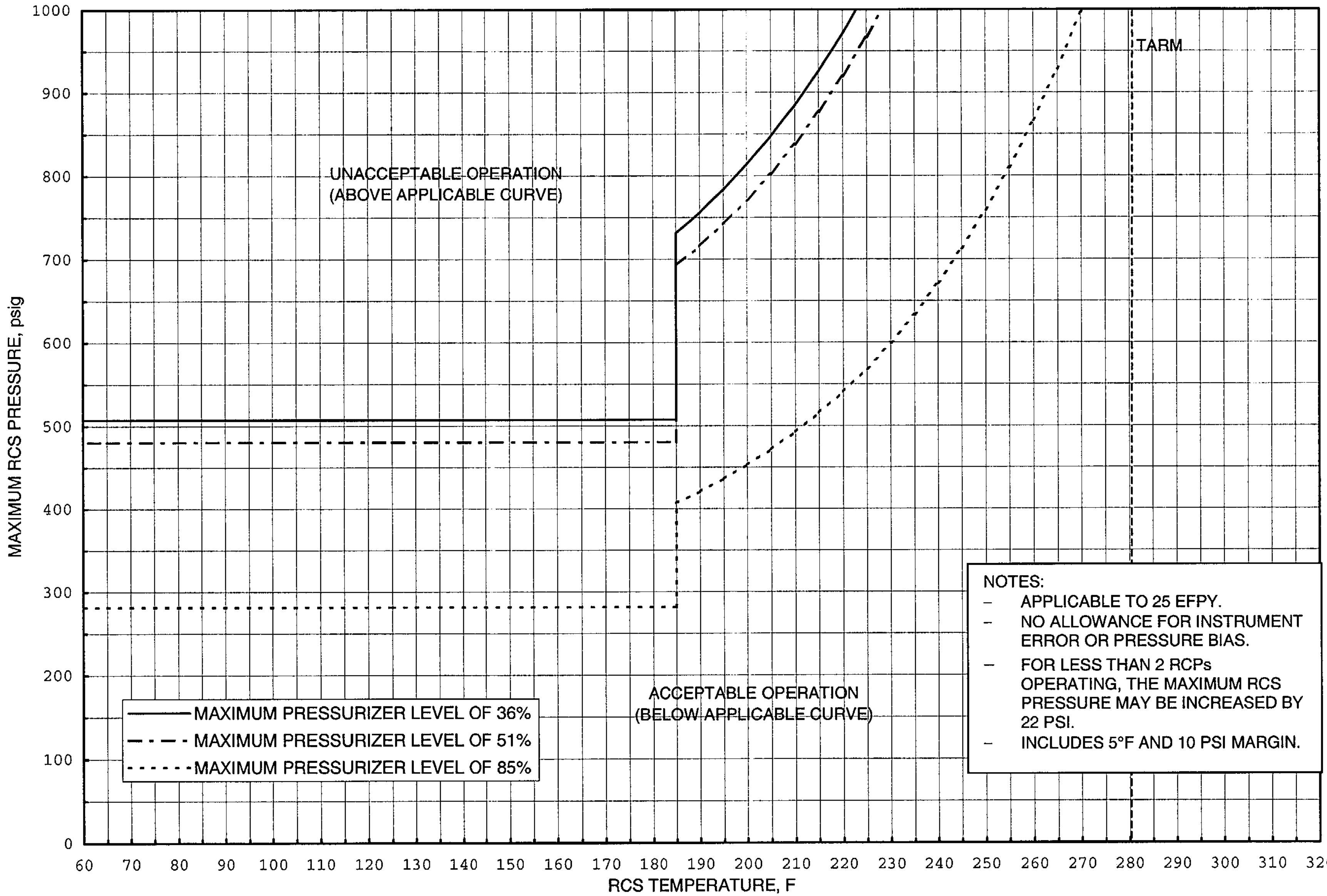

Figure 3.1.A-2 Maximum RCS Pressure: OPS Inoperable and 1 Charging Pump Capable of Injecting into the RCS

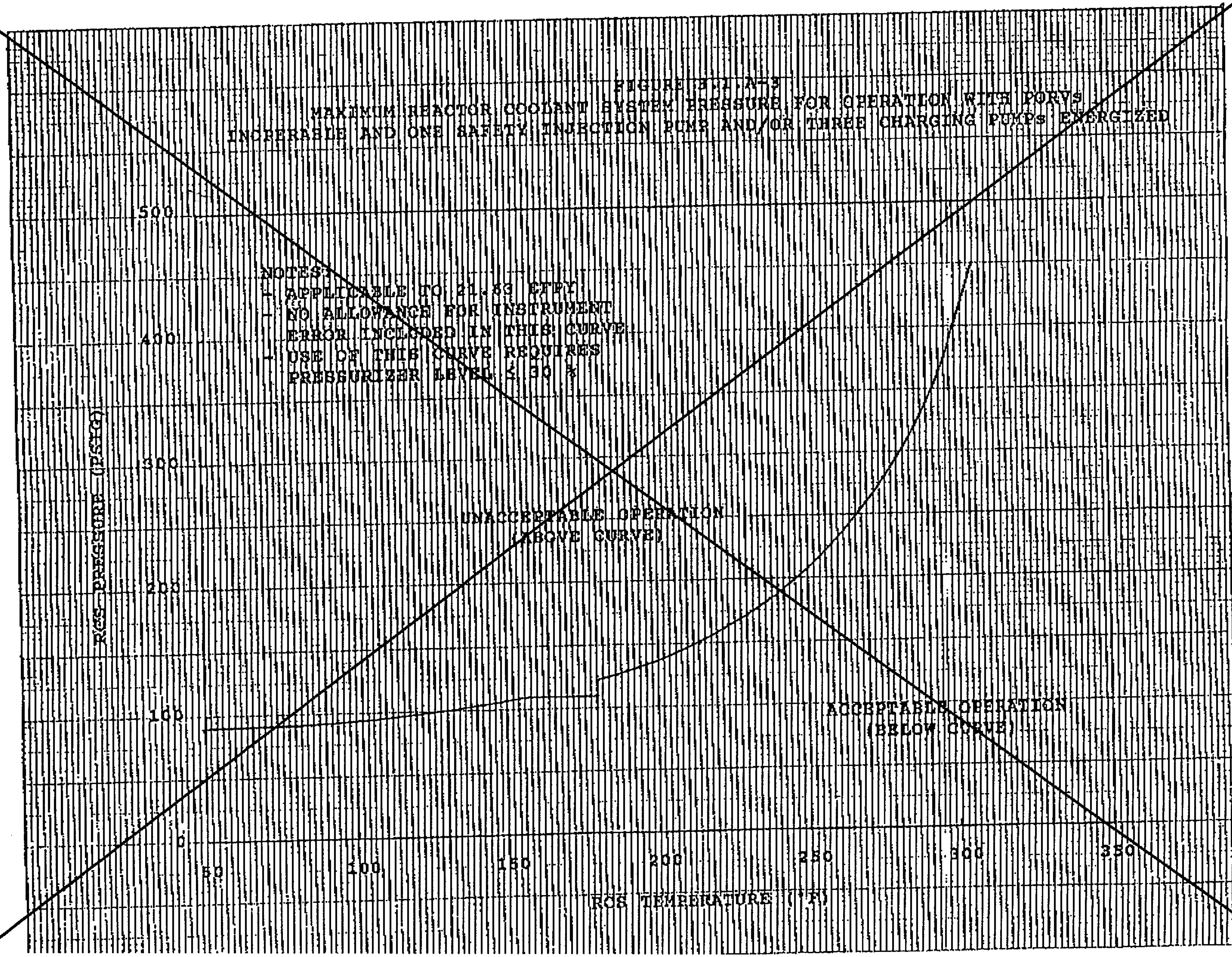
FIGURE 3.1.A-3
MAXIMUM REACTOR COOLANT SYSTEM PRESSURE FOR OPERATION WITH PORVs
INOPERABLE AND ONE SAFETY INJECTION PUMP AND/OR THREE CHARGING PUMPS ENERGIZED
NOTES:
- APPLICABLE TO 21.83 EFPY
- NO ALLOWANCE FOR INSTRUMENT
ERROR INCLUDED IN THIS CURVE
- USE OF THIS CURVE REQUIRES
PRESSURIZER LEVEL ≤ 30 %
UNACCEPTABLE OPERATION
(ABOVE CURVE)
ACCEPTABLE OPERATION
(BELOW CURVE)
RCS PRESSURE (psig)
500
400
300
200
100
0
50
100
150
200
250
300
350
RCS TEMPERATURE (°F)

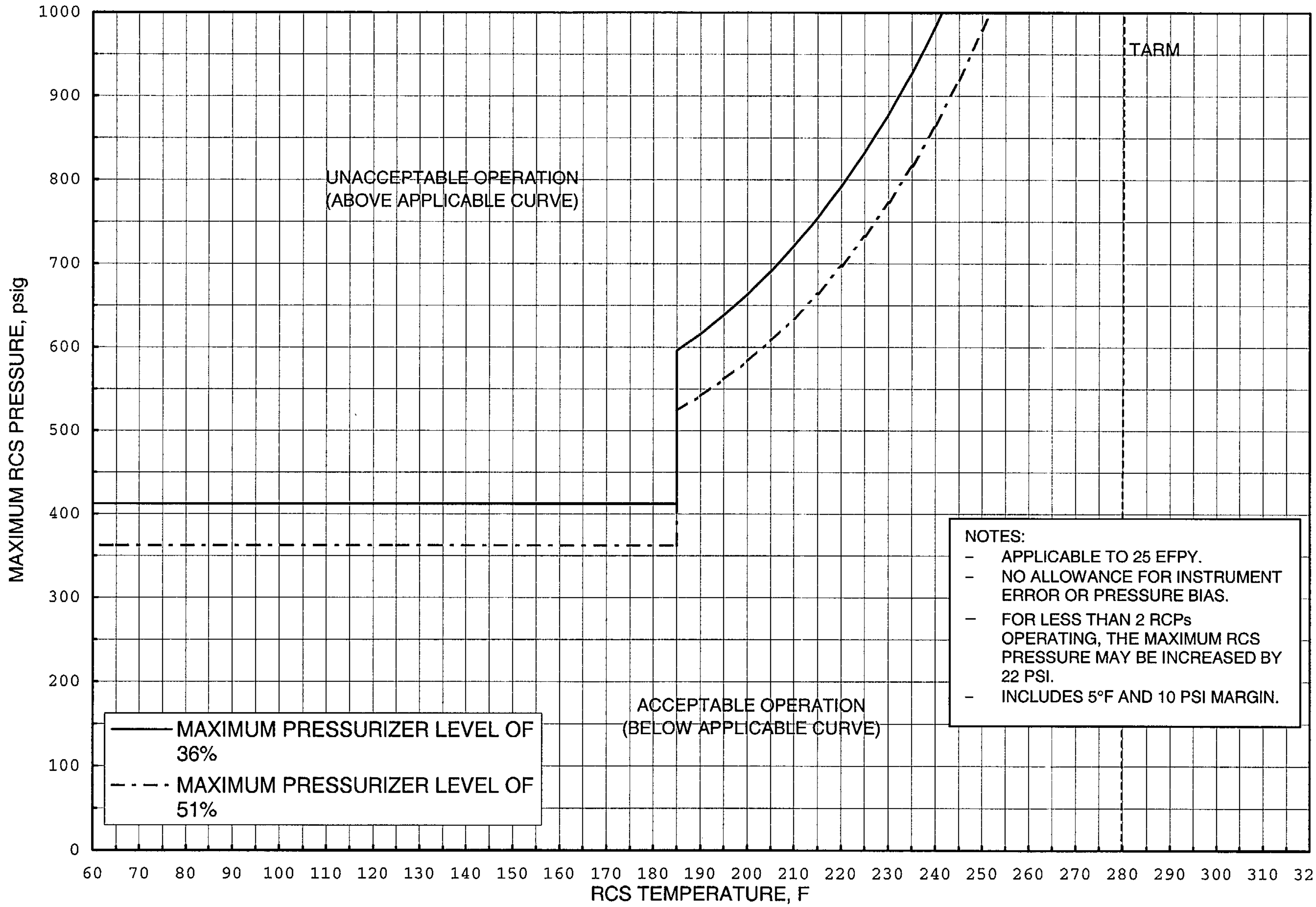

Figure 3.1.A-3 Maximum RCS Pressure: OPS Inoperable and 2 Charging Pumps Capable of Injecting into the RCS

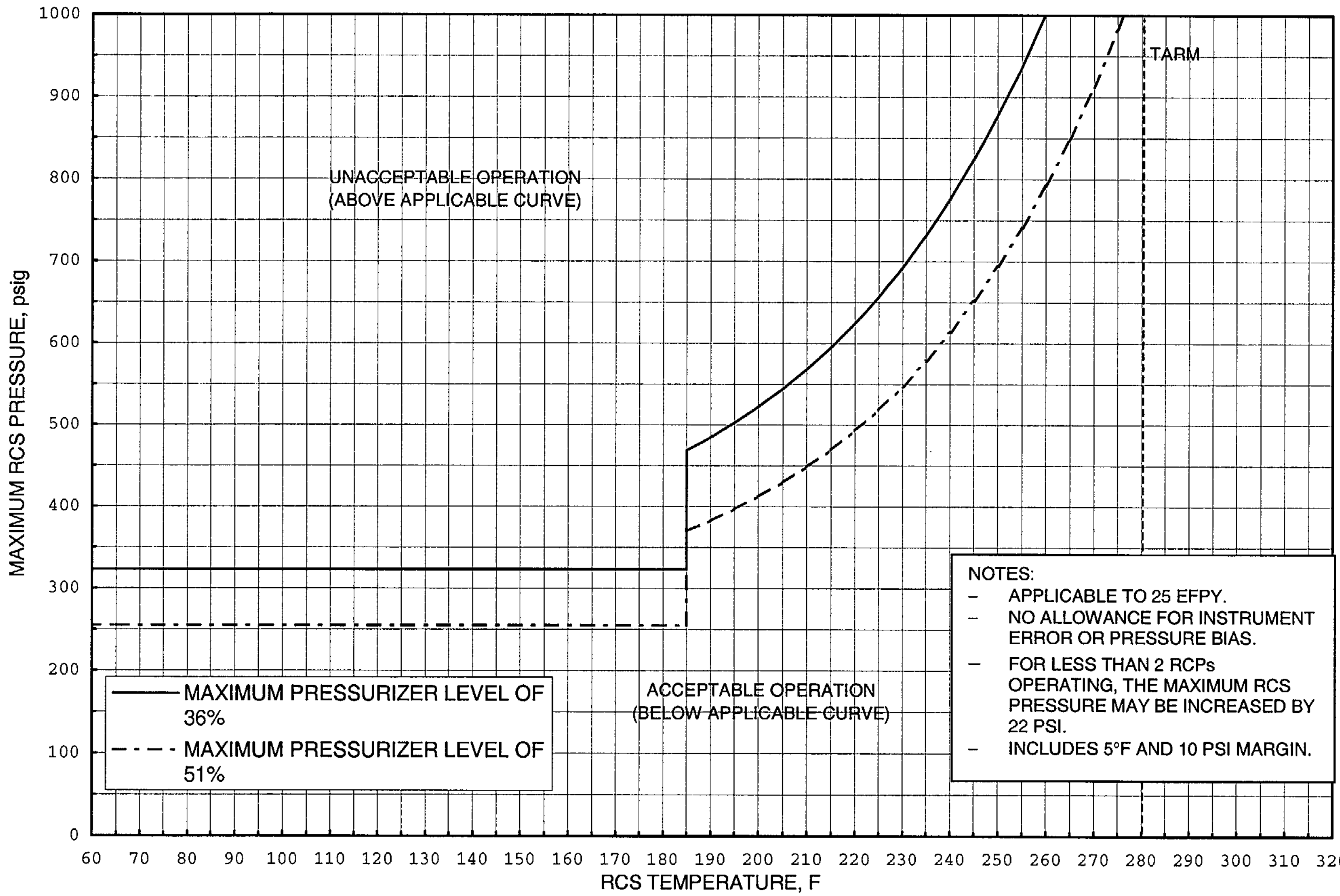

Figure 3.1.A-4 Maximum RCS Pressure: OPS Inoperable and 3 Charging Pumps Capable of Injecting into the RCS

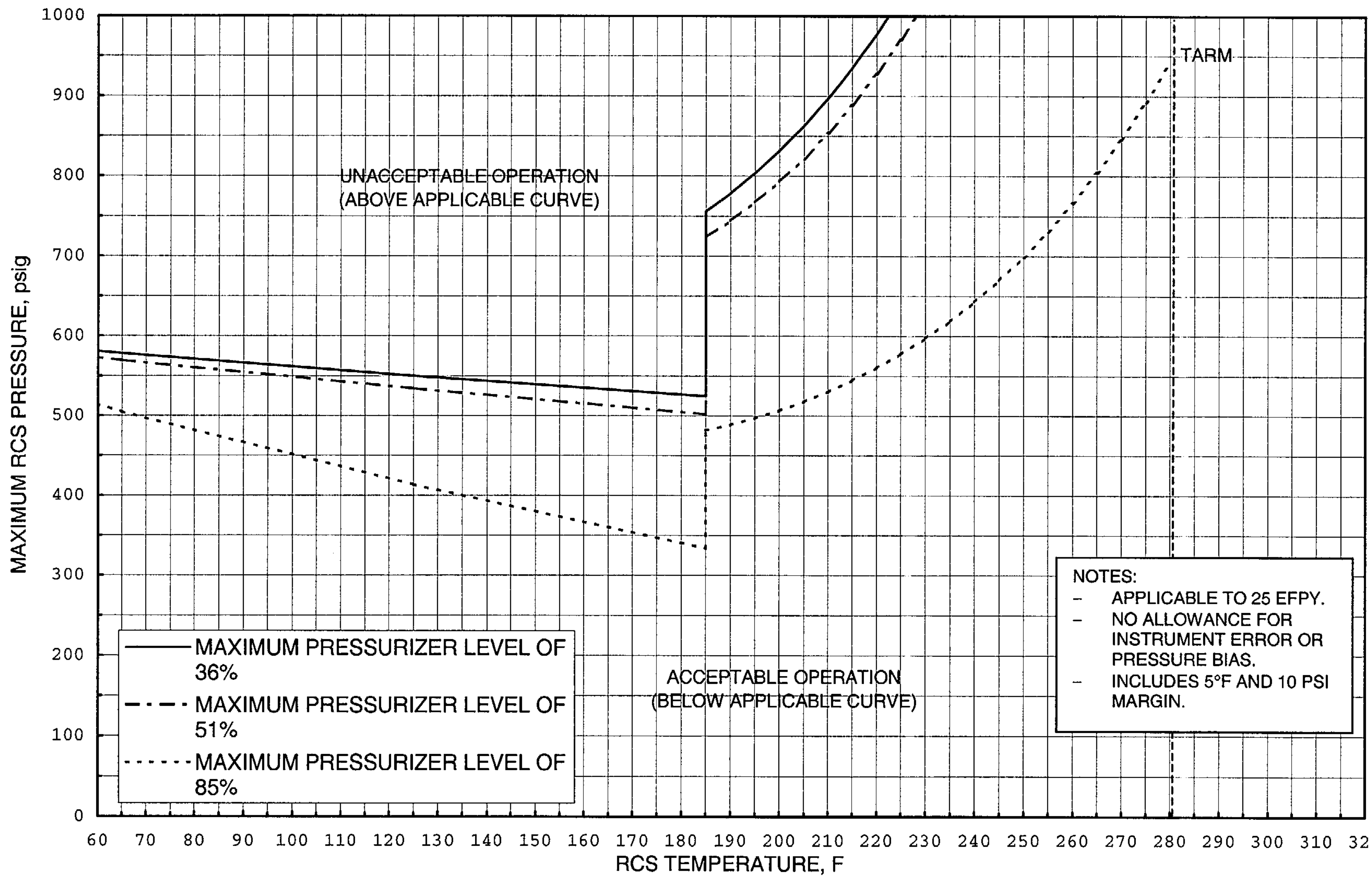

Figure 3.1.A-5 Maximum RCS Pressure: OPS Inoperable and Start of 1 RCP with SGs 40°F Hotter than RCS

