Mr. Paul H. Kinkel
Vice President, Nuclear Power
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenue
Buchanan, NY 10511

February 27, 1998

SUBJECT: ISSUANCE OF AMENDMENT FOR INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 (TAC NO. M96944)

Dear Mr. Kinkel:

The Commission has issued the enclosed Amendment No. 195 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2. The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated October 2, 1996, as supplemented July 31, 1997.

The amendment revises Figures 3.1.A-1, 3.1.A-2 and 3.1.A-3, Section 3.1.B and its Bases, Figures 3.1.B-1 and 3.1.B-2, and the Bases of Section 4.3 and Figure 4.3-1 of the TSs to incorporate the revised Indian Point Unit 2 Heatup and Cooldown Limit Curves for Normal Operation.

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

Sincerely,

ORIGINAL SIGNED BY:

Jefferey F. Harold, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-247

Enclosures: 1. Amendment No. 195 to DPR-26
2. Safety Evaluation

cc w/encls: See next page

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

February 27, 1998

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Vice President, Nuclear Power
Consolidated Edison Company
of New York, Inc.
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A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

Jeffrey F. Harold

Jefferey F. Harold, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-247

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2. Safety Evaluation

cc w/encls: See next page

Paul H. Kinkel
Consolidated Edison Company
of New York, Inc.

Indian Point Nuclear Generating
Station Units 1/2

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

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

DOCKET NO. 50-247

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 195
License No. DPR-26

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

   A. The application for amendment by Consolidated Edison Company of New York, Inc. (the licensee) dated October 2, 1996, supplemented July 31, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

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(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No.195, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

S. Singh Bajwa, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 27, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 195

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Pages
Figure 3.1.A-1	Figure 3.1.A-1
Figure 3.1.A-2	Figure 3.1.A-2
Figure 3.1.A-3	Figure 3.1.A-3
3.1.B-1 thru 3.1.B-8	3.1.B-1 thru 3.1.B-7
Figure 3.1.B-1	Figure 3.1.B-1
Figure 3.1.B-2	Figure 3.1.B-2
4.3-1	4.3-1
4.3-2	4.3-2
Figure 4.3-1	Figure 4.3-1