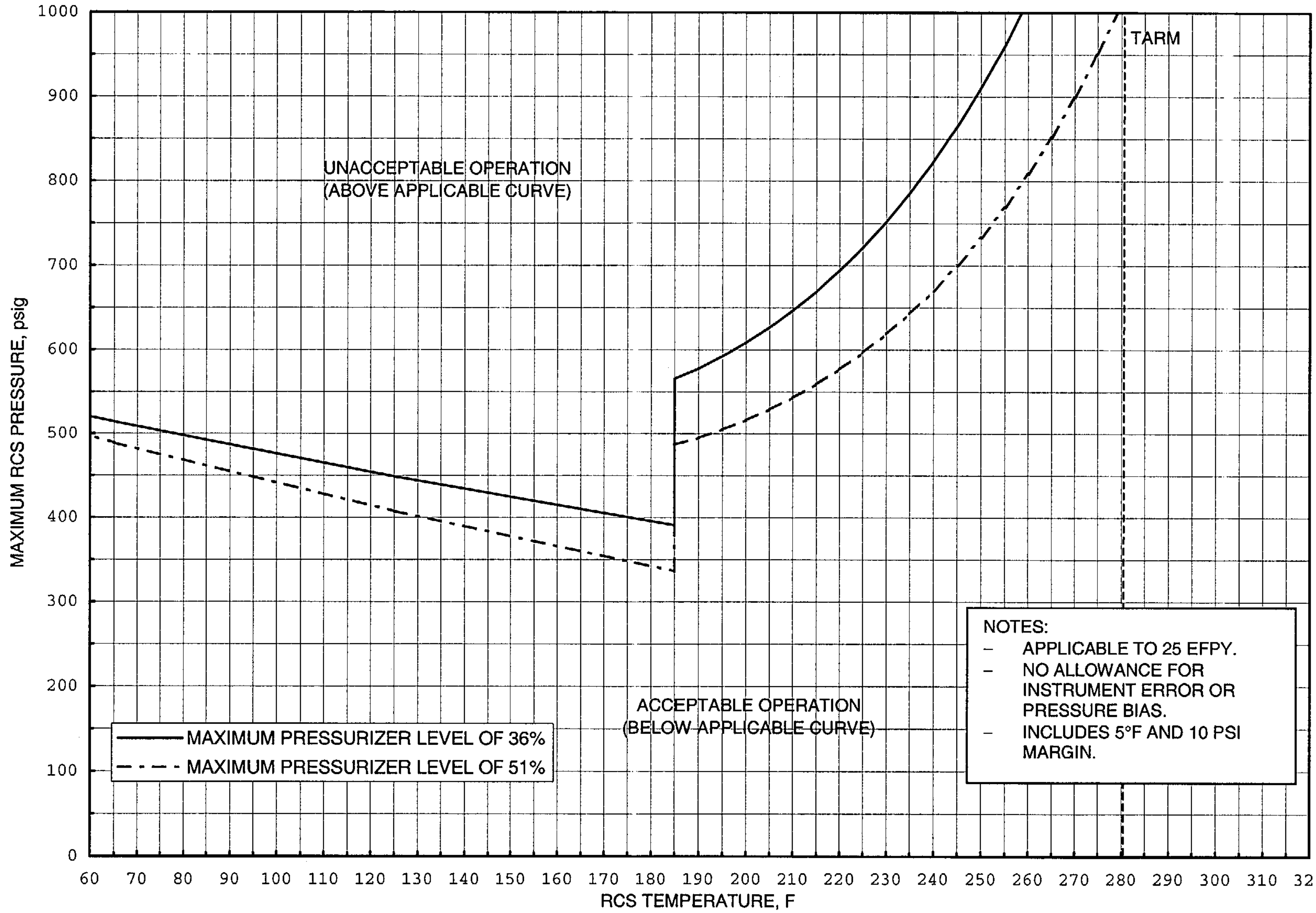

Figure 3.1.A-6 Maximum RCS Pressure: OPS Inoperable and Start of 1 RCP with SGs 100°F Hotter than RCS

B. <u>HEATUP AND COOLDOWN</u>

Specifications

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) averaged over one hour shall be limited in accordance with Figure 3.1.B-1 and Figure 3.1.B-2 for the service period up to ~~21.63~~ 25 effective full-power years. The heatup or cooldown rate shall not exceed 100°F/hr.

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

   b. Figure 3.1.B-1 and Figure 3.1.B-2 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. The limit lines shown in Figure 3.1.B-1 and Figure 3.1.B-2 shall be recalculated periodically using NRC approved methods ~~discussed in WCAP-7924A and WCAP-12796~~ and results of surveillance specimen testing as covered in WCAP-7323(7) and as specified in Specification 3.1.B.3 below. The order of specimen removal may be modified based on the results of testing of previously removed specimens. The NRC will be notified in writing as to any deviations from the recommended removal schedule no later than six months prior to scheduled specimen removal.

3. The reactor vessel surveillance program* includes six specimen capsules to evaluate radiation damage based on pre-irradiation and post-irradiation tensile and Charpy V notch (wedge open loading) testing of specimens.

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* Refer to UFSAR Section 4.5, WCAP-7323, and Indian Point Unit No. 2, "Application for Amendment to Operating License," sworn to on February 3, 1981.

The specimens will be removed and examined at the following intervals:

Capsule 1	End of Cycle 1 operation
Capsule 2	End of Cycle 2 operation
Capsule 3	End of Cycle 5 operation
Capsule 4	End of Cycle 8 operation
Capsule 5	End of Cycle 16 operation
Capsule 6	Spare

4. The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator is below 70°F.

5. The pressurizer heatup and cooldown rates averaged over one hour shall not exceed 100°F/hr and 200°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

6. Reactor Coolant System integrity tests shall be performed in accordance with Section 4.3 of the Technical Specifications.

Basis

All components in the Reactor Coolant System are designed to withstand the effects of the cyclic loads due to reactor system temperature and pressure changes[1]. These cyclic loads are introduced by normal unit load transients, reactor trips, and startup and shutdown operation. The number of thermal and loading cycles used for design purposes are shown in Table 4.1-8 of the UFSAR. During unit startup and shutdown, the rates of temperature and pressure changes are limited. The maximum plant heatup and cooldown rate of 100°F per hour is consistent with the design number of cycles and satisfies stress limits for cyclic operation[2].

Heatup and cooldown limit curves define acceptable regions for normal operation. The heatup and cooldown limit curves establish operating limits that provide a margin to non-ductile failure of the reactor coolant pressure boundary. The reactor vessel is the most limiting component most subject to non-ductile failure. Therefore, the reactor vessel limits control the heatup and cooldown limits provided in Figures 3.1.B-1 and 3.1.B-2. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

Reactor Vessel Fracture Toughness Properties

The reactor vessel plate opposite the core has been purchased to a specified Charpy V-notch test result of 30 ft-lb or greater at a Nil-Ductility Transition Temperature (NDTT) of 40°F or less. The material has been tested to verify conformity to specified requirements and a NDTT value of 20°F has been determined. In addition, this plate has been 100 percent volumetrically inspected by ultrasonic test using both longitudinal and shear wave methods. The remaining material in the reactor vessel, and other Reactor Coolant System components, meet the appropriate design code requirements and specific component function[3].

As a result of fast neutron irradiation in the region of the core, there will be an increase in the Reference Nil-Ductility Transition Temperature (RT_{NDT}) with nuclear operation. The techniques used to measure and predict the integrated fast neutron ($E > 1$ Mev) fluxes at the sample location are described in Appendix 4A of the UFSAR. The calculation method used to obtain the maximum neutron ($E > 1$ Mev) exposure of the reactor vessel is identical to that described for the irradiation samples.

Since the neutron spectra at the samples and vessel inside radius are identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of reactor vessel for some later stage in plant life. The maximum exposure of the vessel will be obtained from the measured sample exposure by appropriate application of the calculated azimuthal neutron flux variation.

10CFR50 Appendix G requires the establishment of P/T limits for specific material fracture toughness requirements of the Reactor Coolant Pressure Boundary materials. 10CFR50 Appendix G requires an adequate margin to non-ductile failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section XI, Appendix G. IP2 has obtained exemptions from the requirements of 10CFR50 Appendix G to allow the use of ASME code Cases N-588 and N-640 in conjunction with ASME Code Section XI, Appendix G. The neutron embrittlement effect on the material toughness is reflected by increasing the nil ductility reference temperature (RTNDT) as exposure to neutron fluence increases. The actual shift in the RTNDT of the vessel beltline material is established periodically in accordance with the Regulatory Guide 1.99 Revision 2 requirements. The operating P/T limit curves are periodically adjusted based on the evaluation findings and the recommendations of Regulatory Guide 1.99.

The actual shift in RT_{NDT} ~~will be~~ is established periodically during plant operation by testing vessel material samples which are irradiated cumulatively by securing them ~~near the inside wall of the vessel in the core area~~ on the outside surface of the thermal shield within the active core region. These samples are removed using the specified schedule and evaluated according to ~~ASTM E185~~[6] the requirements of 10CFR50 Appendix H. ~~To compensate for any increase in the RT_{NDT} caused by irradiation, the limits on the pressure-temperature relationship are periodically changed to stay within the stress limits during heatup and cooldown, in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, 1974 Edition, Section III, Appendix G, and the calculation methods described in WCAP-7924A[4] and WCAP-12796[3] and MSE-REME-0076[14].~~

The first reactor vessel material surveillance capsule was removed during the 1976 refueling outage. That capsule was tested by Southwest Research Institute (SWRI) and the results were evaluated and reported[8,9]. The second surveillance capsule was removed during the 1978 refueling outage. That capsule has been tested by SWRI and the results have been evaluated and reported[10]. The third vessel material surveillance capsule was removed during the 1982 refueling outage. This capsule has been tested by SWRI and the results have been evaluated and reported[11]. The fourth surveillance capsule was removed during the 1987 refueling outage. This capsule has been tested by SWRI and the results have been evaluated and reported[12]. ~~Heatup and cooldown curves (Figures 3.1.B-1 and 3.1.B-2) were developed by Westinghouse[13]. These curves are essentially identical to those obtained using Appendix G methods[14].~~

The ~~current~~ 25 EFPY heatup and cooldown curves are based upon a maximum fluence of ~~0.98~~ 1.02 x 10^{19} n/cm^2 at the inner reactor vessel surface (45° angle, vessel belt line). This fluence is based upon plant operation for a nominal period of ~~21.63~~ 25 EFPYs (Operation up to Cycle 9 for 9.63 EFPYs at 2758 MWt power level, operation during Cycle 10 for 1.13 EFPY at 2758 or 2948 MWt, operation from Cycle 11 until Cycle 15 at 3071.4 MWt and operation beyond Cycle 15 at 3216 MWt). ~~and beyond Cycle 9 for 12 EFPYs at 3071.4 MWt power level and T average of 579.7 °F).~~ The curves are based on operation beyond cycle 15 with T average of 579°F. Any changes in the operating conditions could result in an ~~extension of~~ change to the allowable EFPYs, since the fluence (or ΔRT_{NDT} due to irradiation) is the controlling factor in the generation of these curves.

The ART at a fluence of ~~0.98~~ 1.02 X 10^{19} n/cm^2, (nominal ~~21.63~~ 25 EFPYs of operation) is ~~projected~~ to be ~~155.5~~ 200°F at the 1/4 T vessel wall location for the Intermediate to Lower shell girth weld and ~~105~~ 145°F at the 3/4 T vessel wall locations, ~~per~~ for Plate B2002-3 ~~the controlling plate.~~ ~~The initial value of RT_{NDT} for this plate of the IP2 reactor vessel was 21°F.~~

Heatup and Cooldown Curves

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods ~~derived from Non-Mandatory Appendix G in Section III 1974 Edition of the ASME Boiler and Pressure Vessel Code and are~~ discussed in detail in ~~WCAP-7924A[4] and~~ WCAP-~~12796~~ 15629[13] ~~and MSE-REME-0076[14]~~ and as modified by

ASME Code Cases N-588 and N-640. Also, the 1995 Edition with the 1996 Addenda of the ASME Section XI, Appendix G was used for the operating period of up to 25 EFPY.

The heatup and cooldown curves for operation up to 25 EFPY have been computed on the basis of the RT_{NDT} ~~of~~ for both the Intermediate to Lower shell girth weld and Plate B2002-3. ~~because it is anticipated that the RT_{NDT} of the reactor vessel beltline material will be highest for Plate B2002-3, at least for the above fluence[12]~~ It was determined that heatup and cooldown curves based on Plate B2002-3 are more limiting than those calculated for the girth weld. Hence, the Heatup and Cooldown Limitation curves of Figures 3.1.B-1 and 3.1.B-2 are based on the Plate B2002-3 being the limiting vessel beltline material.

The Heatup and Cooldown curves of Figures 3.1.B-1 and 3.1.B-2 are not adjusted to account for pressure and temperature instrument error. Those adjustments are made in procedures implementing the Heatup and Cooldown curves of Figures 3.1.B-1 and 3.1.B-2.

~~The approach specifies that the allowable total stress intensity factor (K_I), at any time during heatup or cooldown, cannot be greater than that shown on the K_{IR} curve[5] for the metal temperature at that time. Furthermore, the approach applies an explicit safety factor of 2.0 on the stress intensity factor induced by pressure gradients. Thus, the governing equation for the heatup-cooldown analysis is:~~

$$\cancel{2 K_{Im} + K_{It} \leq K_{IR} \qquad (1)}$$

~~where:~~

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

~~K_{It} is the stress intensity factor caused by the thermal gradients,~~

~~K_{IR} is provided by the code as a function of temperature relative to the RT_{NDT} of the material.~~

~~During the heatup analysis, Equation (1) is evaluated for two distinct situations. First, allowable pressure-temperature relationships are developed for steady state (i.e., zero rate of change of temperature) conditions assuming the presence of the code reference 1/4 T deep flaw at the ID of the pressure vessel. Due to the fact that, during heatup, the thermal gradients in the vessel wall tend to produce compressive stresses at the 1/4 T location, the tensile stresses induced by internal pressure are somewhat alleviated. Thus, a pressure-temperature curve based on steady state condition (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the 1/4 T location is treated as the governing factor.~~

~~The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which the 3/4 T location becomes the controlling factor. Unlike the situation at the 1/4 T location, at the 3/4 T position (i.e., the tip of the 1/4 T deep O.D. flaw) the thermal gradients established during heatup produce stresses which are tensile in nature, and thus tend to reinforce the pressure stresses present. These thermal stresses are, of course, dependent on both the rate of heatup and the time (or water temperature) along the heatup ramp. Furthermore, since the thermal stresses at 3/4 T are tensile and increase with increasing heatup rate, a lower bound curve similar to that described in the preceding paragraph cannot be defined. Rather, each heatup rate of interest 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 in the following fashion. First, a composite curve is constructed 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 two values taken from the curves under consideration. The composite curve is then adjusted to allow for possible errors in the pressure- and temperature-sensing instruments.~~

~~The use of the composite curve becomes mandatory in setting heatup limitations because it is possible for conditions to exist such that, over the course of the heatup ramp, the controlling analysis switches from the O.D. to the I.D. location, and the pressure limit must, at all times, be based on the most conservative case.~~

~~The cooldown analysis proceeds in the same fashion as that for heatup, with the exception that the controlling location is always at 1/4 T. The thermal gradients induced during cooldown tend to produce tensile stresses at the 1/4 T location and compressive stresses at the 3/4 T position. Thus, the I.D. flaw is clearly the worst case.~~

~~As in the case of heatup, allowable pressure-temperature relations are generated for both steady and finite cooldown rate situations. Composite limit curves are then constructed for each cooldown rate of interest. Again adjustments are made to account for pressure and temperature instrumentation error.~~

~~The use of the composite curve in the cooldown analysis is necessary because system control is based on a measurement of reactor coolant temperature, whereas the limiting pressure is calculated using the material temperature at the tip of the assumed reference flaw. During cooldown, the 1/4 T vessel location is at a higher temperature than the fluid adjacent to the vessel I.D. This condition is, of course, not true for the steady-state situation. It follows that the ΔT induced during cooldown results in a calculated higher allowable K_{IR} for finite cooldown rates than for steady state under certain conditions.~~

~~Because operation control is on coolant temperature, and cooldown rate may vary during the cooldown transient, the limit curves shown in Figure 3.1.B-2 represent a composite curve~~

~~consisting of the more conservative values calculated for steady state and the specific cooling rate shown.~~

Pressurizer Limits

Although the pressurizer operates at temperature ranges above those for which there is reason for concern about brittle fracture, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, 1965 Edition and associated Code Addenda through the Summer 1966 Addendum.

References

(1) Indian Point Unit No. 2 UFSAR, Section 4.1.5.

(2) ASME Boiler & Pressure Vessel Code, Section III, Summer 1965, N-415.

(3) Indian Point Unit No. 2 UFSAR, Section 4.2.5.

(4) WCAP-7924A, "Basis for Heatup and Cooldown Limit Curves," W. S. Hazelton, S.L. Anderson, S.E. Yanichko, April 1975.

(5) ASME Boiler and Pressure Vessel Code, Section XI, 1996 Edition, Appendix G

(6) ~~ASTM E185-79, Surveillance Tests on Structural Materials in Nuclear Reactors.~~ DELETED

(7) WCAP-7323, "Consolidated Edison Company, Indian Point Unit No. 2 Reactor Vessel Radiation Surveillance Program," S.E. Yanichko, May 1969.

(8) Final Report - SWRI Project No. 02-4531 - "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule T," E.B. Norris, June 30, 1977.

(9) Supplement to Final Report - SWRI Project No. 02-4531 - "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule T," E.B. Norris, December 1980.

(10) Final Report - SWRI Project No. 02-5212 - "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule Y," E.B. Norris, November 1980.

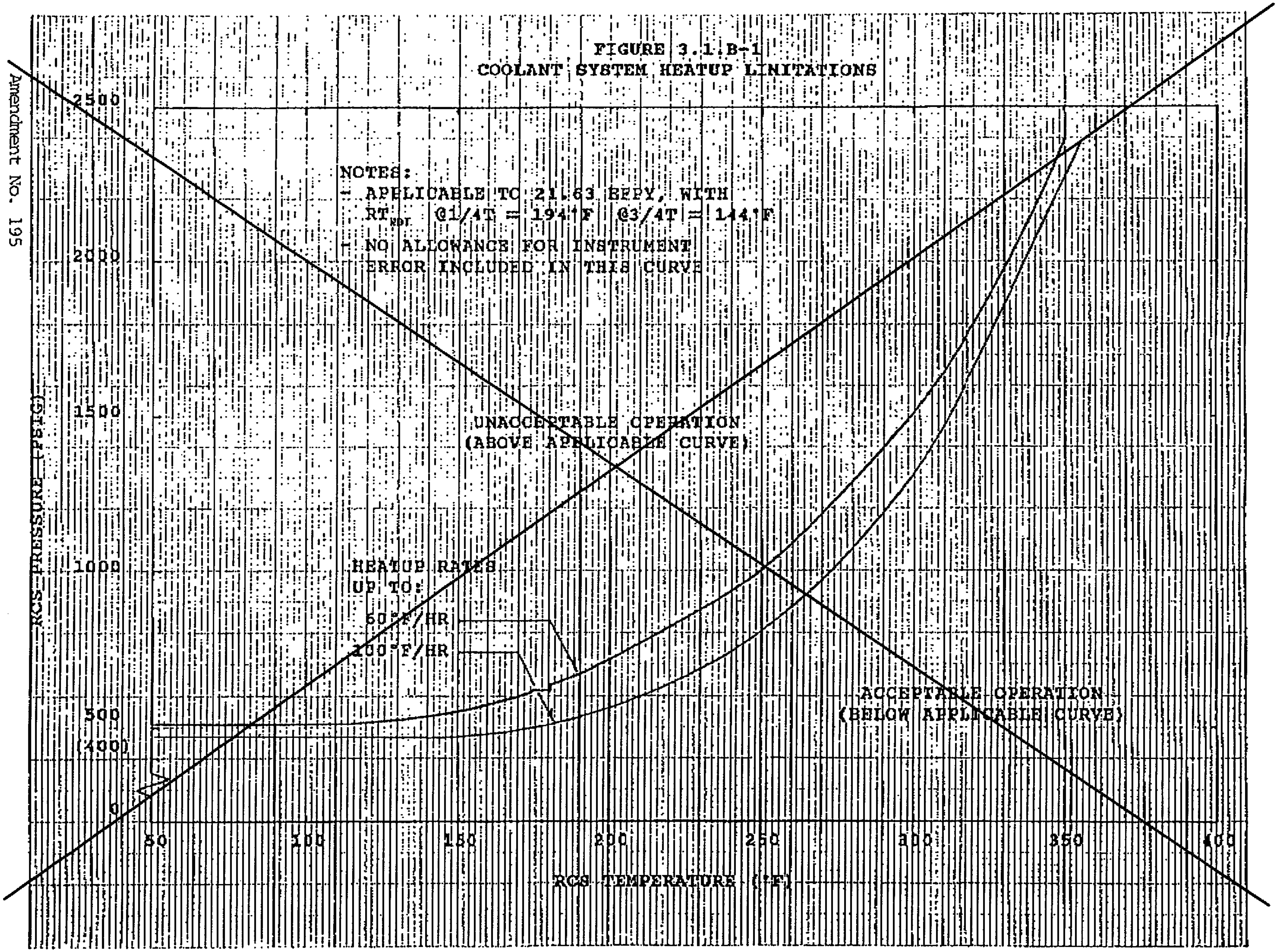
(11) Final Report - SWRI Project No. 06-7379 - "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule Z," E.B. Norris, April 1984.

(12) Final Report - SWRI Project No. 17-2108 (Revised)- "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule V," F.A. Iddings - SWRI, March, 1990.

(13) ~~WCAP-12796, "Heatup and Cooldown Limit Curves for the Consolidated Edison Company Indian Point Unit 2 Reactor Vessel," N.K. Ray, Westinghouse Electric Corporation.~~ WCAP-15629, Indian Point Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation

(14) ~~MSE-REME-0076, "Indian Point Unit 2 Heatup and Cooldown Limit Curves for Normal Operation," P.A. Grendys, Westinghouse Electric Corporation.~~ WCAP-14040-NP-A, Rev. 2, Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves.

Amendment No. 195



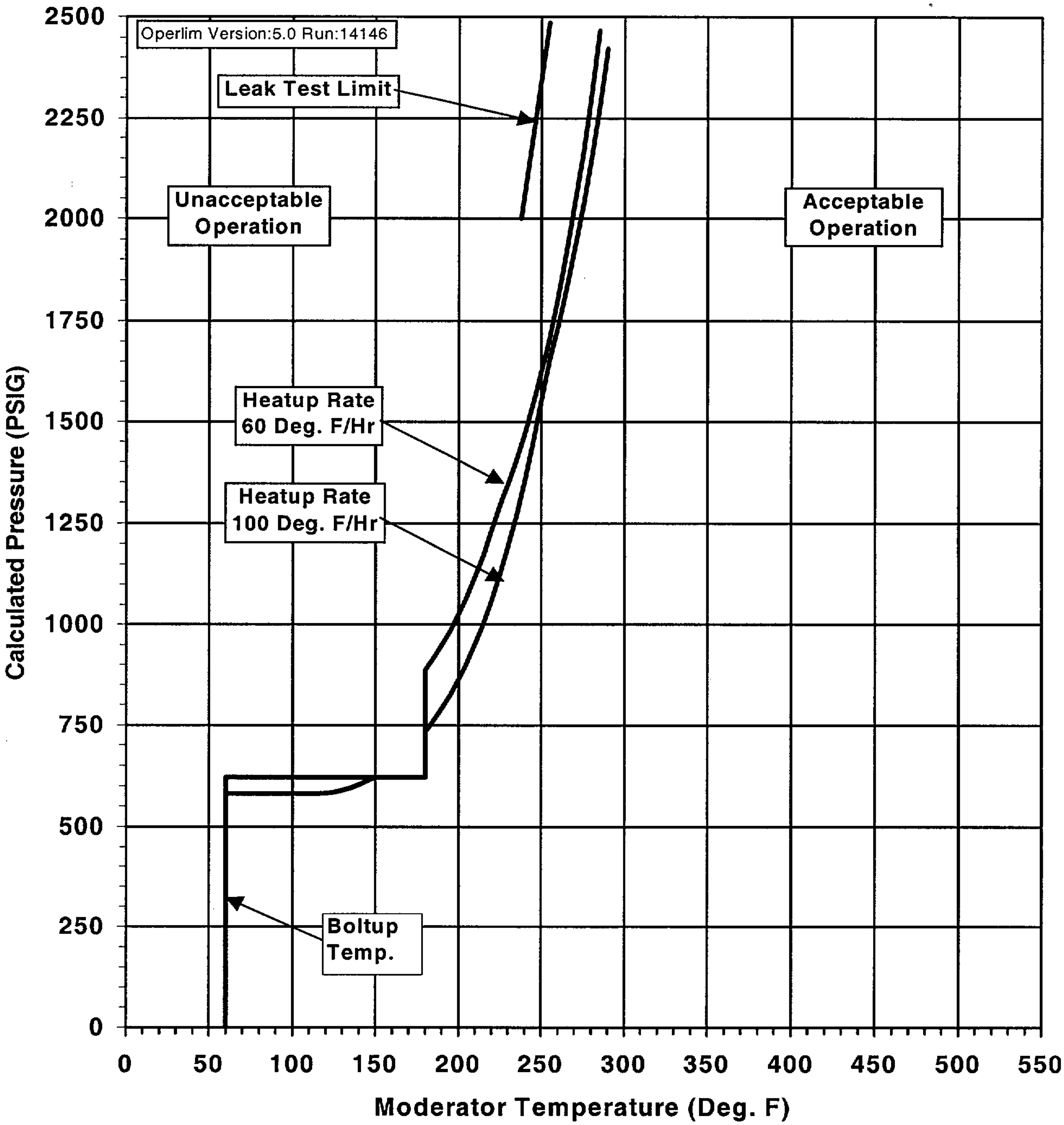
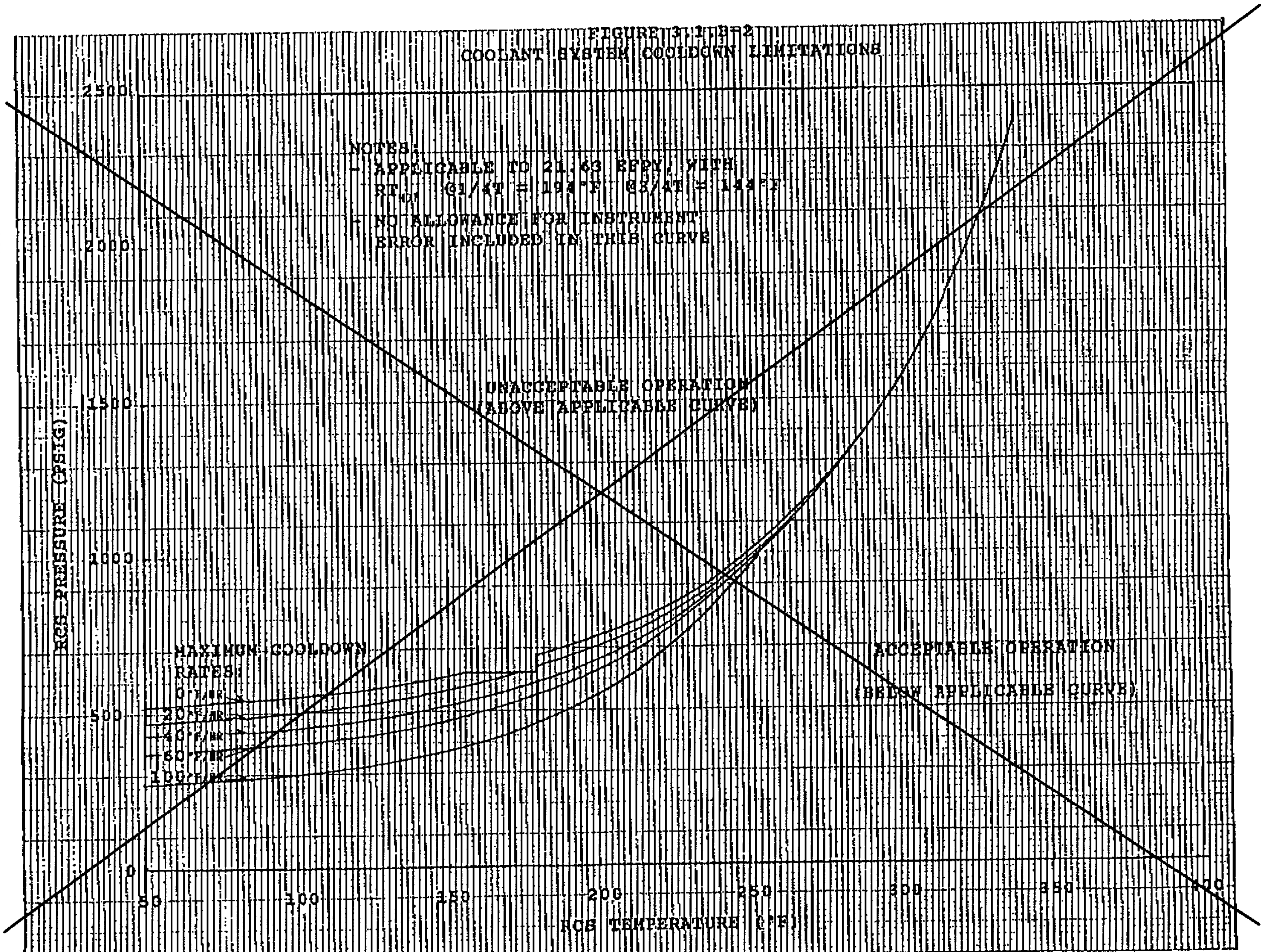


Figure 3.1.B-1 Reactor Coolant System Heatup and Leak Test Limitations Applicable for the First 25 EFPY (Without Margins for Instrumentation Errors)

Acceptable operation is to the right of or below the applicable curve. Unacceptable operation is to the left of or above the applicable curve.

FIGURE 3.1.B-2
COOLANT SYSTEM COOLDOWN LIMITATIONS



Amendment No. 195

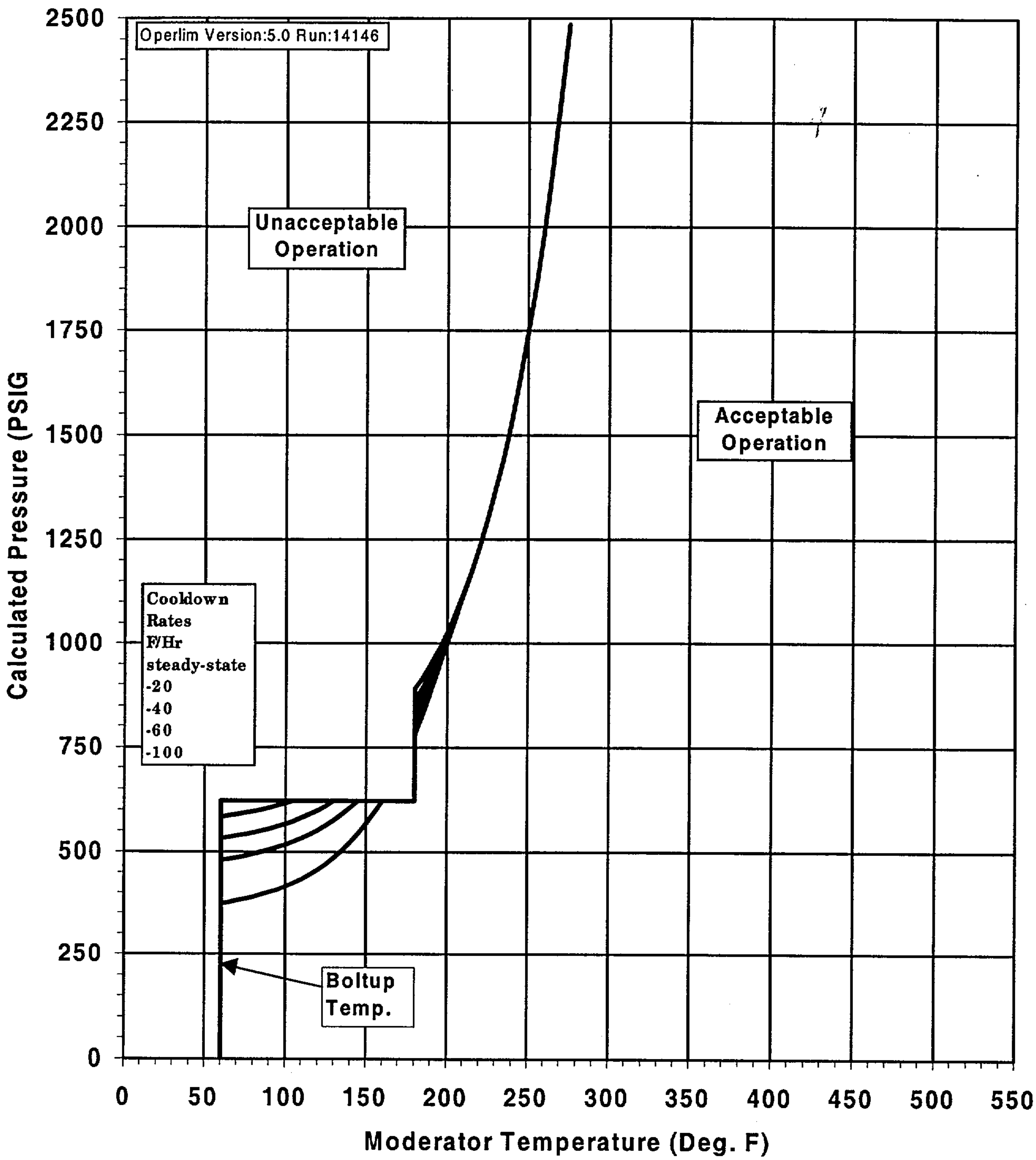


Figure 3.1.B-2 Reactor Coolant System Cooldown Limitations Applicable for the First 25 EFPY (Without Margins for Instrumentation Errors)

Acceptable operation is to the right of or below the applicable curve.
Unacceptable operation is to the left of or above the applicable curve.

Amendment

physics tests to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special operating precautions will be taken.

The requirement that the reactor is not to be made critical below 450ºF provides increased assurance that the proper relationship between reactor coolant pressure and temperature will be maintained during system heatup and pressurization in accordance with the requirements of 10 CFR 50 Appendix G~~, as amended February 2, 1976~~. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.

If the shutdown margin specified in 3.1.C.3 is maintained, there is no possibility of an accidental criticality as a result of a decrease of coolant pressure.

The requirement for bubble formation in the pressurizer when the reactor has passed the threshold of 1% subcriticality will assure that the Reactor Coolant System will not be solid when criticality is achieved.

<u>References</u>

(1) UFSAR Section 3.2

C. During power operation, the requirements of 3.2.B may be modified to allow any one of the following components to be inoperable. If the system is not restored to meet the requirements of 3.2.B within the time period specified, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the requirements of 3.2.B are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

1. One of the two operable charging pumps may be removed from service provided a second charging pump is restored to operable status within 24 hours.

2. The boric acid storage system (including the boric acid transfer pumps) may be inoperable provided the RWST is operable and provided that the boric acid storage system and at least one boric acid transfer pump is restored to operable status within 48 hours.

3. One channel of heat tracing for the flow path from the boric acid storage system to the Reactor Coolant System may be out of service provided the failed channel is restored to an operable status within 7 days and the redundant channel is operable during that period.

4. Both channels of heat tracing for the flow path from the boric acid storage system to the Reactor Coolant System may be out of service provided at least one channel is restored to operable status within 48 hours, the required flow path is shown to be clear of blockage, and the second channel is restored to operable status within 7 days.

D. When RCS temperature is less than or equal to ~~305~~ 280°F, the requirements of Table 3.1.A-2 regarding the number of charging pumps allowed to be ~~energized~~ capable of injecting into the RCS shall be adhered to.

d. Any valve required for the functioning of the system during and following accident conditions may be inoperable provided that it is restored to operable status within 24 hours and all valves in the system that provide the duplicate function are operable.

e. Deleted

f. One refueling water storage tank low-level alarm may be inoperable for up to 7 days provided the other low-level alarm is operable.