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

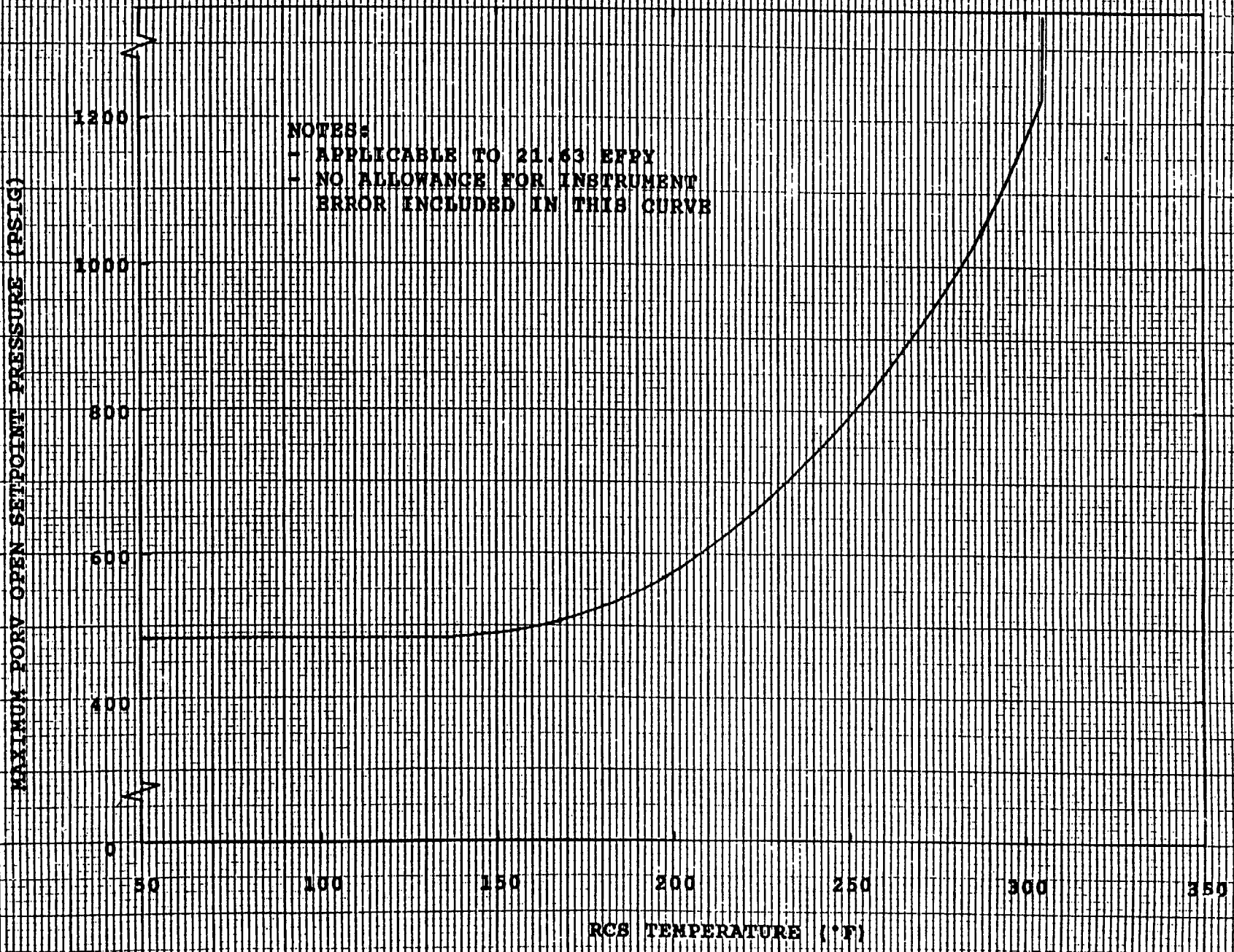


FIGURE 3.1.A-2
MAXIMUM PRESSURIZER LEVEL WITH PORVs
INOPERABLE AND ONE CHARGING PUMP ENERGIZED

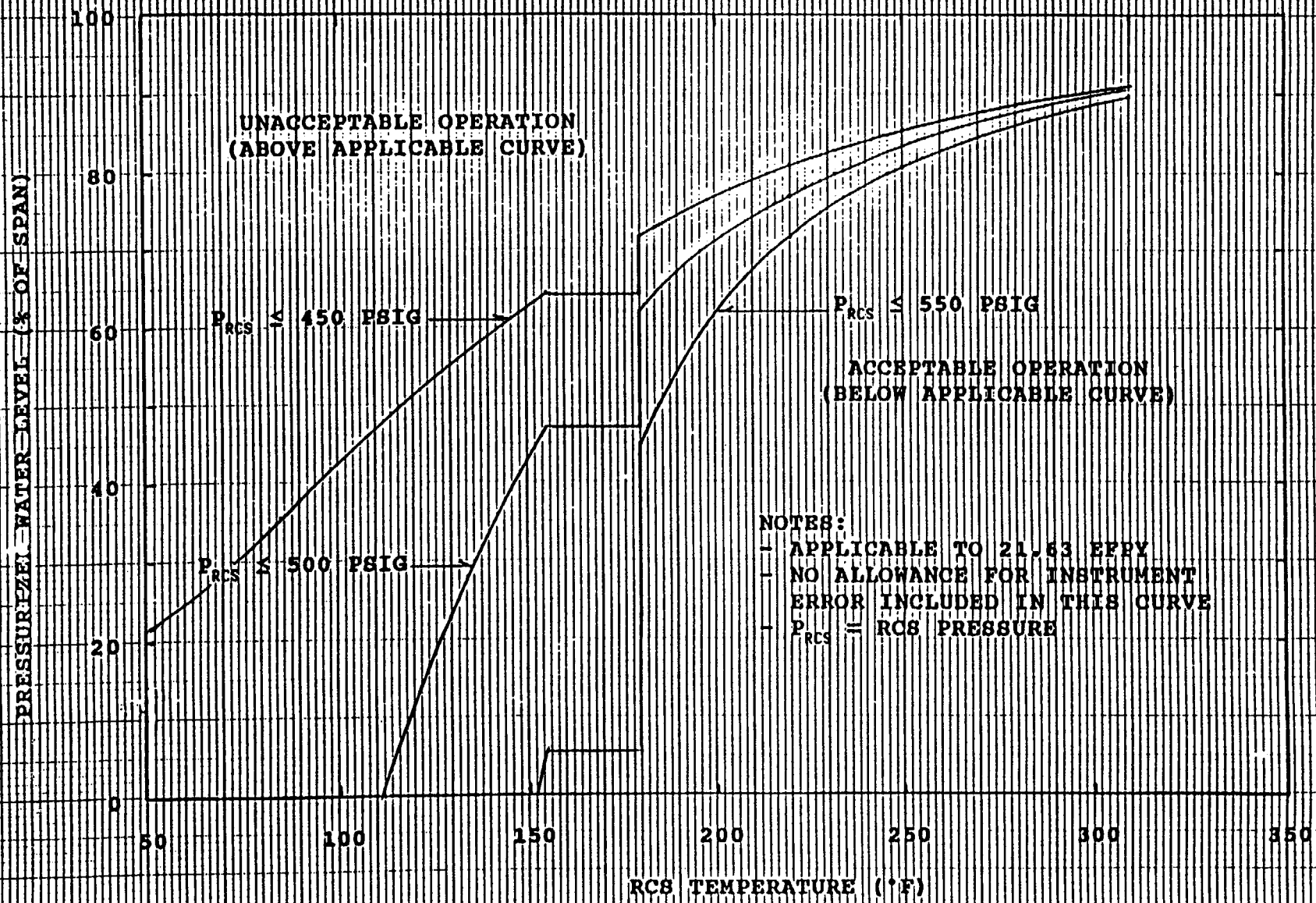
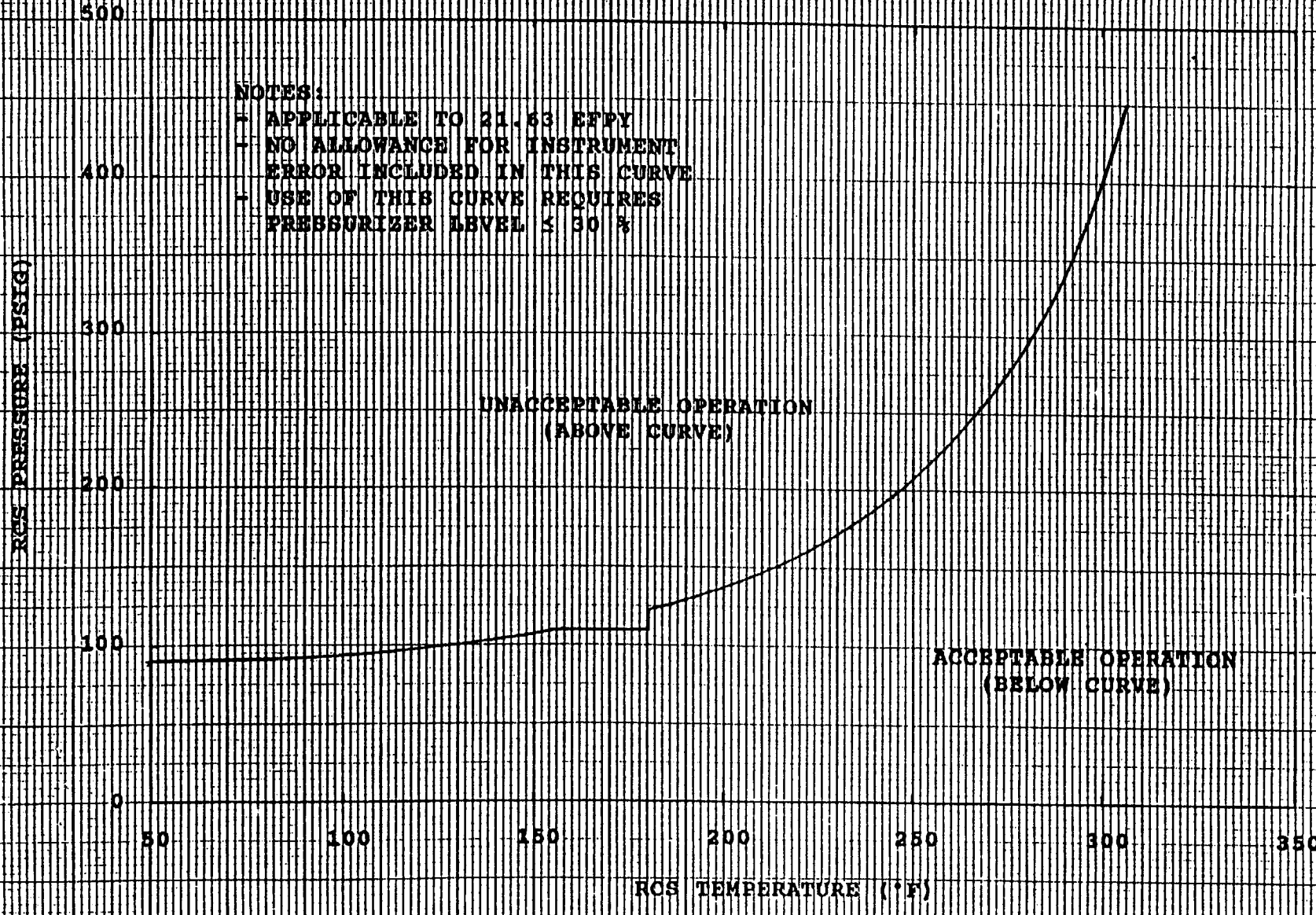