3. When RCS temperature is less than or equal to ~~305~~ 280°F, the requirements of Table 3.1.A-2 regarding the number of safety injection (SI) pumps allowed to be ~~energized~~ capable of injecting into the RCS shall be adhered to.

B. CONTAINMENT COOLING AND IODINE REMOVAL SYSTEMS

1. The reactor shall not be made critical unless the following conditions are met:

a. The recirculation fluid pH control system shall be operable with ≥ 8000 lbs. (148 cu. ft.) of trisodium phosphate (w/12 hydrates), or equivalent, available in storage baskets in the containment.

b. The five fan cooler units and the two spray pumps, with their associated valves and piping, are operable.

2. During power operation, the requirements of 3.3.B.1 may be modified to allow any one of the following components to be inoperable. If the system is not restored to meet the requirements of 3.3.B.1 within the time period specified, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the requirements of 3.3.B.1 are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

a. One fan cooler unit may be inoperable during normal reactor operation for a period not to exceed 7 days provided both containment spray pumps are operable.

b. One containment spray pump may be inoperable during normal reactor operation, for a period not to exceed 72 hours, provided the five fan cooler units and the remaining containment spray pump are operable.

1. assurance with high reliability that the safeguard system will function properly if required to do so, and

2. allowance of sufficient time to effect repairs using safe and proper procedures.

Assuming the reactor has been operating at full-rated power for at least 100 days, the magnitude of the decay heat decreases after initiating hot shutdown. Thus the requirement for core cooling in case of a postulated loss-of-coolant accident while in the hot shutdown condition is significantly reduced below the requirements for a postulated loss-of-coolant accident during power operation. Putting the reactor in the hot shutdown condition significantly reduces the potential consequences of a loss-of-coolant accident, and also allows more free access to some of the engineered safeguards components in order to effect repairs.

Failure to complete repairs within 48 hours of going to the hot shutdown condition is considered indicative of a requirement for major maintenance, and therefore in such a case the reactor is to be put into the cold shutdown condition.

Valves 1810, 744 and 882 are kept in the open position during plant operation to assure that flow passage from the refueling water storage tank will be available during the injection phase of a loss-of-coolant accident. As an additional assurance of flow passage availability, the valve motor operators are de-energized to prevent an extremely unlikely spurious closure of these valves to take place. This additional precaution is acceptable since failure to manually re-establish power to close valves 1810 and 882, following the injection phase, is tolerable as a single failure. Valve 744 will not need to be closed following the injection phase. The accumulator isolation valve motor operators are de-energized to prevent an extremely unlikely spurious closure of these valves from occurring when accumulator core cooling flow is required.

With respect to the core cooling function, there is some functional redundancy for certain ranges of break sizes. The measure of effectiveness of the Safety Injection System is the ability of the pumps and accumulators to keep the core flooded or to reflood the core rapidly where the core has been uncovered for postulated large area ruptures. The result of the performance is to sufficiently limit any increase in clad temperature below a value where emergency core cooling objectives are met[9]. The range of core protection as a function of break diameter provided by the various components of the Safety Injection System is presented in Figure 6.2-9 of the UFSAR.

The requirement regarding the maximum number of SI pumps that can be ~~energized~~ capable of injecting into the RCS when RCS temperature is less than or equal to ~~305~~ 280°F is discussed under Specification 3.1.A.

Table 4.1-1

Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels

Footnotes:

*1 By means of the movable incore detector system.

*2 Prior to each reactor startup if not done previous week.

*3 Monthly visual inspection of condensate weirs only.

*4 Within 31 days prior to entering a condition in which the Control Rod Protection System is required to be operable unless the reactor trip breakers are manually opened during RCS cooldown prior to T_{cold} decreasing below 381°F and the breakers are maintained opened during RCS cooldown when T_{cold} is less than 381°F.

*5 Except when block valve operator is deenergized.

*6 ~~Within 31 days prior to entering a condition in which OPS is required to be operable or and at~~ At monthly intervals ~~thereafter~~ when OPS is required to be operable. Not required to be performed until 12 hours after entering a condition in which OPS is required to be operable if performed within the prior 24 months.

*7 Acceptable criteria for calibration are provided in Table II.F-13 of NUREG-0737.

*8 Calibration will be performed using calibration span gas.

*9 Each train shall be tested at least every 62 days on a staggered test basis (i.e., one train per month).

4.3 REACTOR COOLANT SYSTEM INTEGRITY TESTING

Applicability

Applies to test requirements for Reactor Coolant System integrity.

Objective

To specify tests for Reactor Coolant System integrity after the system is closed following refueling, repair, replacement or modification.

Specifications

a. The Reactor Coolant System shall be tested for leakage at normal operating pressure prior to plant startup following each refueling outage, in accordance with the requirements of the applicable edition and addenda of the ASME Section XI Code.

b. Testing of repairs, replacements or modifications for the Reactor Coolant System shall meet the requirements of the applicable edition and addenda of the ASME Section XI Code.

c. The Reactor Coolant System leak test temperature-pressure relationship shall be in accordance with the limits of Figure ~~4.3-1~~ 3.1.B-1 for heatup for the first ~~21.63~~ 25 effective full-power years of operation. Figure ~~4.3-1~~ 3.1.B-1 will be recalculated periodically. Allowable pressure during cooldown for the leak test temperature shall be in accordance with Figure 3.1.B-2.

Basis

Leak test of the Reactor Coolant System is required by the ASME Boiler and Pressure Vessel Code, Section XI, to ensure leak tightness of the system during ~~operaton~~ operation. The test frequency and conditions are specified in the Code.

For repairs on components, the thorough non-destructive testing gives a very high degree of confidence in the integrity of the system, and will detect any significant defects in and near the new welds. In all cases, the leak test will assure leak-tightness during normal operation.

The inservice leak temperatures are shown on Figure ~~4.3-1~~ 3.1.B-1. The temperatures are calculated in accordance with ~~ASME Code Section III, 1974 Edition, Appendix G and~~ the methods described in ~~reference 13~~ the Basis of Technical Specification 3.1.B. ~~This code requires that a safety factor of 1.5 times the stress intensity factor caused by pressure be applied to the calculation.~~

~~For the first 21.63 effective full-power years, it is predicted that the highest RT_{NDT} in the core region taken at the 1/4 thickness will be 194°F.~~ The minimum inservice leak test temperature requirements for periods up to ~~21.63~~ 25 effective full-power years are shown on Figure ~~4.3-1~~ 3.1.B-1.

The heatup limits specified on the heatup curve, Figure ~~4.3-1~~ 3.1.B-1, must not be exceeded while the reactor coolant is being heated to the inservice leak test temperature. For cooldown from the leak test temperature, the limitations of Figure 3.1.B-2 must not be exceeded. Figures ~~4.3-1~~ 3.1.B-1 and 3.1.B-2 are recalculated periodically, using methods discussed in the Basis of Technical Specification 3.1.B. ~~WCAP-7924A, WCAP-12796 and MSE-REME-0076 and results of surveillance specimen testing, as covered in WCAP-7323.~~

~~The current heatup and cooldown curves are based upon a maximum fluence of 0.98 x 10^{19} n/cm^2 at the inner reactor vessel surface (45° angle, vessel belt line). This fluence is based upon plant operaton for a nominal period of 21.63 EFPYs (Operation up to Cycle 9 for 9.63 EFPYs at 2758 MWt power level and beyond Cycle 9 for 12 EFPYs at 3071.4 MWt power level and T average of 579.7°F). Any changes in the operating conditions could result in an extension of the allowable EFPYs, since the fluence (or ΔRT_{NDT} due to irradiation) is the controlling factor in the generation of these curves.~~

<u>Reference</u>

UFSAR Section 4

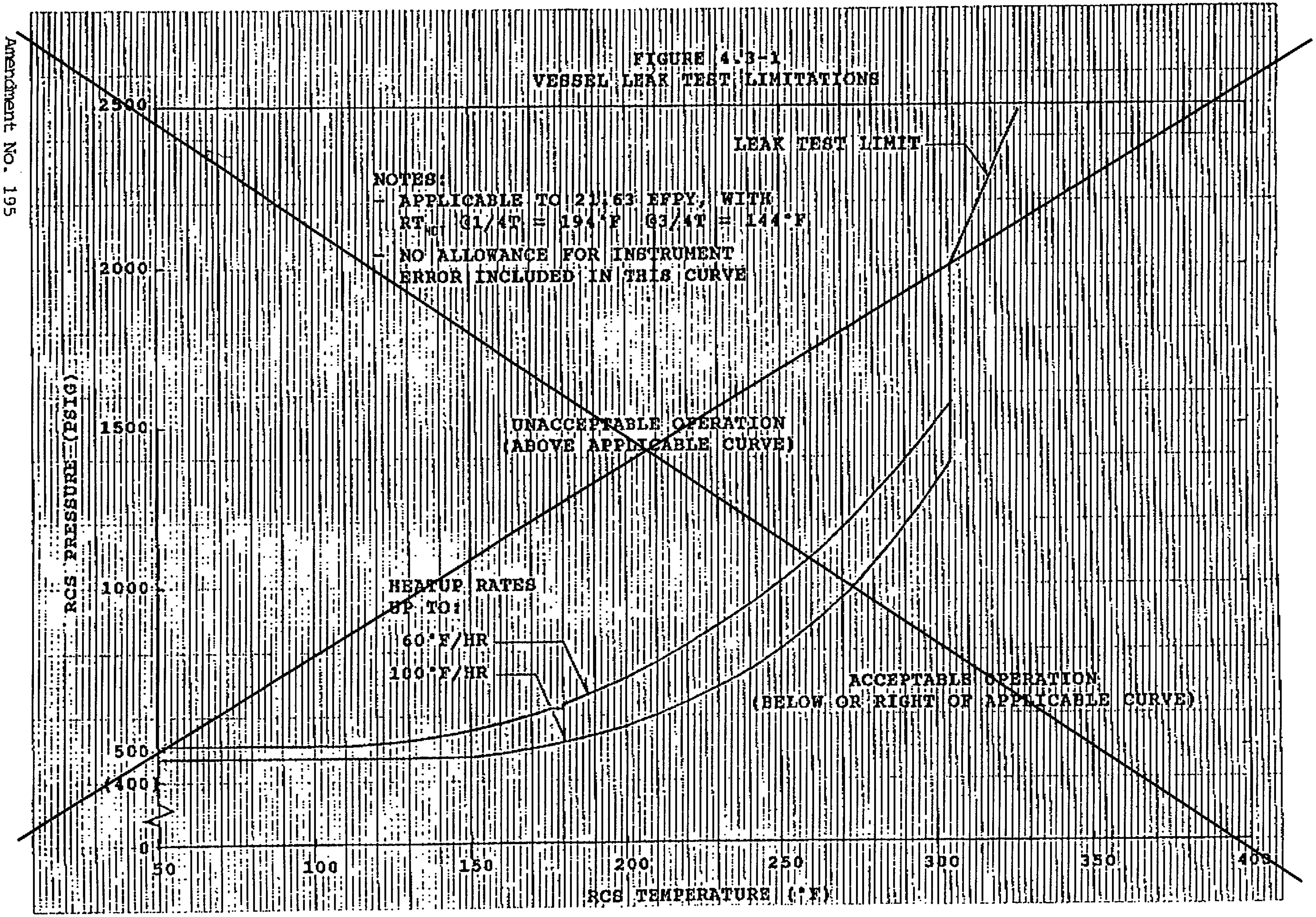


FIGURE 4.3-1
VESSEL LEAK TEST LIMITATIONS
LEAK TEST LIMIT
NOTES:
- APPLICABLE TO 21.63 EFPY, WITH
RT$_{NDT}$ @1/4T = 194°F @3/4T = 144°F
- NO ALLOWANCE FOR INSTRUMENT
ERROR INCLUDED IN THIS CURVE
UNACCEPTABLE OPERATION
(ABOVE APPLICABLE CURVE)
HEATUP RATES
UP TO:
60°F/HR
100°F/HR
ACCEPTABLE OPERATION
(BELOW OR RIGHT OF APPLICABLE CURVE)
RCS PRESSURE (PSIG)
2500
2000
1500
1000
500
(400)
0
RCS TEMPERATURE (°F)
50
100
150
200
250
300
350
400

4.18 OVERPRESSURE PROTECTION SYSTEM

Applicability

This specification applies to the surveillance requirements for the OPS provided for prevention of RCS overpressurization.

Objective

To verify the operability of OPS.

Specifications

A. When the OPS PORVs are being used for overpressure protection as required by Specification 3.1.A.4, their associated series MOVs shall be verified to be open at least twice weekly with a maximum time between checks of 5 days.

B. When RCS venting is being used for overpressure protection as permitted by Specification 3.1.A.4, the vent(s) shall be verified to be open at least daily. When the venting pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then only these valves need be verified to be open at monthly intervals.

C. When pressurizer pressure and level control is being used for overpressure protection, as permitted by Specification 3.1.A.4, then these parameters shall be verified to be within their limits at least once per shift.

D. When safety injection pumps and/or charging pumps are required to be ~~de-energized~~ not capable of injecting into the RCS per Specification 3.1.A.4, the pumps shall be demonstrated to be inoperable at monthly intervals by verifying lockout of the pump circuit breakers at the 480 volt switchgear, or once per shift if other means of ~~de-energizing~~ making the pumps not capable of injecting into the RCS are used.

E. The PORV backup nitrogen system shall be demonstrated to be operable at Refueling Intervals (#).

ATTACHMENT 3 TO NL 01-092

Information Supporting a Request for Exemption from the Requirements of 10CFR50.60(a) and Appendix G

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247

REQUEST FOR EXEMPTION FROM 10CFR50.60(A) AND 10CFR50 APPENDIX G

In accordance with 10CFR50.12, "Specific Exemptions," the Consolidated Edison Company of new York, Inc, (Con Edison) is requesting an exemption from the requirements of 10CFR50.60(a)," Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation." The exemption would permit the use of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI Code Case N-640, "Alternative Requirement Fracture Toughness for Development of P-T Limit Curves for ASME Section XI, Division 1," and ASME B&PV Section XI Code Case N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels, Section XI, Division 1," in lieu of 10CFR50 Appendix G, paragraph I.

JUSTIFICATION FOR USE OF CODE CASE N-640

10CFR50.12(a) Requirements

The requested exemption to allow use of ASME Code Case N-640 in conjunction with ASME B&PV Code Section XI, Appendix G to determine the pressure-temperature (P-T) limits for the Reactor Pressure Vessel (RPV) meets the criteria of 10CFR50.12 as discussed below.

10CFR50.12 states that the commission may grant an exemption from requirements contained in 10CFR50 provided that the following is met:

1. The requested exemption is authorized by law.

   No law exists which precludes the activities covered by this exemption request. 10CFR50.60(b) explicitly allows the use of alternatives to 10CFR50 Appendices G and H when an exemption is granted by the Commission under 10CFR50.12.

2. The requested exemption does not present an undue risk to the public health and safety.

   The revised P-T limits being proposed for IP2 rely in part on the requested exemption. These revised P-T limits have been developed using the K_{Ic} fracture toughness curve shown on ASME XI, Appendix A, Figure A-4200-1, in lieu of the K_{Ia} fracture toughness curve of ASME XI, Appendix G, Figure G-2210-1, as the lower bound for fracture toughness. The other margins involved with the ASME B&PV Code, Section XI, Appendix G process of determining P-T limit curves remain unchanged.

   Use of the K_{Ic} curve in determining the lower bound fracture toughness in the development of P-T operating limits curve is more technically correct than the K_{Ia} curve. The K_{Ic} curve models the slow heat-up and cooldown process of a RPV.

   Use of this approach is justified by the initial conservatism of the K_{Ia} curve when the curve was codified in 1974. This initial conservatism was necessary due to limited knowledge of RPV material fracture toughness. Since 1974, additional knowledge has been gained about the fracture toughness of RPV materials and their fracture response to applied loads. The additional knowledge demonstrates the lower bound fracture toughness provided by the K_{Ia} curve is well beyond the margin of safety required to protect against potential RPV failure. The lower bound K_{Ic} fracture toughness provides an adequate margin of safety to protect against potential RPV failure and does not present an undue risk to public health and safety. P-T curves based on the K_{Ic} fracture toughness limits will enhance overall plant safety by opening the P-T operating window, that is, more margin will be available to saturation and reactor coolant pump net positive suction head limits.

3. The requested exemption will not endanger the common defense and security.

   The common defense and security are not endangered by approval of this exemption request.

4. In accordance with 10CFR50.12(a)(2), the NRC will consider granting an exemption to the regulations if special circumstances are present.

   Special circumstances are present which necessitate the request for an exemption to the regulations of 10CFR50.60. This requested exemption meets the special circumstances of the following paragraphs of 10CFR50.12:

   (a)(2)(ii) - demonstrates the underlying purpose of the regulation will continue to be achieved
   (a)(2)(iii) - would result in undue hardship or other costs that are significant if the regulation is enforced and;
   (a)(2)(v) - will provide only temporary relief from the applicable regulation and the licensee has made good faith efforts to comply with the regulations.

Justification for Special Circumstance of 10CFR50.12(a)(2)(ii):

ASME B&PV Code, Section XI, Appendix G, provides procedures for determining allowable loading on the RPV and is approved for that purpose by 10CFR50 Appendix G. Application of these procedures in the determination of P-T operating and test curves satisfy the underlying requirement that:

1) The reactor coolant pressure boundary be operated in a regime having sufficient margin to ensure, when stressed, the RPV boundary behaves in a non-brittle manner and the probability of a rapidly propagating fracture is minimized, and
2) P-T operating and test limit curves provide adequate margin in consideration of uncertainties in determining the effects of irradiation on material properties.

The ASME B&PV Code, Section XI, Appendix G, procedure was conservatively developed based on the level of knowledge existing in 1974 concerning RPV materials and the estimated effects of operation. Since 1974, the level of knowledge about these topics has been greatly expanded. This increased knowledge permits relaxation of the ASME B&PV Code, Section XI, Appendix G, requirements via application of ASME Code Case N-640 while maintaining the underlying purpose of the ASME B&PV Code and the NRC regulations to ensure an acceptable margin of safety.

Justification for Special Circumstance of 10CFR50.12(a)(2)(iii):

The RPV P-T operating window is defined by the P-T operating and test limit curves developed in accordance with the ASME B&PV Code, Section XI, Appendix G procedure. Continued operation of IP2 with these P-T curves without the relief provided by ASME Code Case N-640 would unnecessarily restrict the P-T operating window. This restriction would challenge the operations staff when operating at lower reactor temperatures.

This constitutes an unnecessary burden that can be alleviated by the application of ASME Code Case N-640 in the development of the proposed P-T curves. Implementation of the proposed P-T curves as allowed by ASME Code Case N-640 does not significantly reduce the margin of safety below that established by the original requirement.

Justification for Special Circumstance of 10CFR50.12(a)(2)(v)

The requested exemption provides only temporary relief from the applicable regulation and IP2 has made a good faith effort to comply with the regulation. Con Edison requests that the exemption be granted until such time that the NRC generically approves ASME Code Case N-640 for use by the nuclear industry.

Code Case N-640, Conclusion for Exemption Acceptability

Compliance with the specified requirement of 10CFR50.60(a) and Appendix G would result in hardship and unusual difficulty without a compensating increase in the level of quality and safety. ASME Code Case N-640 allows an increase in the lower bound fracture toughness used in ASME B&PV Code Section XI Appendix G in the determination of reactor coolant system (RCS) P-T limits. This proposed alternative is acceptable because the ASME Code Case maintains the relative margin of safety commensurate with that which existed at the time ASME B&PV Code, Section XI, Appendix G, was approved in 1974. Therefore, application of ASME Code Case N-640 for IP2 will ensure an acceptable margin of safety and does not present an undue risk to the public health and safety.

JUSTIFICATION FOR USE OF CODE CASE N-588

10CFR50.12(a) Requirements

The requested exemption to allow use of ASME Code Case N-588 to determine stress intensity factors for postulated flaws and postulated flaw orientation for circumferential welds meets the criteria of 10CFR50.12 is discussed below. 10CFR50.12 states that the Commission may grant an exemption from requirements contained in 10CFR50 provided that the following are satisfied:

1. The requested exemption is authorized by law.

   No law exists which precludes the activities covered by this exemption request. 10CFR50.60(b) explicitly allows the use of alternatives to 10CFR50 Appendices G and H when an exemption is granted by the Commission under 10CFR50.12.

2. The requested exemption does not represent an undue risk to the public health and safety.

   10CFR50 Appendix G requires that Article G-2120 of ASME B&PV Code, Section XI, Appendix G, be used to determine the maximum postulated defects in RPVs for the P-T limits. These limits are determined for normal operation and test conditions. Article G-2120 specifies in part, that the postulated defect be in the surface of the RPV material and normal (i.e., perpendicular) to the direction of maximum stress. ASME B&PV Code, Section XI, Appendix G, also provides a methodology for determining the stress intensity factors for a maximum postulated defect normal to the maximum stress. The purpose of this article is, in part, to ensure the prevention of non-ductile fractures by providing procedures to identify the most limiting postulated fractures to be considered in the development of P-T limits.

Code Case N-588 provides relief from the Appendix G requirements, in terms of calculating P-T limits, by revising the Article G-2120 reference flaw orientation for circumferential welds in RPVs. The reference flaw is a postulated flaw that accounts for the possibility of a prior existing defect that may have gone undetected during the fabrication process. Thus, the intended application of a reference flaw is to account for defects that could physically exist within the geometry of the weldment. The current ASME Section XI, Appendix G approach mandates the consideration of an axial reference flaw in circumferential welds for purposes of calculating the P-T limits. Postulating the Appendix G reference flaw in a circumferential weld is physically unrealistic and overly conservative, because the length of the flaw is 1.5 times the RPV wall thickness, which is much longer than the width of circumferential welds. The possibility that an axial flaw may extend from a circumferential weld into a plate or axial weld is already adequately covered by the requirement that defects be postulated in plates/forgings and axial welds.

The fabrication of RPVs for nuclear power plant operation involved precise welding procedures and controls designed to optimize the resulting weld microstructure and to provide the required material properties. These controls were also designed to minimize defects that could be introduced into the weld during the fabrication process. Industry experience with the repair of weld indications found during pre-service inspection, in-service non-destructive examinations, and data taken from destructive examination of actual RPV welds, confirms that any remaining defects are small and do not cross transverse to the weld bead. Therefore, any postulated defects introduced during the fabrication process, and not detected during subsequent non-destructive examinations, would only be expected to be oriented in the direction of weld fabrication. For circumferential welds this indicates a postulated defect with a circumferential orientation.

ASME Code Case N-588 addresses this issue by allowing consideration of maximum postulated defects oriented circumferentially in circumferential welds. ASME Code Case N-588 also provides appropriate procedures for determining the stress intensity factors for use in developing RPV P-T limits per ASME B&PV Code, Section XI, Appendix G procedures. The procedures allowed by ASME Code Case N-588 are conservative and provide a margin of safety in the development of RPV P-T operating and pressure test limits that will prevent non-ductile fracture of the RPV.

The proposed P-T limits include restrictions on allowable operating conditions and equipment operability requirements to ensure that operating conditions are consistent with the assumptions of the accident analysis. Specifically, reactor coolant system pressure and temperature must be maintained within the heatup and cooldown rate dependent P-T limits specified in TS Section 3.1.B, "Heatup and Cooldown." Therefore, this requested exemption does not present an undue risk to the public health and safety.

3. The requested exemption will not endanger the common defense and security.

   The common defense and security are not endangered by this exemption request.

4. In accordance with 10CFR50.12(a)(2), the NRC will consider granting an exemption to the regulations if special circumstances are present.

   Special circumstances are present which necessitate the request for an exemption to the regulations of 10CFR50.60. This exemption meets the special circumstances of paragraphs:

   (a)(2)(ii) - demonstrates that the underlying purpose of the regulation will continue to be achieved;

   (a)(2)(iii) - would result in undue hardship or other costs that are significant if the regulation is enforced and;

   (a)(2)(v) - will provide only temporary relief from the applicable regulation and the licensee has made good faith efforts to comply with the regulations.

Justification for Special Circumstance of 10CFR50.12(a)(2)(ii)

The underlying purpose of 10CFR50 Appendix G and ASME B&PV Code, Section XI, Appendix G, is to satisfy the requirement that:

1) The RCS pressure boundary be operated in a regime having sufficient margin to ensure that when stressed the RPV boundary behaves in a ductile manner and the probability of a rapidly propagating fracture is minimized, and
2) P-T operating and test curves provide margin in consideration of uncertainties in determining the effects of irradiation on material properties.

Application of ASME Code Case N-588 when determining P-T operating and test limit curves per ASME B&PV Code, Section XI, Appendix G, provides appropriate procedures for determining limiting maximum postulated defects and considering those defects in the P-T limits. This application of the code case maintains the margin of safety originally contemplated when ASME B&PV Code, Section XI, Appendix G was developed.

Therefore, use of ASME Code Case N-588, as described above, satisfies the underlying purpose of the ASME B&PV Code and the NRC regulations to ensure an acceptable level of safety.

Justification for Special Circumstance of 10CFR50.12(a)(2)(iii)

The RPV P-T operating window is defined by the P-T operating and test curves developed in accordance with the ASME B&PV Code, Section XI, Appendix G procedure. Continued operation with these P-T curves without the relief provided by ASME Code Case N-588 would unnecessarily restrict the P-T operating window for IP2.

This constitutes an unnecessary burden that can be alleviated by the application of ASME Code Case N-588 in the development the proposed P-T curves. Implementation of the proposed P-T curves as allowed by ASME Code Case N-588 does not reduce the margin of safety originally contemplated by either the NRC or ASME.

Justification for Special Circumstance of 10CFR50.12(a)(2)(v):
The requested exemption provides only temporary relief from the applicable regulation and IP2 has made a good faith effort to comply with the regulation. Con Edison requests that the exemption be granted until such time that the NRC generically approves ASME Code Case N-588 for use by the nuclear industry.

ASME Code Case N-588, Conclusion for Exemption Acceptability

Compliance with the specified requirements of 10CFR50.60 would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. ASME Code Case N-588 allows postulation of a circumferential defect in circumferential welds to be considered in lieu of requiring the defect to be oriented across the weld from one plate or forging to the adjoining plate or forging. This circumstance was not considered at the time ASME B&PV Code, Section XI, Appendix G was developed and imposes restrictions on P-T operating limits beyond those originally contemplated.

This proposed alternative is acceptable because the code case maintains the relative margin of safety commensurate with that which existed at the time ASME B&PV Code, Section XI, Appendix G, was approved in 1974. Therefore, application of ASME Code Case N-588 for IP2 will ensure an acceptable margin of safety. The approach is justified by consideration of the overpressurization design basis events and the resulting margin to RPV failure.

Restrictions on allowable operating conditions and equipment operability requirements have been established to ensure that operating conditions are consistent with the assumptions of the accident analysis. Specifically, RCS pressure and temperature must be maintained within the heatup and cooldown rate dependent P-T limits specified in TS Section 3.1.B. Therefore, this exemption request does not present an undue risk to the public health and safety.

Previously Granted Exemptions to Code Cases N-588 and N-640

Exemptions in accordance with 10CFR50.12 from the requirement of 10CFR50.60(a), "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation," to comply with 10CFR50, Appendix G, "Fracture Toughness Requirements" to allow use of ASME Code Cases N-588 and N-640 in developing updated P-T limits have been granted to Duke Energy for the Oconee Nuclear Station, Commonwealth Edison for the Quad Cities Nuclear Power Station, and PECO Nuclear for the Limerick Generating Station, Unit 1. See:

- Letter from D. E. Labarge (NRC) to W. R. McCollum, Jr. (Duke Energy) "Oconee Nuclear Station, Units 1,2, and 3 RE: Exemption From the Requirements of 10 CFR Part 50, Section 50.60(a) (TAC NOS. MA5473, MA5474, and MA5475)" dated July 29,1999
- Letter from D. E. Labarge (NRC) to W. R. McCollum, Jr. (Duke Energy) Amendment No. 307 to Facility Operating Licenses DPR-38, DPR-47, and DPR-55 dated October 1,1999.

- Letter from S. N. Bailey (NRC) to 0. D. Kingsley (Commonwealth Edison), "Quad Cities - Exemption from the Requirements of 10 CFR Part 50, Section 50.60(a) and Appendix C (TAC Nos. MA7140 and MA7141)," dated February 4, 2000.
- Letter from S. N. Bailey (NRC) to 0. D. Kingsley (Commonwealth Edison), "Quad Cities - Issuance of Amendments - Revised Pressure-Temperature Limits (TAC Nos. MA7138 and MA7139)," dated February 4, 2000.
- Letter from B.C. Buckley (NRC) to J.A. Hutton (PECO Nuclear), Limerick Generating Station, Unit 1 – Issuance of Amendment re: Update Pressure-temperature (P-T Limit Curves (TAC No. MA8953), dated September 15, 2000
- Letter from B.C. Buckley (NRC) to J.A. Hutton (PECO Nuclear), Limerick Generating Station, Unit 1 – Exemption from the Requirements of 10CFR50, Section 50.60(a) and Appendix G (TAC No. MA8954) dated September 7, 2000

ATTACHMENT 4 TO NL 01-092

WCAP-15629

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

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247

Westinghouse Non-Proprietary Class 3

WCAP-15629

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

Westinghouse Electric Company LLC

WCAP-15629

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

T. J. Laubham

April 2001

Prepared by the Westinghouse Electric Company LLC
for the Consolidated Edison Company

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:

E. Terek *Ed Terek* Section 1 through 6 and Appendices A, C through G

S.L. Anderson *S.L. Anderson* Appendix B

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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} + ΔRT_{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 2. These CF values are summarized in Table 3. 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	0.19	0.65	34°F
Intermediate Shell Plate B-2002-2	0.17	0.46	21°F
Intermediate Shell Plate B-2002-3	0.25	0.60	21°F
Lower Shell Plate B-2003-1	0.20	0.66	20°F
Lower Shell Plate B-2003-2	0.19	0.60	-20°F
Intermediate & Lower Shell Longitudinal Weld Seams (Heat # W5214)[b]	0.21	1.01	-56°F
Intermediate to Lower Shell Girth Weld (Heat # 34B009)[c]	0.19	1.01	-56°F
Indian Point Unit 2 Surveillance Weld (Heat # W5214)[b]	0.20	0.94	---
Indian Point Unit 3 Surveillance Weld (Heat # W5214)[b]	0.16	1.12	---
H.B. Robinson Unit 2 Surveillance Weld (Heat # W5214)[b]	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 Point 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).

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
	$CF_{B-2002-1} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (160.24) \div (1.405) =$ **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
	$CF_{B-2002-2} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (242.04) \div (2.048) =$ **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
	$CF_{B-2002-2} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (366.44) \div (2.015) =$ **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
	$CF_{Surv. Weld} = \Sigma(FF * RT_{NDT}) + \Sigma(FF^2) = (1598.86°F) \div (6.278) =$ **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	142°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:

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

where,

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

This K_{Ic} curve is based on the lower bound of static critical K_I values measured as a function of temperature on specimens of SA-533 Grade B 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:

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

where,

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

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

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

C = 2.0 for Level A and Level B service limits

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

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

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

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 $2 \leq \sqrt{t} \leq 3.464$,
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 $2 \leq \sqrt{t} \leq 3.464$,
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 $K_{It} = 0.953 \times 10^{-3}$ x CR x $t^{2.5}$, where CR is the cooldown rate in °F/hr., or for a postulated outside surface defect, $K_{It} = 0.753 \times 10^{-3}$ x HU x $t^{2.5}$, 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.

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

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	142	0.861	-20	122.3	34	136
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} + 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.

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	142	0.588	-20	83.5	34	98
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

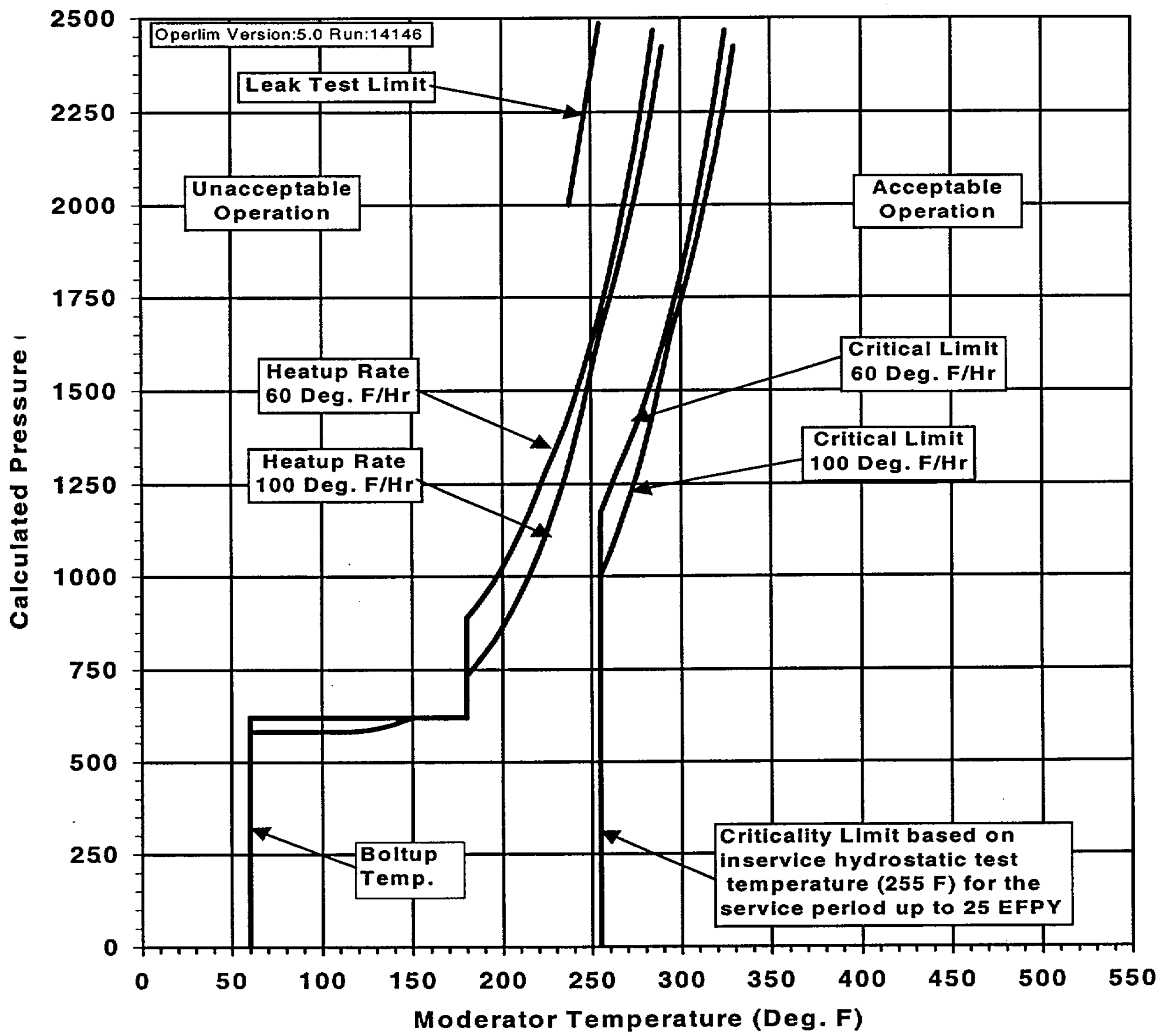

Figure 1 **Indian Point Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 60 & 100°F/hr) Applicable for the First 25 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology**

MATERIAL PROPERTY BASIS

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

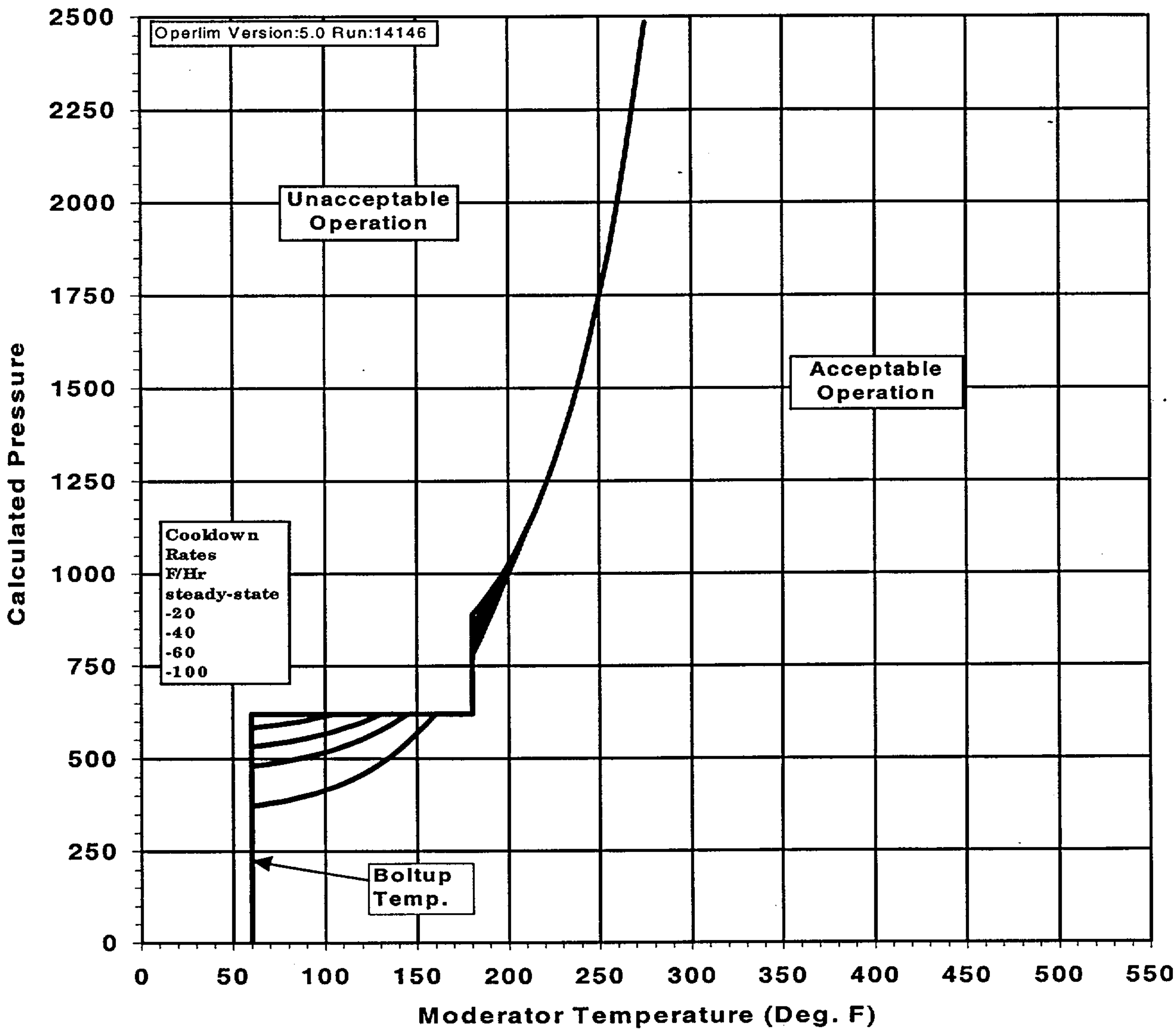

Figure 2 **Indian Point Unit 2 Reactor Coolant System Heatup Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 25 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology**

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."