FIGURE 3.1.A-3
MAXIMUM REACTOR COOLANT SYSTEM PRESSURE FOR OPERATION WITH PORVs
INOPERABLE AND ONE SAFETY INJECTION PUMP AND/OR THREE CHARGING PUMPs ENERGIZED



B. HEATUP AND COOLDOWN

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 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 present 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 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.

---

* 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

Fracture Toughness Properties

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].

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.

The current heatup and cooldown curves are based upon a maximum fluence of 0.98×10^{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 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.

The actual shift in RT_{NDT} will be 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. These samples are evaluated according to ASTM E185[6]. 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[13] 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 the new Appendix G methods[14].

The maximum shift in RT_{NDT} at a fluence of 0.98 X 10^{19} n/cm^2, (nominal 21.63 EFPYs of operation) is projected to be 155.5°F at the 1/4 T and 105°F at the 3/4 T vessel wall locations, per Plate B2002-3 the controlling plate. The initial value of RT_{NDT} for this plate of the IP2 reactor vessel was 21°F. The heatup and cooldown curves have been computed on the basis of the RT_{NDT} of 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].

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[13] and MSE-REME-0076[14].

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:

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

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 III, 1974 Edition, Appendix G. (6) ASTM E185-79, Surveillance Tests on Structural Materials in Nuclear Reactors.

(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.

(14) MSE-REME-0076, "Indian Point Unit 2 Heatup and Cooldown Limit Curves for Normal Operation," P.A. Grendys, Westinghouse Electric Corporation.