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

(c) $\Delta RT_{PTS} = CF * FF$

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 $\pm$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 $\pm$ 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 $\pm$ 4.2% (1σ). These comparisons also fall well within the $\pm$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 x 10^{18}	90.29	55.0	21	16
	Z	1.02 x 10^{19}	144.86	125.0	29	21
Intermediate Shell Plate B-2002-2	T	2.53 x 10^{18}	72.17	95.0	19	17
	Z	1.02 x 10^{19}	115.79	120.0	26	23
	V	4.92 x 10^{18}	92.31	77.0	22	4
Intermediate Shell Plate B-2002-3	T	2.53 x 10^{18}	110.35	115.0	25	20
	Y	4.55 x 10^{18}	137.46	145.0	28	28
	Z	1.02 x 10^{18}	177.06	180.0	34	28
Surv. Program Weld Metal	Y	4.55 x 10^{18}	167.37	195.0	28	45
	V	4.92 x 10^{18}	171.87	204.0	29	38
Heat Affected Zone Material	Y	4.55 x 10^{18}	--	165	---	13
	V	4.92 x 10^{18}	--	150	---	0
Correlation Monitor Material	T	2.53 x 10^{18}	--	75	---	0
	Y'	4.55 x 10^{18}	--	70	---	6
	Z	1.02 x 10^{19}	--	102	---	15
	V	4.92 x 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	61
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	32	56

Notes:

(a) These values were obtained from Reference 12. 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.

APPENDIX E

UPDATED SURVEILLANCE CAPSULE REMOVAL SCHEDULE

Withdrawal Schedule To Be Provided in Later Revision

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.

<u>32 EFPY</u>

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

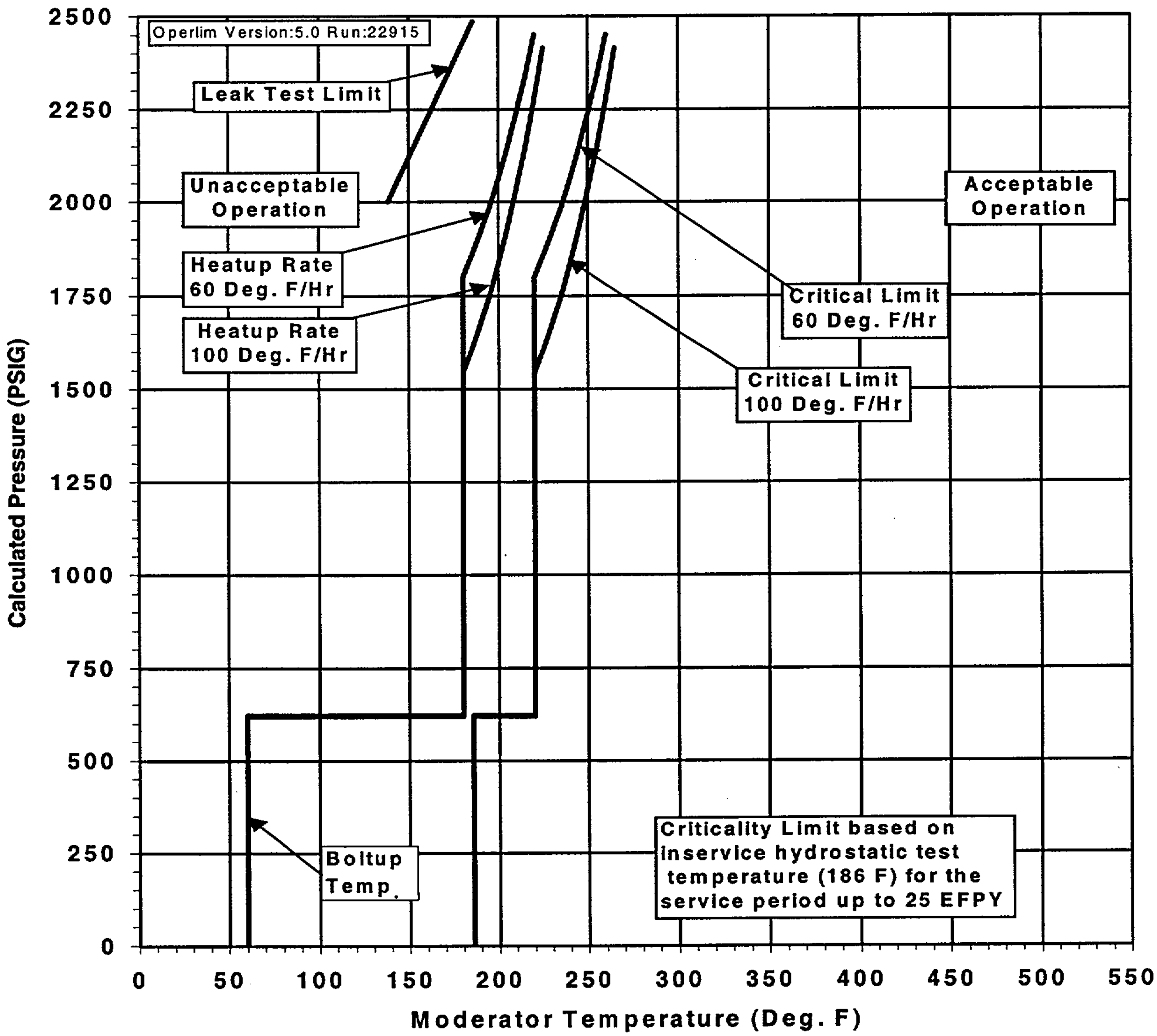

Figure G-1 Indian Point Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 60°F/hr) Applicable for the First 25 EFPY (Without Margins for Instrumentation Errors) Using Code Case N-588

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

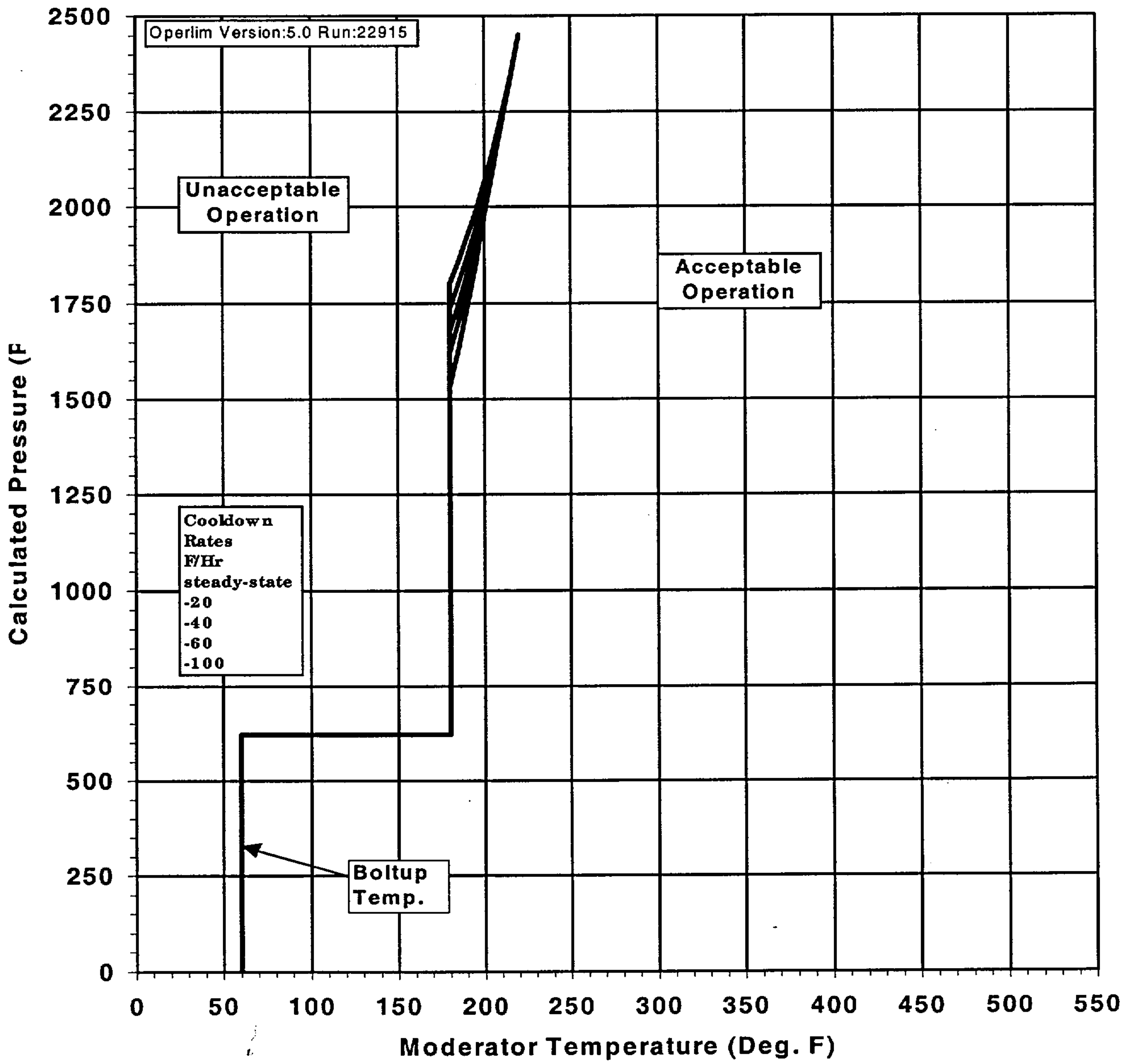

Figure G-2 Indian Point Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable for the First 25 EFPY (Without Margins for Instrumentation Errors) Using Code Case N-588

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								

ATTACHMENT 5 TO NL 01-092

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Indian Point Unit 2 Overpressure Protection System (OPS) Thermal Hydraulic Analysis, Setpoint Development and Technical Specification Revision for 25 EFPY

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
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Indian Point Unit 2 Overpressure Protection System (OPS) Thermal Hydraulic Analysis, Setpoint Development and Technical Specification Revision for 25 EFPY

Prepared by:

Northeast Technology Corp.
108 N Front St
UPO Box 4178
Kingston NY 12402
914-331-8511

Prepared for
Consolidated Edison Company of NY
under
Purchase Order No. 000744-001

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0	May 7 2001	[signature]	[signature]	L. P. Mariani
1	July 3 2001	[signature]	[signature]	L. P. Mariani

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1.0 INTRODUCTION

Based on an increase in the reactor vessel beltline region Reference Nil-Ductility Transition Temperature, or RTNDT (i.e., decreased ductile characteristics due to increased fluence or neutron flux), the latest capsule removal and analysis, and revisions in acceptable analysis methods; the low temperature pressure and temperature limits were developed for 25 EFPY, in Ref.[1].

The current report documents the thermal-hydraulic analysis used to transition from the pressure temperature (PT) limit curve to the Tech Spec limits, as well as the actual plant limits. The former (Tech Spec) limits include the effect of temperature bias due to the vessel thermally lagging behind the RCS liquid. The latter (actual plant limits) include the effects of pressure bias; uncertainties in RCS pressure, temperature, and level; and the uncertainty in SG-to-RCS delta-T, as well as elevation differences between the pressure sensor and pressurizer water level and elevation of the PORVs.

Implementation of these limits will ensure the capability of the OPS to relieve the RCS pressure for the analyzed overpressure transients to prevent these events from causing the peak RCS pressure from exceeding the 10CFR50 Appendix G limits (see Ref.[12]).

The following provides the correspondence between the following report sections, and the IP2 Tech Spec figures and tables.

Figure 3.1.A-1 .... Section 6.0 and 7.0
Figure 3.1.A-2,3 and 4 .... Section 9.0
Figure 3.1.A-5 and 6 .... Section 10.0
Table 3.1.A-2 (pg 1 of 2) .... Section 7.0 and 10.0
Table 3.1.A-2 (pg 2 of 2) .... Section 6.0 and 7.0

The reanalysis/evaluation includes the following OPS <u>operable</u> cases:

- 3 charging pumps, or 1 SI and 2 charging pumps*
- Start of 1 RCP with SGs 40°F hotter than the RCS
- Minimum RCS pressure after actuation of 2 PORVs (RCP operational consideration)

* The PORV activation setpoint is plotted for 3 charging pumps; however as an option the SI and charging pumps case is also given (on Figure 3.1.A-1).

The reanalysis/evaluation includes the following OPS <u>inoperable</u> cases:

- 1 charging pump
- 2 charging pumps
- 3 charging pumps
- Start of 1 RCP with SGs 40°F hotter than the RCS, and start of RCP with SGs 100°F hotter than the RCS
- Fixed RCS vent size for up to 3 SI and 3 charging pumps.

These cases and analysis assumptions are discussed in the following applicable sections.

2.0 THEORY

The analysis for the OPS include: RCS pressure overshoot for liquid relief through the PORV, RCS pressure overshoot for gas relief through the PORV, RCS pressure undershoot for both liquid and gas relief through both PORVs, and compression of a pressurizer gas bubble for OPS inoperable. The analysis methods were used in the prior Ref.[2] analysis.

The analysis methods for the undershoot are similar to the overshoot analysis, except the minimum pressure corresponds to the pressure after the PORVs go through an entire closure cycle--with no incoming flow.

2.1 OPS Operable Mass Addition Analysis

For OPS operable the mass addition analysis assumes the RCS is water solid, and calculates the transient pressure rise associated with mass (i.e., liquid volume) addition due to charging and/or SI pumps, liquid relief through the PORV as the flow area increases with time, and the RCS fluid compressibility.

For such a system the total volume is fixed, as is the temperature; however, the pressure and mass are variable, resulting in:

$$\frac{dV}{dt} = \frac{\partial V}{\partial M}\frac{dM}{dt} + \frac{\partial V}{\partial P}\frac{dP}{dt} = 0,$$

From the definitions of specific volume ($v = V/M = \partial V/\partial M$) and coefficient of compressibility ($\beta = -1/V\ \partial V/\partial P$), conservation of mass ($dM/dt = \dot{M}_{IN} - \dot{M}_{OUT}$), and assuming the specific volume of the incoming and outgoing fluid are equal, we obtain,

$$\frac{dP}{dt} = \frac{(Q_{IN} - Q_{OUT})}{\beta V},$$

where Q_{IN} and Q_{OUT} are the volumetric (rate of) inflow due to charging and/or SI pumps, and the PORV volumetric outflow, respectively.

It is assumed the initial pressure is at the PORV open setpoint (P_0), and after a PORV delay time (t_1), the PORV begins to open, with an area that varies linearly with time, from closed to open over time (t_2). The maximum pressure rise (overshoot) occurs at time (t^*) when the inflow and outflow are equal (as the valve is opening). The result is,

$$t^* = t_2\frac{Q_{IN}}{Q_{OUT}} + t_1,$$

$$P_{MAX} - P_0 = \frac{Q_{IN} t^* - \dfrac{Q_{OUT}(t^* - t_1)^2}{2t_2}}{\beta V}.$$

The volumetric flow in and out are described below. To be a valid solution, t* must occur after the valve has started to open, and prior to its reaching full open.

$t_1 < t^* < t_1 + t_2$.

The volume inflow is either a constant (charging pump flow) or decreases with increasing pressure (SI flow), thus it is conservative to evaluate the inflow at the setpoint pressure P_0.

The volume outflow (PORV relief capacity) is given by the PORV C_v value, the downstream (back) pressure (P_B), the fluid specific volume relative to water specific volume at ambient pressure and 60°F (v/v_{60}), and the stagnation pressure at the PORV--which is conservatively taken as PORV setpoint pressure.

$$Q_{OUT} = C_v \sqrt{\left(\frac{v}{v_{60}}\right)(P_0 - P_B)}$$

For 3 charging pumps (or less), the resulting PORV flow rate (to match the incoming flow) is less than the saturated liquid water critical flow rate (HEM model); however, at larger flow rates (i.e., 1 SI or more flow) it is necessary to credit some subcooling at the PORVs.

As a bounding calculation the PORV relief capacity is 981 gpm (at the lowest PORV actuation pressure, adjusted to the PORV elevation, and including 100 psig back pressure). Although this is considerably larger flow than required (the mass addition for 1 SI and 2 charging pumps results in 720 gpm) this flow was assumed, and the required subcooling, to not be chocked, was calculated. This resulted in a maximum required subcooling of 66°F (results range from 27 to 66°F over the pressure range of interest).

Assuming the maximum pressurizer water volume (85% level, per Appendix 1), it is possible to calculate the pressurizer subcooling once the pressurizer has filled with water--assuming complete mixing in the pressurizer. The results vary from 142 to 121°F over the pressure range of interest. The 121°F case corresponds to the same condition (pressure and maximum temperature) where the above 66°F required subcooling was calculated.