FIGURE 3.1.B-1
COOLANT SYSTEM HEATUP LIMITATIONS

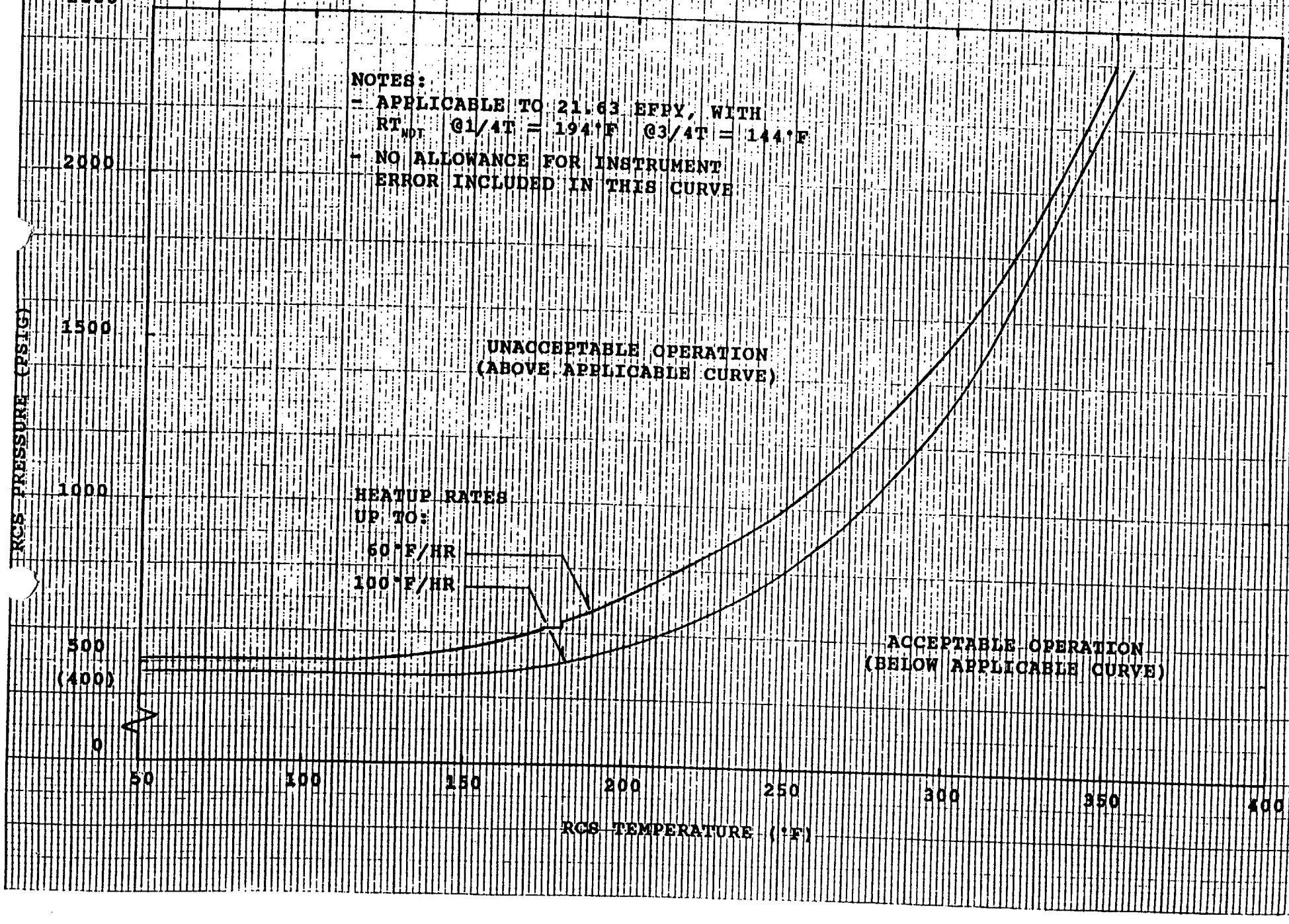
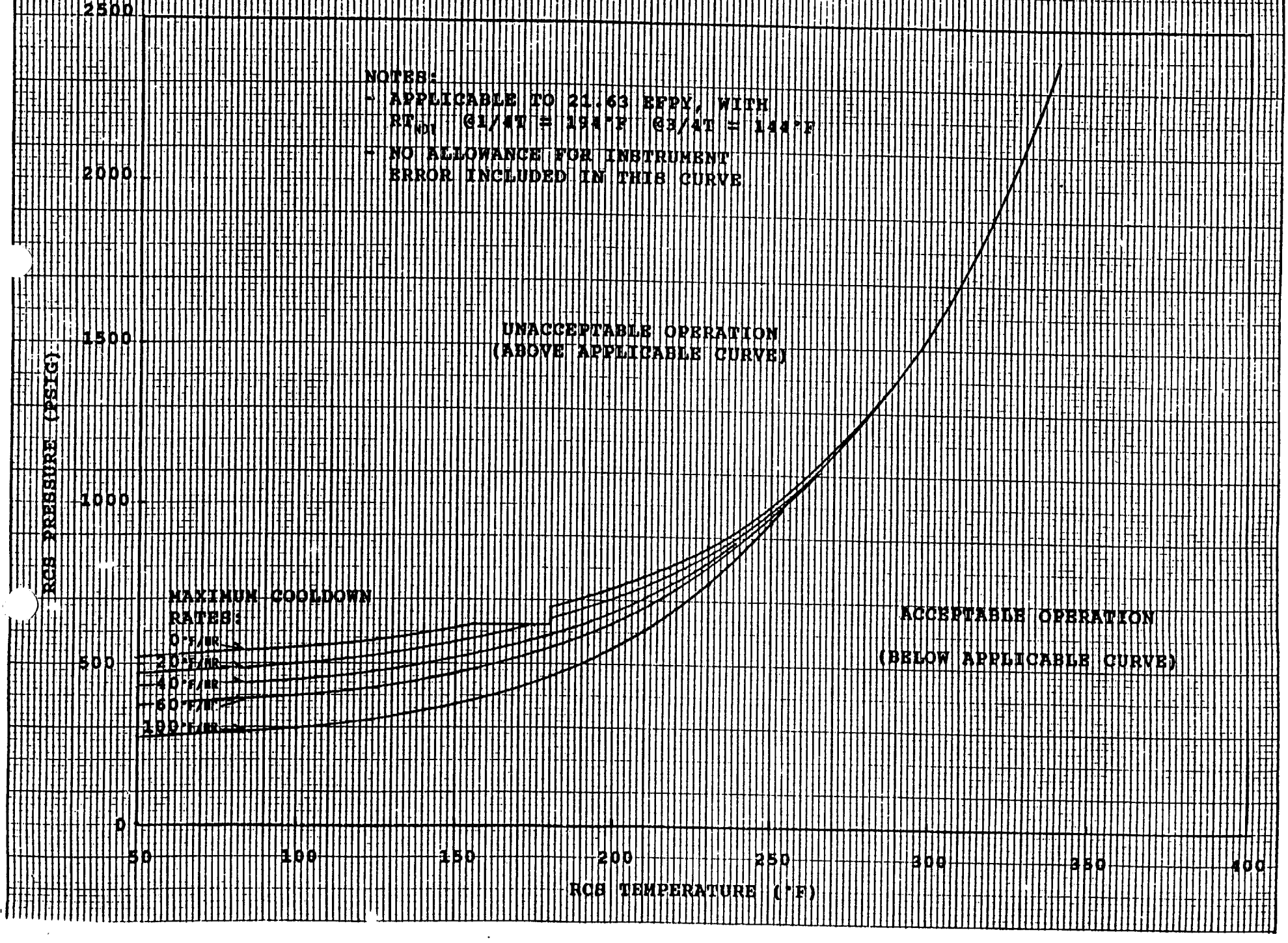


FIGURE 3.1.B-2
COOLANT SYSTEM COOLDOWN LIMITATIONS



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 normal opening, modification or repair.