Based on these analyses it is concluded the subcooling will be sufficient to prevent chocked flow at the PORVs. Although the geometry in the pressurizer does not favor rapid (complete) mixing, even moderate mixing (about 50%) will be sufficient, to ensure 66°F subcooling at the inlet to the PORVs.

The analysis described above (as well as the subsequent analysis) utilize fluid properties: coefficient of expansion (α) and compressibility (β), not generally available over the large range of temperatures of interest, and Ref.[13] the ASME steam tables (principally) was used to develop the required values of α and β as functions of temperature.

2.2 OPS Operable Heat Addition Analysis

The heat addition transient is due to the SG-to-RCS heat transfer caused by a RCP start with SGs hotter than the RCS. For this event the restrictions on the pressurizer level (prior to starting the first RCP) ensure the RCS will not become water solid. The resulting pressurizer gas bubble provides additional volumetric relief capacity, associated with gas flow through the PORV, and the addition of the gas bubble compressibility.

For such a system the change in the total liquid volume (V_f) is a function of pressure and temperature, the coefficient of thermal expansion ($\alpha = 1/V\ \partial V/\partial T$), and the coefficient of compression (β),

$$\frac{dV_f}{dt} = \frac{\partial V_f}{\partial T}\frac{dT}{dt} + \frac{\partial V_f}{\partial P}\frac{dP}{dt}$$
$$= \alpha V_f \frac{dT}{dt} - \beta V_f \frac{dP}{dt},$$

and for fixed RCS volume $dV_g/dt = -\,dV_f/dt$ where V_g is the total gas volume.

Assuming isentropic compression of an ideal gas, the (usual) result is obtained:

$$PV_g^{k} = \text{constant, where } k = c_p/c_v.$$

Differentiating PV_g^k with respect to time, and using the ideal gas law, $PV_g=nRT_g$, and, using the energy equation for the liquid,

$$q = M_f c_f \frac{dT_f}{dt} = V_f \frac{c_f}{v_f}\frac{dT_f}{dt},$$

where, the variables are the energy per time transferred into the RCS (q) and the liquid specific volume and heat capacity (v_f and c_f). Combining these equations, the result is,

$$\frac{dP}{dT} = \frac{\alpha q \dfrac{v_f}{c_f} + \dfrac{V_g}{n}\dfrac{dn}{dt}}{\dfrac{V_g}{Pk} + \beta V_f}.$$

or,

$$\frac{dP}{dt} = \left(Q^*_{IN} - Q^*_{OUT}\frac{(t^* - t_1)}{t_2}\right)\left(\frac{V_g}{Pk} + \beta V_f\right)^{-1}.$$

The first term in the numerator represents the volumetric expansion of the liquid and the second is the gas volumetric flow out the PORV. Assuming an ideal gas,

$$Q^*_{OUT} = \frac{V_g}{n}\frac{dn}{dt} = Area\sqrt{kRT_g}$$

The gas flow rate is obtained from the PORV flow area, and the flowing gas (nitrogen or steam). For nitrogen the PORV value (assuming 100°F, or 560°R)† is 6.3 ft^3/sec, while for steam (HEM model) the value is 8.2 ft^3/sec, over pressures (and thus temperatures) of interest. Thus the results for nitrogen are bounding--lower relief capacity and larger value of k (1.4 versus about 1.33 for steam).

The SG-to-RCS heat transfer (q) is a function of the heat transfer coefficient and temperature difference, SG heat transfer area, and RCS flow rate. Conservatively assuming: a free convection heat transfer coefficient at maximum SG temperature of 350°F in the SG (shell-side) and SG-to-RCS delta-T of 40°F, and infinite heat transfer coefficient in the RCS (tube-side), no SGTP (maximum heat transfer area), and active loop and 3 idle loops flows of 1.116 and 0.117 (fraction of loop thermal design flow--based on pre-operation tests) respectively, the heat transfer (q) is calculated to be 0.2805 million Btu/sec.

The solution for the peak pressure is when the inflow just balances the outflow (as the valve is opening). As before assuming the pressure is at the PORV open setpoint (P_0), in order to integrate, the result is,

$t^* = t_2 Q^*_{IN}/Q^*_{OUT} + t_1$,

$$P_{MAX} - P_0 = \frac{Q^*_{IN}\,t^* - \dfrac{Q^*_{OUT}\,(t^* - t_1)^2}{2t_2}}{\dfrac{V_g}{Pk} + \beta V_f},$$

and, as before, to be a valid solution,

$t_1 < t^* < t_1 + t_2$.

† Although the RCS temperature may be lower than 100°F--i.e., 60°F, the analysis assumes a conservatively large RCS temperature of 350°F. This maximizes the resulting thermal expansion (about six-fold increase over 60°F) which is conservative.

2.3 OPS Inoperable Mass and Heat Addition Analysis

For OPS inoperable, the analysis assumes isentropic compression of the pressurizer gas bubble. The gas bubble is reduced due to 10 minutes of charging water addition (ΔV_g), or due to RCS water expansion on heat-up due to RCP start with hotter SGs.

For the latter case, the total RCS water temperature rise is obtained from an energy balance, mixing the primary and secondary water to obtain the equilibrium temperature, and then calculating the fluid expansion ($\Delta V_f = \alpha\ V\ dT = -\Delta V_g$).

The change in gas pressure is given by,

$$P_g(0) = P_g(f)[(V_g(0)+\Delta V_g)/V_g(0)]^{\gamma},$$

where the final pressure P(f) is the Appendix G limit--after correction for elevation between the pressurizer gas and the cold leg pressure sensor location, and the initial pressure P(0) is the maximum allowable RCS pressure--also corrected for elevation differences.

3.0 REVISED ARMING TEMPERATURE AND PT LIMIT CURVE

The Ref.[1] provides the revised arming temperature of 280°F, reduced from the previous value of 305°F. In addition the steady state (0°F/hr) curve, from Table 13 in Ref.[1], is used to develop the revised limits on RCS pressure, temperature, and level. These are applicable for operation to 25 EFPY.

As described in Ref.[1] the PT curves were generated based on the latest available reactor vessel information and updated fluences. The new Indian Point Unit 2 heatup and cooldown pressure-temperature limit curves were generated using ASME Code Case N-640, Ref.[10], (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 Unit 2 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 are developed using the axial flaw methodology with the most limiting axial flaw adjusted reference temperatures (ARTs). These PT curves bound the PT curves that used the ASME Code Case N-588, Ref.[11] (Circ. Flaw Methodology), with the most limiting Circ Flaw ARTs.

4.0 OVERPRESSURE PROTECTION SYSTEM CHARACTERISTICS

The OPS is comprised of the two power operated relief valves (PORVs) and an actuation (setpoint) curve--pressure as a function of RCS temperature. This setpoint curve is established to ensure the RCS pressure does not exceed the (low temperature OPS) PT limit curve for the following events:

- mass addition due to the start of three (3) charging pumps with loss of letdown, or

- heat addition due to the start of a reactor coolant pump (RCP) with steam generators (SGs) 40°F hotter than the RCS.

- Additionally, as an option, the mass addition due to the start of two (2) charging pumps and one (1) SI pump is also analyzed.

In the event of a mass or heat addition to the RCS, the pressure will increase. Depending on the event, plant conditions, etc., the OPS pressure setpoint may be reached and OPS actuated.

The OPS system utilizes cold leg wide range temperature and hot leg pressure in each of 3 loops. Pressure and temperature in the same loop are compared to see if the OPS setpoint is exceeded, and if this occurs in 2 (out of the 3) loops OPS is actuated. In the event of a single failure this becomes a 1 (out of 2) loops logic, and it is this 1 out of 2 that is used herein.

Due to the finite PORVs delay time and opening time there is a small additional pressure increase (overshoot) above the PORV open setpoint occurs--until the PORV(s) have opened sufficiently to mitigate the event.

Once the PORV(s) are sufficiently open, to mitigate the pressure increase, further valve opening will cause the pressure to fall, and if it falls below the PORV close setpoint the PORV(s) will immediately begin to close. Once the PORV(s) are sufficiently closed, to mitigate the pressure decrease, further valve closing will cause the pressure to rise again, and the cycle will be repeated. The nitrogen supply is sized to provide 10 minutes operation of both PORVs - should instrument air be unavailable.

Rapid valve opening and closing can occur if the PORV close setpoint is not set sufficiently below the open value. The IP2 OPS close setpoint is at least 1/2% of span below the open setpoint (i.e., at least 7.5 psi below the open setpoint); and it is possible to adjust this to a value greater than 1/2%.

Note, that the RCPs may be damaged by a spurious or real OPS event, as significant pressure drop in the RCS can occur. Analysis (for the undershoot) demonstrates this is not a problem, **so long as the PORV close setpoint is not reduced (significantly) below the current 7.5 psi value.**

The design value for the delay and open (and close) times were 0.3 and 1.2 seconds respectively, and the overshoot value obtained from the Westinghouse study [9] is 39 psi for one SI pump operating without letdown.

IP2 specific analyses demonstrate additional margins and, as part of the present study, analysis was performed with increased PORV stroke (opening or closing) times. This is done to address possible uncertainties in test measurement of these values (to allow raising the value for the stroke time in a test), etc.

In addition (for the heat addition event) since the OPS pressure setpoint varies with RCS temperature, and since the reactor vessel would lag behind the RCS temperature, it is necessary to incorporate a temperature lag or shift in developing the OPS setpoint curve. This temperature shift provides for the RCP start with SGs 40°F hotter than the RCS, which could establish (over a very short time) a reasonably large temperature difference between the RCS water and the vessel metal.

Unlike the mass addition case the heat addition transient is self terminating and is not an inadvertent action--thus it is feasible to ensure a pressurizer bubble is present. This is done by limiting the allowable SG-to-RCS delta-T on RCP start (i.e., limiting RCS fluid thermal expansion) and specifying a maximum pressurizer liquid level. The resultant benefit is significant as compared to a water solid system.

The OPS system has uncertainties associated with the pressure and temperature measurements, and in prior analyses the OPS setpoint included temperature and pressure uncertainties of 20 psi and 3°F. It was shown previously, that the margin between the OPS actuation curve, and the PT limit curve was adequate to include these uncertainties and the RCS pressure bias.

However, in the present analysis a detailed review and reanalysis of the uncertainties was performed, and at the higher RCS temperatures, the pressure uncertainties are an order of magnitude larger. These uncertainties are documented herein.

5.0 OPS PRESSURE, TEMPERATURE, AND LEVEL UNCERTAINTIES AND BIAS

There are several uncertainties and biases that must be included in the OPS analysis. These include the (OPS) arming temperature uncertainty, hot leg pressure bias and uncertainty, the wide range cold leg temperature bias and uncertainty, pressurizer level uncertainty, and the steam generator (SG) pressure uncertainty.

In the past the OPS Tech Spec has explicitly included the temperature bias, and it is included in the revised Tech Spec curves/values. This bias is due to the start of an RCP with SGs hotter than the RCS: it is a direct result of the analysis assumptions, and it is calculated in the analysis section of this report. In addition there is a pressure bias between the PORV pressure (and thus relieving capacity) and the (sensor) RCS pressure, and this is included in the Tech Spec figures and tables.

The remaining uncertainties and bias are not included in the Tech Spec--instead they are applied to the Tech Spec curves/values when these are implemented in plant procedures, etc. This is the method used in prior IP2 OPS Tech Spec limits, and it has the advantage of permitting revisions in OPS limits (e.g., due to increased uncertainty for a particular device) without requiring changes to the Tech Spec.

For the OPS operable the single failure assumption results in only one channel being used; however, for OPS inoperable, the operator may utilize more than one loop and/or channel indications. In this case those independent contributors to the uncertainties can be statistically combined. For example, in the limit of 12 totally independent indications, as compared to one indication, this would result in 71% reduction in the uncertainty. This is of particular importance in the SG-to-RCS delta-T uncertainty, which would otherwise essentially preclude starting an RCP--if the measured SG-to-RCS delta-T and the control room SG outlet steam pressure indications are used.

As part of the OPS analysis these uncertainties and biases were developed, and the changes to the Tech Spec values obtained. These are summarized in the Appendix 2. for OPS not operable, and Appendix 3 for OPS operable. For the pressure, temperature, and level uncertainties, generally Ref.[5] 30 month cycle uncertainties are used. In cases where an uncertainty was unavailable (notably the arming temperature) a conservative value was developed/used; however, the uncertainties are explicitly stated, in order that future changes to uncertainties can be accounted for.

Appendix 1 provides the Tech Spec revisions and Appendix 2 identifies what the Tech Spec value will become after including the uncertainties and bias--i.e., it provides the transition document from the Tech Spec limits to the limits in procedures, etc.

5.1 OPS Operable and Not Operable: RCS Pressure Bias

The pressure difference between the vessel beltline region (lower downcomer at elevation of bottom active fuel) and the wide range pressure (sensors in hot legs and transmitters about 11 feet below same) was calculated for OPS conditions: all combinations (0 to 4) of RCPs operating, including elevation effects, and low RCS temperature.

The results, from Ref.[2], for the pressure bias (indicated value minus actual value at the lower beltline region) are: -3.2, -8.0, -13.0, -21.4, and -30.3 psi; for 0, 1, 2, 3, and 4 operating RCPS, respectively. These values are conservative for the vessel flange region which is at a higher elevation. In addition these results are applicable to OPS operable and OPS inoperable.

In addition there is 72.75 feet elevation difference between the PORVs and the RCS (transmitter) location; and the reduced pressure at the PORVs was included in the OPS operable analyses.

5.2 OPS Operable: Arming Temperature

For the OPS arming temperature a conservative uncertainty value of ±20°F is assumed. (Ref.[3], which is being updated by Con Edison, had a value of about ±15°F).

5.3 OPS Operable: Combined Pressure and Temperature Uncertainty

For OPS operable, the single channel (protection channel) uncertainties, from Ref.[4] were used. These include RCS hot leg wide range pressure: ± 22.35 psi, and RCS cold leg wide range temperature uncertainty: ± 7.26°F and the bistable error: ± 13.19 psi. These are combined, to form a single overall pressure uncertainty - see Appendix 3.

The slope of the OPS RCS pressure limit curve, $(dP/dT)_{OPS}$, is required--and the resulting uncertainty ranges from about 30 psi at low temperatures, to 260 psi at high temperatures. (**It is highly nonlinear, and cannot be calculated with a single slope value**).

5.4 OPS Operable and Not Operable Pressurizer Level

A revised RCS pressurizer level uncertainty of 5.7% (rounded to 6.0%) is used. This uncertainty takes credit for two (2) channels. For the present application this uncertainty is used; however, **it is likely the OPS temperature range may require development of a level bias versus fluid temperature, or a larger uncertainty.**

5.5 OPS Not Operable: RCS Pressure

The RCS wide range pressure uncertainty is based on a single channel value. The pressure uncertainty is ± 45.9 psi.

5.6 OPS Not Operable: RCS Temperature

The RCS temperature uncertainty is based on a single channel value. The temperature uncertainty is ± 13.5°F.

5.7 OPS Operable and Not Operable: SG-TO-RCS Delta-T Uncertainty

In the case of the heat addition it is also necessary to develop the uncertainty in the SG-to-RCS delta-T (ΔT_{SG-RCS}). This involves using the uncertainty in the SG outlet steam pressure and the slope of the saturation pressure versus temperature $(dP/dT)_{SAT}$ to obtain the total SG temperature uncertainty. The total uncertainty in the ΔT_{SG-RCS} is obtained by statistically combining the SG pressure uncertainty, the SG temperature calculation uncertainty (±2°F for the analysis herein), and the RCS temperature uncertainty (from above).*

$$\Delta T_{SG-RCS}\ UNC = [(SGPUNC/(dP/dT)_{SAT})^2 + 2^2 + 12.6^2]^{1/2}.$$

The SG pressure (SG outlet steam pressure indication) uncertainty depends on the type and number of indications used. Two cases are considered. In each case credit is taken for all four loop SG pressures canvassed, and the highest loop pressure chosen (i.e., statistical combination of loop independent uncertainties). The (SGPUNC) uncertainties are:

local pressure gauges (**using one channel per loop**): ± 7.5 psi

CR pressure indication (**for 3 channels per loop**): ±50.1 psi

Note- for CR pressure, this requires that 12 pressure indications be canvassed. Three channels per loop are to be averaged, and the average for the hottest SG used (alternatively, the highest channel in each SG could be used). The results are provided in the figure in Appendix 2.

5.8 Additional OPS Operable and Not Operable Margins

For the OPS operable and inoperable analysis, the curves (Tech Spec and w/Unc) include a 5°F and 10 psi margin – i.e. in effect the overshoot was increased by 10 psi, and the temperature bias was increased by 5°F. These margins are available to offset minor potential changes in uncertainty, etc.

* The RCS temperature uncertainty was increased from 12.6 (for 3 channels) to 13.5°F; however, the change is not made to the SG-to-RCS delta-T uncertainty. The effect is insignificant, and for this uncertainty the RCS temperatures in each loop are used to develop the loop SG-to-RCS delta-T.

6.0 OPS OPERABLE - OVERSHOOT ANALYSIS FOR MASS ADDITION

For the present analysis the 3 charging pumps case used for the mass addition event, and as an option the case of 1 SI pump and 2 charging pumps was also analyzed. This is consistent with Con Edison plans to exclude SI pumps from being energized in the OPS regime; however, to provide for potential future changes to this, the 1 SI and 2 charging pumps case was also analyzed.

The IP2 OPS was previously analyzed for mass and heat addition events. Previously the mass addition event was the inadvertent start of a single high head SI pump, or the operation of 3 charging pumps without letdown (although the Tech spec figure identified it as 1 SI and/or 3 charging pumps).

Initially, the addition of the fluid to the RCS results in a compression of the pressurizer (nitrogen or steam) bubble; however, with continued charging and/or SI flow to the RCS the pressurizer may become filled with water, and the resulting overshoot will be the largest. Thus the RCS is assumed to be water solid--i.e., this is consistent with 10 minute operator action time assumed for termination of the charging and/or SI flow.

The OPS actuation pressure (setpoint curve) is established so that the resulting pressure (actuation pressure plus overshoot) is below the PT limit curve. The analysis assumptions maximize the resulting (pressure) overshoot.

The RCS pressure is adjusted to account for elevation differences between sensor location, vessel beltline, and the elevation of sensor and pressurizer PORVs. In addition the pressure and temperature uncertainties are included. These are discussed in prior section and it is appropriate to point out that (except for the PORV-to-sensor elevation difference) these are not included in the Tech Spec limits (curves) for the PORV actuation pressure (in Appendix 1). They are included in the actual OPS PORV actuation pressures. This allows the actual PORV actuation curve to be revised, if future bias and uncertainty values should change significantly, without requiring a revision to the Tech Spec.

The mass addition analysis (for pressure overshoot) previously performed, Ref.[2], were reviewed. These analyses assumed: (1) pressurizer water solid, (2) SI flow is a function of RCS pressure, (3) one PORV fails to open, (4) after a delay time the PORV flow area varies linearly over the valve stroke time, (5) PORV capacity (C_V) of 50 GPM/(psi)$^{1/2}$, (6) RCS temperature constant at 100°F, (7) initial RCS pressure of 450 psig, (8) PRT (pressurizer rupture tank) pressure of 100 psig, and (9) 0% steam generator tube plugging (SGTP).

Based on the review it was concluded that, in order to consider the possible effect of tube plugging, the SGTP should be increased to 25%. This reduces the RCS water volume by 6%, and results in a minor increase (of 2.36 psi) in the overshoot. However, as discussed earlier the mass addition event is revised to 3 charging pumps, and as an option 1 SI and 2 charging pumps. For the former case the overshoot is reduced; and for the latter case it is increased.

In addition for the present study, the allowable RCS pressure has increased to 621 psi, at low temperatures. Allowing for pressure overshoot, and pressure bias and uncertainty, the new minimum OPS actuation pressure for 2 charging pumps and 1 SI pump is calculated to be 500 psig. Thus, the use of 450 psig, in the prior analysis, is conservative.

For the current analysis the SI flows were recalculated, based on the current IP2 SI flow balancing methods, including additional conservatism, and the results are similar to those previously used. Specifically, over the RCS pressure range of interest (450 psig and above): the new SI flows are as much as 70 gpm larger (at high RCS pressures); are the same at about 600 psig (512 gpm SI flow to the RCS); and at 450 psig, the new SI flows are about 6 gpm lower (a difference of 1%). These are provided in Appendix 3.

For the charging pumps a flow rate of 100 gpm per pump is conservatively assumed -- this is greater than the design valve of 98 gpm and ignores the RCP seal leakage.

Finally the RCS temperature and the associated fluid properties (density and coefficient of compressibility), were reviewed. It was determined these effects would not increase the overshoot.

The effect of PORV delay time and open time was considered in the prior analysis-- specifically it was (conservatively) increased from the original OPS design values: a delay time of 0.3 seconds, and valve stroke time of 1.2 seconds, which resulted in a calculated overshoot of 21.7 psi. The revised analysis (see following table) includes variation in delay time (t_1), open time (t_2) and valve C_v.

Table of Mass Addition Overshoot Results

C_v	t_1 (sec)	t_2 (sec)	Overshoot (psi) for:		
$\frac{gpm}{psi^{1/2}}$	delay time	open time	3 charging pumps (300 gpm)	1SI pump (520 gpm)	1 SI & 2 charging pumps (720 gpm)
50	0.3	1.2	-	20.1	33.3
50	0.4	1.5	11.9	26.0	42.8
50	0.4	1.9	-	29.4	49.3
50	0.4	2.4	-	33.7	-
40	0.4	1.5	14.4	29.2	48.9

The 3 charging pump case conservatively assumes 0.4 second delay and 1.5 second (linear) opening; for this case the overshoot is calculated to be 11.9 psi, and a value of

20 psi was conservatively used. For the 2 charging and 1 SI pump case (also for 0.4 and 1.5 seconds) the overshoot was calculated to be 42.8 psi, and a value of 50 psi was conservatively used.

Finally the mass addition PORV actuation curve (and subsequently developed heat addition PORV actuation curve) are compared and the minimum value (of the two), at the same temperature, is used.

7.0 OPS OPERABLE - OVERSHOOT ANALYSIS FOR HEAT ADDITION

The IP2 OPS was previously analyzed for mass and heat addition events; the heat addition event is the start of one RCP with SGs hotter than the RCS. As a result of the cooler RCS water passing through the SG tubes, the RCS will be heated. This heating of the RCS leads to a liquid volume increase, and (thus) compression of pressurizer bubble. Unlike the mass addition case, the RCS liquid volume increase is limited by the total RCS temperature increase, due to the RCS and SGs reaching an equilibrium temperature. (There is also the minor long term heat input from the RCP, and core decay heat; however, these are not significant, for the overshoot analysis.)

The heat transfer rate, from the secondary-side to the primary-side, will depend on the RCS flow rates. In the original IP2 OPS analyses the RCS flow was varied linearly over 10 seconds, from zero to the final value; however, in the Ref.[2] it was conservatively assumed the RCP flow reaches it's maximum value instantly, and the remaining 3 (idle) loops also instantaneously reach their respective maximum flows (in the reverse direction). This is a significant conservatism, and addresses the minor addition of RCP heat input and/or decay heat.

The resulting heat transfer to the RCS is calculated based on free convection in the SGs, neglecting the resistance of the SG tube wall, and assuming a large (infinite) primary side convective heat transfer coefficient. In addition, 0% SGTP is assumed, the RCS flow rate is 111.6% of the original IP2 TDF (thermal design flow) in the active loop, and is 11.7% in each of the 3 idle loops. These values were measured during pre-operational tests conducted at Indian Point Unit 2.

The heat addition analysis with OPS operable was performed for 40°F hotter SGs. Although the starting of an RCP with SGs 100°F hotter than the RCS is permitted, the establishment of a sufficiently low pressurizer level and pressure is used to mitigate the resulting heat addition--i.e., no credit is taken for the OPS. (The analysis is described in the OPS not operable section of this report). Unlike the mass addition analysis, it is necessary to apply a temperature bias to account for the increase in the RCS temperature, while the vessel metal is assumed to be unchanged from the value when the RCP was started.

Assuming a maximum SG water level of 80% narrow range level, and a minimum RCS water level of 0%, and 25% SGTP; the RCS heat up was calculated to be 59°F for SGs 100°F hotter than the RCS, or 23.6°F for 40°F hotter SGs (however, as discussed below, a 30.8°F shift is applied for OPS operable heat addition).

Although not part of the pressure overshoot calculation, it is noted that the limitations on starting a RCP (with SGs hotter than the RCS) preclude exceeding 100% pressurizer level. For the above 23.6°F heat up, the resulting pressurizer level increase (at the highest temperature at which a RCP start is permitted, with SGs 40°F hotter than the

RCS) is 8.2%.

In addition to the temperature bias, during the heat addition event, the SG tubes initially contain hotter water, and this water will (assuming single failure of one OPS channel) result in a temperature bias. This bias is present only in one of four loops (the active loop), and is present for about 7 seconds. It is calculated to be 30.8°F--which is larger than the above 23.6°F, and thus for the heat addition event, the OPS actuation pressures versus temperature, are shifted by 30.8°F.

The heat addition analysis (for pressure overshoot) previously performed, Ref.[2], were reviewed. These analyses assumed: (1) pressurizer water level of 85% (318 ft^3 gas), (2) the above assumptions for the SG-to-RCS heat transfer, which include 0% SGTP, (3) one PORV fails to open, (4) after a delay time the PORV flow area varies linearly over the valve stroke time, (5) PORV flow area of 1.398 in^2, (6) pressurizer filled with nitrogen (c_p/c_v=1.4) at 100°F, (7) RCS temperature initially at 350°F, (8) RCS pressure of 1200 psig, (9) pressurizer rupture tank (PRT) pressure of 100 psig, and (10) 0% SGTP.

The PORV flow area of 1.398 in^2 was calculated from the HEM critical flow (Ref.[6]), and the PORV capacity of 179,000 lb/hr saturated steam at 2350 psia (Ref.[7]). The compression of the pressurizer bubble was based on isentropic compression of an ideal gas with gamma (c_p/c_v) of 1.4. This is appropriate for nitrogen and conservative for steam.

The assumption of the pressurizer gas at 100°F, and RCS at 350°F, are conservative assumptions. The use of a low gas temperature results in a low sonic velocity (and thus relief rate) at the PORV. Although it is conceivable the pressurizer gas temperature could be lower (as low as 60°F), the use of 350°F is sufficiently conservative to offset this.

The new arming temperature is 280°F, and thus to avoid RCS temperatures going above this value, the maximum allowable RCS temperature (when starting the first RCP with SGs 40°F hotter than the RCS) is about 250°F, as discussed below. The coefficient of volumetric expansion is about 25% lower at 250°F, while the sonic velocity is 7% lower at 100°F--thus the net effect is a reduction in overshoot.

In addition, for the present study the allowable RCS pressure (at 250-23.6°F) is about 1300 psi. Including pressure, overshoot, bias, and uncertainty would result in a value below 1200 psig. Thus the use of 1200 psig in the analysis is conservative.

Finally, as regards the fluid volume, the heat transfer and fluid volume expansion were calculated for 0% SGTP. If 25% SGTP was assumed for each, the heat transfer would be 25% lower, while the effect of total water volume, in the overshoot analysis, is trivial (increases it about 0.2 psi). Also, the RCS water volume was assumed to be at 85% level, and decreasing the level would lead to larger gas bubble (which would reduce the overshoot significantly)--thus use of 85% (the Tech Spec maximum) is conservative.

The effect of PORV delay time and open time was considered in the prior analysis--specifically it was (conservatively) increased from the original OPS design values: a delay time of 0.3 seconds, and valve stroke time of 1.2 seconds, which resulted in a calculated overshoot of 7.6 psi. The revised analysis assumed a delay time and stroke time of 0.4 seconds and (a bounding) 2.4 seconds respectively; which resulted in a overshoot of 12.7 psi.

In order to avoid disarming the OPS prematurely, the RCP start with hotter SGs must be avoided for a range of temperature leading up to the arming temperature. For the Tech Spec (Table 3.1.A-2) for SGs 40°F hotter than the RCS, the RCP start is to be precluded above (T=280-30.8°F) 249°F.

For the plant procedures (i.e., Table 3.1.A-2 w/Unc) for SGs 40°F hotter than the RCS, the arming temperature plus uncertainty is 300°F, and the RCS temperature uncertainty is 13.5°F--thus the RCP start is to be precluded above (T=300-30.8-13.5°F) 256°F.

As the arming temperature uncertainty is larger than the RCS temperature uncertainty - the minimum temperature range where RCS start is precluded actually increases in transitioning from the Tech Spec to the plant procedures - i.e. this is because the arming temperature is increased when uncertainties are included.

In developing the heat addition PORV actuation curve the overshoot was assumed to be 20 psi, which is conservative compared to the analysis value of 12.7 psi. Finally the heat addition PORV actuation curve and previously developed mass addition PORV actuation curve are compared and the minimum value (of the two), at the same temperature, is used. For the (minimum) horizontal portion of the OPS actuation curve the mass addition is more limiting; and for the remainder of the curve the heat addition is more limiting.

8.0 OPS OPERABLE - UNDERSHOOT ANALYSIS

The undershoot analysis scenario is consistent with inadvertent mass addition at low RCS temperature. The analysis is meant to be reasonably realistic (not a licensing/FSAR type). The concern is operational, and potentially significant as too low an RCS pressure will endanger the RCP seals.

Indian Point Unit 2 plant procedures allow operation of RCPs down to 325 psig and require tripping them at 300 psig. These procedures were developed with OEM guidance.

Although current IP2 operation is limited to a non-water solid pressurizer the present analysis includes spurious SI (or charging) leading to a water solid pressurizer. Since the heat addition case does not go water solid (and is self limiting) it is much less severe than the mass addition event. For the mass addition event (and assuming the operator terminates the mass addition while the PORVs are fully open), the pressure will decrease to the close setpoint at which point the PORVS will start to close, closing over the stroke time. The minimum pressure will be reached when the PORVS have closed (note that the pressure drop will be essentially instantaneous and it will remain at the minimum value for a considerable time).

The analysis was provided in Ref.[2], and the assumptions include: (1) pressurizer water solid, (2) initially 466 psig RCS pressure, (3) 0 psig back pressure, (4) no mass addition to the RCS, (5) both (two) PORVs initially open, C_V = 2 x 50 GPM/(psi)$^{1/2}$, (6) PORV flow area varies linearly from open to close over the valve stroke time, (7) constant RCS temperature of 100°F, (8) PORV close setpoint 7.5 psi below the open (466 psi) value, and 0% SGTP.

These generally are conservative/realistic assumptions, except that the new OPS actuation pressure will be about 515 psig (at low RCS temperatures), and the case of a pressurizer bubble is also analyzed. (Also 0% SGTP is realistic, and the effect of changes in RCS liquid volume is minor).

A worst case scenario, the pressurizer water solid case, was reanalyzed assuming the PORVs were fully open at the OPS close setpoint of 507.5 psig (515-7.5) i.e. the initial pressure was not further reduced by pressure uncertainty as this is not a licensing type analysis.

For a best estimate scenario, the revised assumption is: (5) two PORVS, each with a flow area of 1.398 in^2, and a nitrogen bubble (c_p/c_v=1.4). As discussed for heat addition (OPS operable), the analysis assumes sonic velocity at the PORV.

A best estimate scenario is initial pressurizer level at 50% and 5 minute operator action time to secure one operating SI (resulting in a level of 71.5%). At 5 minutes the PORVs are fully open at the close setpoint of 502.5 psig, and the gas temperature is assumed to be at the isentropic compression value, consistent with the pressurizer level change from 50% to 71.5%.

Table of minimum indicated RCS pressures (psig) at elevation 62 ft		
PORV Stroke Time (s)--->	1.2	2.4
Worst Case	420	348
Best Estimate	487	476

As shown in the above table, the minimum indicated RCS pressures is 348 psig (for an very conservative PORV close time of 2.4 seconds). Assuming the (realistic) design PORV close time (full open to full close) of 1.2 seconds, the minimum indicated pressure is 420 psig.* Including the OPS uncertainty of 30 psi (at low pressure) the result is 390 psig--considerably above the plant procedure value of 300 psig (for trip of operating RCPs). This difference (90 psi) provides considerable margin even if the actual OPS setpoint is established below the maximum permitted values.* Also, at low temperatures the subcooling margin will be large (at 200 psia the saturation temperature is 382°F) so this should not be a problem.

* The actual minimum RCP operating pressure at the inlet to the pump (for a realistic 0% SGTP) will be about 75 psi below the idle loop hot leg (OPS) indicated pressure--assuming the worst case of only 1 RCP operating.

9.0 OPS NOT OPERABLE - MASS ADDITION, ANALYSIS FOR RCS PRESSURE AND LEVEL LIMITS

Mass addition analysis was previously performed for the case of PORVs unavailable. This requires limits be established on the RCS pressurizer level (gas volume), the RCS pressure, and the number of pumps capable of injecting into the RCS*. Alternatively a RCS vent path area, and limitations on the number of pumps capable of injecting into the RCS, can be specified. If the former limitations are to be developed, it is necessary to terminate the mass addition--and this (operator action) is assumed to occur after 10 minutes.

9.1 Required Vent Area

The analysis for required vent size, utilizes the maximum mass addition flow, and calculates the vent size that will relieve this incoming flow at the lowest of the allowable RCS pressures including elevation difference to the PORVs. The vent size is based on ideal gas critical flow--i.e., the vent geometry does not include significant friction and/or other minor losses. The RCS relief and/or safety valves, or SG manways or vessel head, are the most likely vent paths--e.g., a safety can be gagged open.

The current Tech Spec values are: 1 PORV, or alternate vent(s) of minimum 2 in^2 area--for 3 charging pumps and 1 SI pump energized; and vent(s) of minimum 5 in^2 area--for 3 charging and 3 SI pumps energized. These vent areas ensure the RCS pressure will not exceed the (actual) minimum OPS actuation pressure. This (minimum OPS actuation pressure) value was 466 psig, and with the current (revised) analysis is 510 psig.

Assuming the RCS liquid volume at 0% SGTP, 493 psia at the PORVs, or 510 psig (at the pressure transducer elevation), and 100 psig PRT pressure: the inflow--due to 3 charging pumps and one SI--is less than one PORV relieving capacity. As discussed earlier the PORV flow area is 1.398 in^2, which is less than 2 in^2--so 2 in^2 vent area is still applicable.

Repeating the above analysis for various flows, the values are summarized in the following table.

* In prior analyses and Tech Spec, the description "energized pumps" was used. This is clarified by the description "capable of injecting into the RCS".

Table of Vent Size vs. Flow to the RCS

Flow Area (sq. in.)	Calculated Flow Rate (gpm)	Maximum No. Pumps
2.0	1003	1 SI and 3 charging (≤834 gpm)
5.0	2508	3 SI and 3 charging (≤ 2000 gpm)
1 PORV (1.398)	981	1 SI and 1 charging (≤ 834 gpm)
2 PORVs (2.796)	1962	2 SI and 3 charging (≤ 1768 gpm)

9.2 Mass Addition Due To Charging Pump(s)

For the mass addition (without fixed vent size) the previous analysis assumptions included: (1) 10 minutes of charging flow of 100 GPM per pump or (2) 10 minutes of constant SI flow to RCS, calculated at the initial (minimum) RCS pressure, (3) isentropic compression of (pressurizer) nitrogen bubble with $c_p/c_v = 1.4$.

For the present analysis the flow is assumed for 1, 2 or 3 charging pumps and the maximum RCS fluid volume (at 0% SGTP) is used.

In order to transition from the Tech Spec to plant procedures, the pressure bias, and temperature, pressure and pressurizer level uncertainty, are included.

10.0 OPS NOT OPERABLE - HEAT ADDITION, ANALYSIS FOR RCS PRESSURE AND LEVEL LIMITS

Heat addition analysis was previously performed for the case of PORVs unavailable. As for the mass addition analysis without OPS operable, this requires limits be established on the RCS pressurizer level (gas volume) and the RCS pressure, as well as the SG-to-RCS temperature difference. Unlike mass addition, 10 minute operator action is not required. Except for establishing sufficient letdown to counter any long term RCS heat up, once the RCP is started, with significantly hotter SGs, the bulk of the RCS liquid volume increase will occur rapidly and is essentially independent of operator actions.

Heat addition analysis was previously performed for the start of one RCP with SGs hotter than the RCS. Assumptions included: (1) pressurizer water level at 30%, (2) RCS pressure at 450 psi, (3) RCS temperature at 145°F, (4) isentropic compression of (pressurizer) nitrogen bubble with c_p/c_v = 1.4, (5) SG-to-RCS delta-T of 100°F, (6) and the maximum RCS fluid volume (at 0% SGTP) was used for the liquid expansion due to temperature increase. (The RCS temperature increase, due to RCP start with hotter SGs, was described in the heat addition, OPS operable, section.)

For the present analysis the assumptions are revised as follows: (2) & (3) the RCS pressure is developed as a function of RCS temperature (i.e., the RCS temperature restriction is removed; however, this results in new Tech Spec curves, Figure 3.1.A-5 and Figure 3.1.A-6), and (5) analysis is done both for 40 and 100°F hotter SGs. The allowable RCS pressures versus temperature, are shifted (as described before) for 40°F and 100°F hotter SGs--the shift is 23.6°F and 59°F (plus an additional 5°F as a margin for minor potential future changes to uncertainty, etc.), and 180°F becomes 209°F and 243°F, respectively. The resulting limit of RCS pressure exhibits a small negative slope at low RCS temperatures due to the RCS water expansion increasing with increasing temperature.

11.0 LIST OF REFERENCES

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