Specifications

a. When the Reactor Coolant System is closed after it has been opened, the system will be leak tested at not less than 2335 psig at NDT requirements for temperature.

b. When Reactor Coolant System modification or repairs have been made which involve new strength welds on components, the new welds shall meet the requirements of the applicable version of ASME Section XI as specified in the Con Edison Inservice Inspection and Testing Program in effect at the time.

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

Basis

For normal opening, the integrity of the system, in terms of strength, is unchanged. If the system does not leak at 2335 psig (Operating pressure + 100 psi: ± 100 psi is normal system pressure fluctuation), it will be leak-tight during normal operation.

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. The temperatures are calculated in accordance with ASME Code Section III, 1974 Edition, Appendix G and the methods

described in reference 13 c echnical Specification 3.1.B. This co equires 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 effective full-power years are shown on Figure 4.3-1.

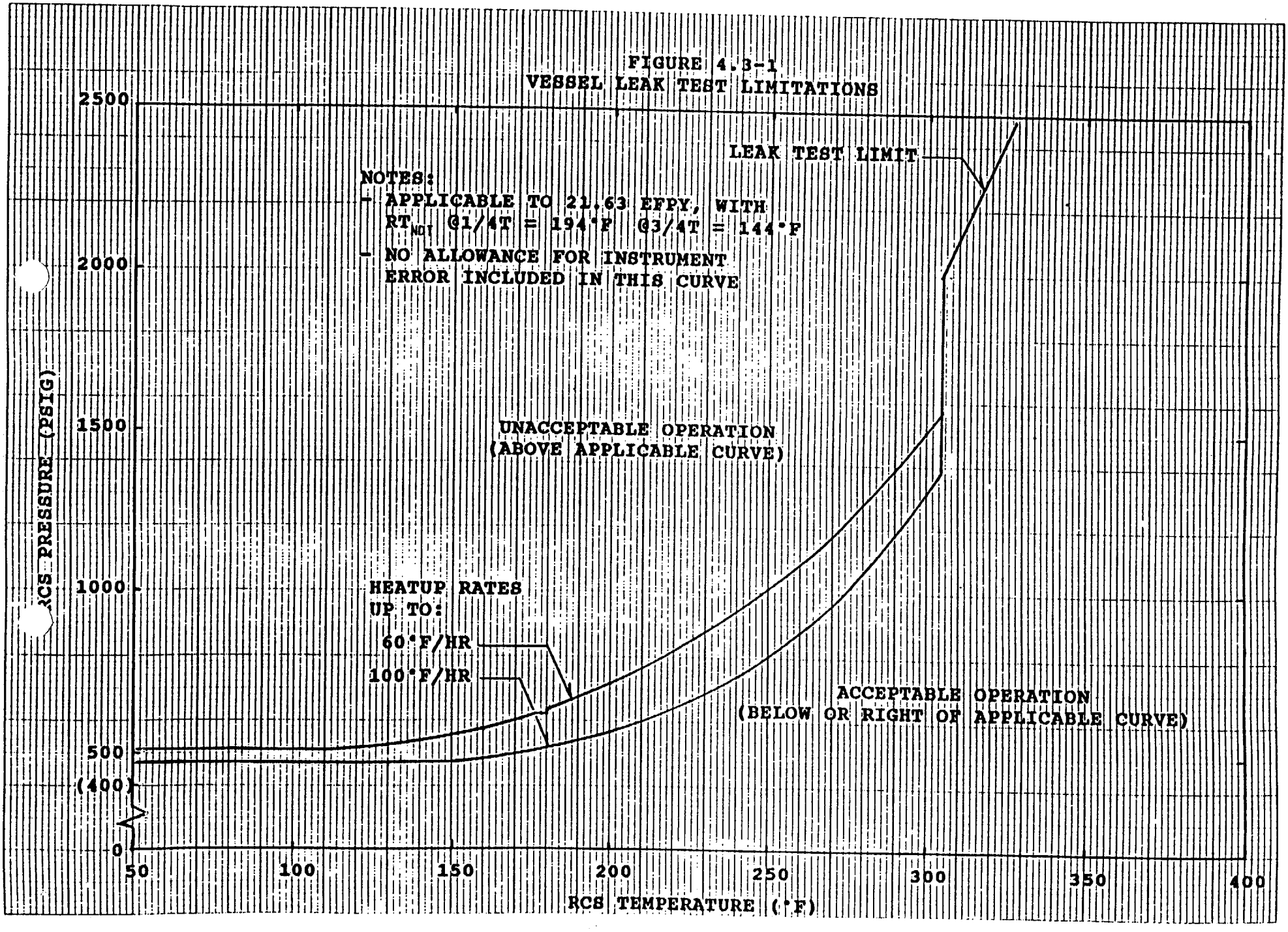
The heatup limits specified on the heatup curve, Figure 4.3-1, must not be exceeded while the reactor coolant is being heated to the inservice leak test temperature. For cooldown from the leak test temperature, the limitations of Figure 3.1.B-2 must not be exceeded. Figures 4.3-1 and 3.1.B-2 are recalculated periodically, using methods discussed in 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 0.98×10^{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.

Reference

UFSAR Section 4

FIGURE 4.3-1
VESSEL LEAK TEST LIMITATIONS



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 195 TO FACILITY OPERATING LICENSE NO. DPR-26

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

1.0 INTRODUCTION

By letter dated October 2, 1996, as supplemented July 31, 1997, the Consolidated Edison Company of New York, Inc. (the licensee) submitted a request to amend the Pressure and Temperature (P-T) limits in the Technical Specifications (TS) for Indian Point 2 (IP2) (Reference 1). This submittal is related to the exemption request (Reference 2) submitted on October 7, 1997, for using the methodology specified in the Appendix G in the 1996 Addenda to Section XI of the American Society for Mechanical Engineers (ASME) Code (the 1996 methodology) for developing P-T limits for IP2. The exemption request was approved by letter dated January 27, 1998. The July 31, 1997, letter provided clarifying information that did not change the initial proposed no significant hazards consideration.

The amendment was intended to remove the limiting conditions on the number of effective full-power years (EFPY) for the P-T limits in the current TS. The current P-T limits were generated using the Raju-Newman method, which is different from that of Appendix G and Standard Review Plan (SRP) 5.3.2, for calculating stress intensity factors. The staff rejected the use of the Raj-Newman method in the P-T limits calculation on October 21, 1991, but approved the then proposed P-T limits for reduced EFPYs: 16 EFPYs for the 60 °F/hr curves and 12 EFPYs for the 100 °F/hr curves. The licensee proposed to extend the validity of the P-T limit curves to 21.63 EFPYs by using the Raj-Newman method, and to demonstrate that the Raj-Newman method is equivalent to the 1996 methodology.

The staff evaluates the P-T limits based on the following NRC regulations and guidance: 10 CFR Part 50, Appendix G; 10 CFR 50.55a; Generic Letter (GL) 88-11; GL 92-01, Revision 1; GL 92-01, Revision 1, Supplement 1; Regulatory Guide (RG) 1.99, Revision 2 (Rev. 2); and SRP Section 5.3.2. GL 88-11 advised licensees that the staff would use RG 1.99, Rev. 2 to review P-T Limit Curves. RG 1.99, Rev. 2 contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy (USE) resulting from neutron radiation. GL 92-01, Rev. 1 requested that licensees submit their reactor pressure vessel (RPV) data for their plants to the staff for review. GL 92-01, Rev. 1, Supplement 1 requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations. These data are used by the staff as the basis for the staff's review of P-T Limit submittals, and as the basis for the staff's review of pressurized thermal shock (PTS) assessments (10 CFR 50.61 assessments). Appendix G to 10 CFR Part 50 requires that P-T limits for the RPV be at least as conservative as those obtained by applying the methodology of

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Appendix G to Section XI of the ASME Code. 10 CFR 50.55a specifies the addenda and edition of the ASME Code that is to be utilized by licensees in determining P-T limits. SRP 5.3.2 provides an acceptable method of calculating the P-T limits for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics (LEFM) methodology of Appendix G to the 1989 Edition of Section XI of the ASME Code. The basic parameter of this methodology is the stress intensity factor K_I, which is a function of the stress state and flaw configuration. The methods of Appendix G postulate the existence of a sharp surface flaw in the RPV that is normal to the direction of the maximum stress. This flaw is postulated to have a depth that is equal to one-fourth of the RPV beltline thickness and a length equal to 1.5 times the RPV beltline thickness. The critical locations in the RPV beltline region for calculating heatup and cooldown P-T Limit Curves are the 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations, which correspond to the depth of the maximum postulated flaw, if initiated and grown from the inside and outside surfaces of the RPV, respectively.

The Appendix G, 10 CFR Part 50 requires that licensees determine the adjusted reference temperature (ART or RT_{NDT}) and the Charpy USE at the maximum postulated flaw depth. The ART is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin (M) term. The ΔRT_{NDT} is a product of a chemistry factor and a fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Rev. 2 or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the chemistry factor was determined using the tables in RG 1.99, Rev. 2 or surveillance data. The margin term is used to account for uncertainties in the values of initial RT_{NDT}, copper and nickel contents, fluence and calculational procedures. RG 1.99, Rev. 2 describes the methodology to be used in calculating the margin term. In addition, Appendix G, 10 CFR Part 50 requires closure flange limitations.

It should be noted that the Combustion Engineering Owners Group (CEOG) provided Report CE NPSD-1039, Rev. 2, "Best Estimate Copper and Nickel Values in CE Fabricated Reactor Vessel Welds," on July 14, 1997, to the NRC for information only. This report contains weld chemistry data that is applicable to the IP2 RPV. The staff has used information in this report in conducting the review.

2.0 EVALUATION

2.2 The Adjusted Reference Temperature of The Limiting Beltline Material

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the reactor vessel of IP2. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The licensee determined that the material with the highest ART at 21.63 EFPY is the intermediate shell B2002-3, with 0.20% copper (Cu), 0.59% nickel (Ni), and an initial RT_{NDT} of 21 °F. The ART calculated by the licensee using the surveillance data (all three surveillance data are credible) according to Section 2.1 of RG 1.99, Rev. 2 is 193.5 °F for this beltline material. The ART calculated by the staff, using the same methodology, is 183.6 °F. This ART was calculated at 1/4T at 21.63 EFPY with a neutron fluence of 0.488E19 n/cm^2, which was derived linearly from the ID fluence of 1.21E19 n/cm^2 at 32 EFPY from NRC's reactor vessel integrity database (RVID).

The discrepancy in ART values was caused by the licensee's use of the peak 1/4T fluence of 0.585 n/cm^2 in its calculation. If the staff used the same peak fluence, there would be no discrepancy. Therefore, the licensee's ART for Plate B2002-3 is conservative and acceptable.

However, after applying the best estimate copper and nickel values in Report CE NPSD-1039, Rev. 2, to IP2 welds, the staff determined that instead of Plate B2002-3, the circumferential weld 9-042 with weld wire heat no. 34B009 is the limiting beltline material. According to the CE report, the best-estimate copper and nickel for this weld is 0.192% Cu and 1.038% Ni. The staff found that the number of EFPYs for weld 9-042 to reach an ART of 193.5 is 18, about 3 years less than the requested 21.63 EFPYs.

Unlike other P-T limits submittals, the licensee did not use the methodology in SRP 5.3.2 to carry out the next step of generating the P-T limits using the ART of the limiting beltline material. Instead, the licensee's proposed P-T limits were generated using the Raju-Newman method which is similar to the methodology in the 1996 Addenda of Section XI of the ASME Code.

2.3 The 1996 Asme Appendix G Methodology and the Raju-Newman Method

The 1996 methodology incorporates the most recent LEFM solutions regarding stress intensity factors due to pressure (K_{Ip}) and radial thermal gradients (K_{It}). These solutions are based on finite element analyses for inside surface flaws performed at Oak Ridge National Laboratories (ORNL), and work published by Electric Power Research Institute (EPRI) for outside surface flaws. The 1996 methodology simply provides better K_{Ip} and K_{It} estimation. It does not reduce safety margins associated with these K_I values. Hence, if applied correctly, it should be adequate for P-T limits application.

The 1996 methodology provides two methods for calculating K_{It} values: one that uses a generic radial thermal gradient (the first method) and the other that uses the worst thermal stress distribution at a specific time during heatup or cooldown from a plant-specific thermal analysis (the second method).

The Raju-Newman method, which uses the finite-element solutions by Raju-Newman, was used to generate the P-T limits in the submittal of 1991. The licensee compared the proposed 21.63 EFPY P-T limits of 1991 (approved by NRC for reduced EFPYs) with those calculated using the 1996 methodology, and reported that they are within 2% of each other and the differences cannot be discerned from the plotted curves.

2.4 Staff's Verification

The licensee is the first to apply a methodology equivalent to the 1996 methodology in its P-T limits submittal, in which the plant-specific thermal stress distribution during heatup and cooldown was used to calculate K_{It}. As a precaution, the staff compared the licensee's results based on the Raju-Newman method with those in the meeting report (Reference 3) dated August 6, 1996, of ASME Section XI Working Group on Operating Plant Criteria. This meeting report contains P-T limits for generic pressurized-water reactor (PWR) vessel by four organizations. One of them is the licensee's contractor, Westinghouse. All four organizations produced nearly identical results. The information in the meeting report played a role in the acceptance of the new Appendix G by the Section XI Subcommittee, the Boiler and Pressure Vessel Main Committee, and the ASME Board of Nuclear Codes and Standards.

The staff presents in the attached Table 1 results of analyses performed by the staff, the ASME Working Group, and the licensee for cooldown (100 °F/hr) using various methodologies. Column 2 is based on SRP 5.3.2 methodology using IP2 plant-specific parameters and was performed by the staff; Column 3 is based on the first method of the 1996 methodology using IP2 plant-specific parameters and was performed by the staff also; Column 4 is based on the second method of the 1996 methodology using generic PWR parameters and was performed by the ASME Working Group; Column 5 is based on the second method of the 1996 methodology using IP2 plant-specific parameters and thermal analysis by the licensee; and Column 6 is based on the Raju-Newman method using IP2 plant-specific parameters and thermal analysis by the licensee. As explained in the footnotes of Table 1 that data from other sources (i.e., Columns 4, 5, and 6) are not direct reproductions. The staff used a new parameter $T\text{-}RT_{NDT}$ to replace T so that an adequate comparison can be made. In addition, additional reviews have been conducted to heatup and leak test P-T limits.

Table 1 indicates that the P-T limits based on the first method of the 1996 methodology using IP2 plant-specific parameters (Column 3) and the P-T limits based on the second method of the 1996 methodology using generic PWR parameters by the ASME Working Group (Column 4) are very similar to those based on SRP 5.3.2 methodology using IP2 plant-specific parameters (Column 2). The table also indicated that the licensee's P-T limits based on the second method of the 1996 methodology (Column 5) and the P-T limits based on the Raju-Newman method (Column 6), using IP2 plant-specific parameters and thermal analysis by the licensee, are very similar. However, the licensee's P-T limits are less conservative than the P-T limits from other sources in the lower pressure range (< 500 psi). To resolve the discrepancy, the licensee provided an explanation and the plant-specific input parameters to the thermal analysis for IP2 in its response (Reference 4) to staff's request for additional information (RAI). The information in Reference 4 revealed that the major contributor to the discrepancy is the plant-specific temperature-dependent properties used in the submittal instead of the constant property values used in the ASME study (Reference 3). The staff examined these input values and considered them reasonable when compared with those from the generic PWR vessel study in Reference 3. The only exception is the film coefficient. The licensee used 7000 BTU/hr-ft-°F instead of a value of 1000 BTU/hr-ft-°F which were used in the current Appendix G and the ASME study (Reference 3) . The staff determined that the licensee's high film coefficient would reduce significantly the difference between the vessel wall and the water temperatures, but would have little effect on the P-T limits. Since the same computer code was used by Westinghouse for generating the P-T limits in the submittal and in the ASME study (Reference 3), the discrepancy must result from the summing effect of the difference in each input parameter. As stated previously, the IP2 plant-specific input values provided in Reference 4 are reasonable, consequently, the staff determines that the 1996 methodology is an acceptable alternative.

The pressure of the proposed P-T limits using the Raju-Newman method of 1991 is about 10 psi higher than those by the 1996 methodology. This was caused by the slightly different influence coefficients used by these two approaches for K_{Ip} and K_{It} calculations due to evolution of the essentially the same methodology. Since the differences cannot be discerned from the plotted curves, the staff concludes that the licensee's Raju-Newman method is equivalent to the 1996 methodology.

2.5 P-T Limitations Based on Closure Flange Materials And Upper Shelf Energy of Beltline Materials

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P-T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.A.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120 °F for normal operation and by 90 °F for hydrostatic pressure tests and leak tests. Based on the flange reference temperatures of 60 °F, the minimum allowable temperature of this region is 180 °F. These limits are shown on Figures 3.1.B-1 and 3.1.B-2 of the submittal, and the staff has determined that the proposed P-T limits satisfy the requirements in Section IV.A.2 of Appendix G.

Appendix G also requires that the predicted Charpy USE at end-of-license (EOL) for vessel beltline materials be above 50 ft-lb or that licensees demonstrate that lower values of Charpy USE will provide margins of safety equivalent to those required by Appendix G of Section XI of the ASME Code. This USE requirement is satisfied because all beltline materials have EOL USEs above 50 ft-lb.

3.0 CONCLUSIONS

The staff concludes that the proposed P-T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid for 18 EFPYs. Since the 1996 methodology simply provides better K_{Ip} and K_{It} estimation, and does not reduce safety margins associated with these K_I values, the staff endorsed the use of it as an acceptable alternative to the current Appendix G methodology for P-T limits generation. The staff accepts the Raju-Neuman method because the P-T limits that were generated by it are very close to those produced by the 1996 methodology.

The proposed P-T limits are for 21.63 EFPYs. However, due to the change of the limiting beltline material from Plate B2002-3 to weld 9-042 based on the best estimate copper and nickel values in Report CE NPSD-1039, Rev. 2, the proposed P-T limits is approved for 18 EFPYs for heatup, cooldown, and hydrotest. The P-T limits for 18 EFPYs meet the intent of the beltline material requirements in Appendix G of 10 CFR Part 50. They also satisfy Generic Letter 88-11 because the method in RG 1.99, Rev. 2 was used to calculate the ART. The proposed P-T limits approved for 18 EFPYs may be incorporated into the IP2 Technical Specifications. The proposed changes to the Bases are consistent with Appendix G in the 1996 Addenda and are also acceptable.

4.0 REFERENCES

1. October 2, 1996, letter from Stephen E. Quinn, (ConEd) to USNRC Document Control Desk, subject: "Indian Point 2 Plant - Proposed Changes to Technical Specifications Regarding Pressure-Temperature (P-T) Limits (Heatup and Cooldown Curves)."

2. October 7, 1997, letter from P. H. Kinkel, (ConEd) to USNRC Document Control Desk, subject: "Request for Exemption from the Requirements of 10 CFR 50.60: Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation, to Use the 1996 Addenda of ASME Section XI, Appendix G, Article G-2000: Vessels."

3. August 6, 1996, Meeting Report of ASME Section XI Working Group on Operating Plant Criteria, Portland, Oregon.

4. July 31, 1997, letter containing licensee's response to staff RAI from Stephen E. Quinn, (ConEd) to USNRC Document Control Desk, subject: "Supplement to the Indian Point Unit No. 2 License Amendment Request Proposing Changes to the Technical Specifications Regarding Pressure-Temperature (P-T) Limits."

Table 1
A Comparison of P/(T-RTndt) Limits from Different Sources
(100 DEG F/hr Cooldown)

P	T-RTndt[1] (SRP 5.3.2 - Staff)	T-RTndt[1] (First Method of the 1996 Methodology - Staff)	T-RTndt[2] (Second Method of the 1996 Methodology - ASME Working Group)	T-RTndt[3] (Second Method of the 1996 Methodology - Licensee)	T-RTndt[4] (Raju-Newman Methodology-Licensee)
300	-14.8	-12.0	-10.1	-52.67	-60.17
400	15.5	17.0	11.4	- 0.64	- 3.88
500	36.5	37.4	34.3	27.15	24.83
600	52.5	53.1	51.5	46.50	44.60
700	65.6	65.9	65.8	61.23	59.83
800	76.8	76.7	74.1	73.35	72.15
900	86.2	86.0	85.0	83.58	82.52

1. Performed by the staff.

2. Extrapolating from the figure on the page after Page 7 of the ASME meeting report (not very accurate).

		P
T(1/4t) - RTndt = (135.0 + 33.9) - 179 =	-10.1	300
T(1/4t) - RTndt = (156.5 + 33.9) - 179 =	11.4	400
T(1/4t) - RTndt = (179.6 + 33.7) - 179 =	34.3	500
T(1/4t) - RTndt = (196.8 + 33.7) - 179 =	51.5	600
T(1/4t) - RTndt = (211.1 + 33.7) - 179 =	65.8	700
T(1/4t) - RTndt = (219.4 + 33.7) - 179 =	74.1	800
T(1/4t) - RTndt = (230.3 + 33.7) - 179 =	85.0	900

3. Results from using new Appendix G methodology - to obtain the metal temperature at 1/4t, a del(T) of 30 deg F was added to the coolant temperature from Table 6 of Attachment III of the submittal.

   $Tw = T(1/4t) - (\text{Cooling Rate})(\text{Thickness}^2)[1-(1-1/4)^2]/[2x(\text{Thermal Diffu.})]$
   $= T(1/4t) - (100/60)(8.7^2)[1-(1-1/4)^2]/[2x0.92]$
   $= T(1/4t) - 30$

   An example of the calculation:
   At 600 P, T=210 °F. $T_{1/4t}$ - RTndt = (210 + 30) - 193.5 = 46.5

4. Results from using Raju-Newman methodology - to obtain the metal temperature at 1/4t, a del(T) of 30 deg F was added to the coolant temperature from Table 7 of Attachment III of the submittal.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (61 FR 58901). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

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

Principal Contributor: S. Sheng

Date: February 27, 1998

DATED: February 27, 1998

AMENDMENT NO.195TO FACILITY OPERATING LICENSE NO. DPR-26-INDIAN POINT UNIT 2

Docket File
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