ATTACHMENT A

Technical Specification

Page Revisions

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
May,1982

HEATUP AND COOLDOWN

Specifications

1. The reactor coolant temperature and pressure and system heatup and cooldown rates averaged over one hour (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1-1 and Figure 3.1-2 for the service period up to 7 effective full-power years. The heatup or cooldown rate shall not exceed 100°F/hr.

   a. Allowable combinations of presssure 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 3.1-1 and Figure 3.1-2 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 3.1-1 and Figure 3.1-2 shall be recalculated periodically using methods discussed in WCAP-7924A and results of surveillance specimen testing as covered in WCAP-7323(7) and as specified in Specification 4.2. 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 6 months prior to scheduled specimen removal.

3. 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.

4. 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.

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

An approximation of the maximum integrated fast neutron ($E > 1Mev$) exposure is given by Figure 2-4 of WCAP 7924A(4). Exposure of the Indian Point Unit No. 2 vessel will be less than that indicated by this figure.

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 & Pressure Vessel Code, 1974 Edition, Section III, Appendix G, and the calculation methods described in WCAP-7924A(4).

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) The second surveillance capsule was removed during the 1978 refueling outage. This capsule has been tested by SWRI and the results have been evaluated and reported.(9) Based on the SWRI evaluation, heatup and cooldown curves (Figures 3.1-1 and 3.1-2) were developed for up to seven (7) effective full power years (EFPYs) of reactor operation.

The maximum shift in RT_{NDT} after 7 EFPYs of operation is projected to be 130°F at the 1/4T and 65°F at the 3/4T vessel wall locations, per Plate B2002-3 the controlling plate. The initial value of RT_{NDT} for the IP2 reactor vessel was 60°F based on Plates B2002-1 and B2002-3 as shown in Table 3.1-1. The heatup and cooldown curves for 7 EFPYs 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 through that time period.

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 discussed in detail in WCAP-7924A.(4)

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 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 3.1-2 represent a composite curve consisting of the more conservative values calculated for steady state and the specific cooling rate shown.

Details of these calculations are provided in WCAP-7924A(4).

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 FSAR, Section 4.1.5

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

(3) Indian Point Unit No. 3 FSAR, 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 02-4531 - "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule T," E.B. Norris, June 30, 1977.

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

Amendment No.

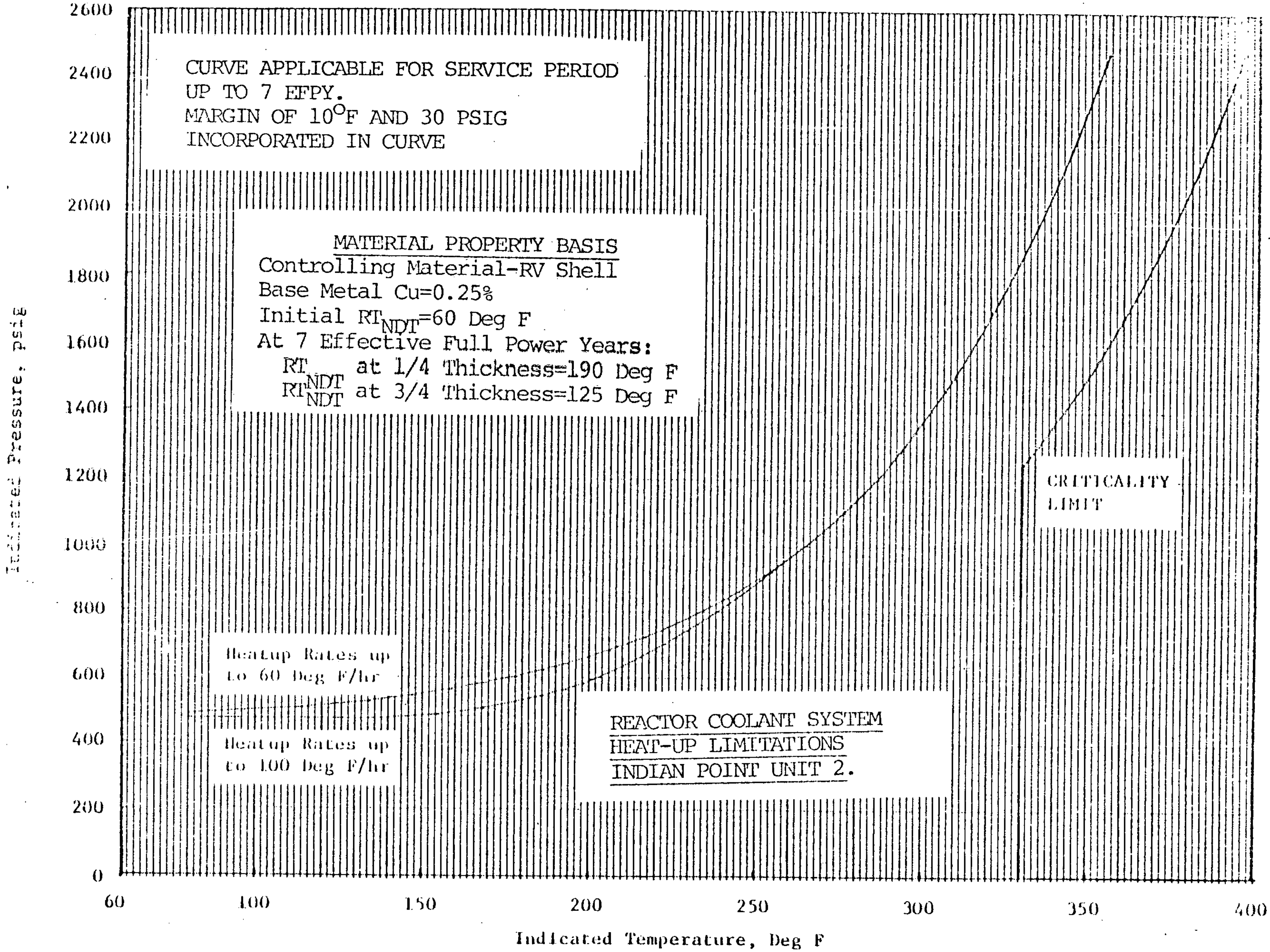

Figure 3.1-1

Amendment No.

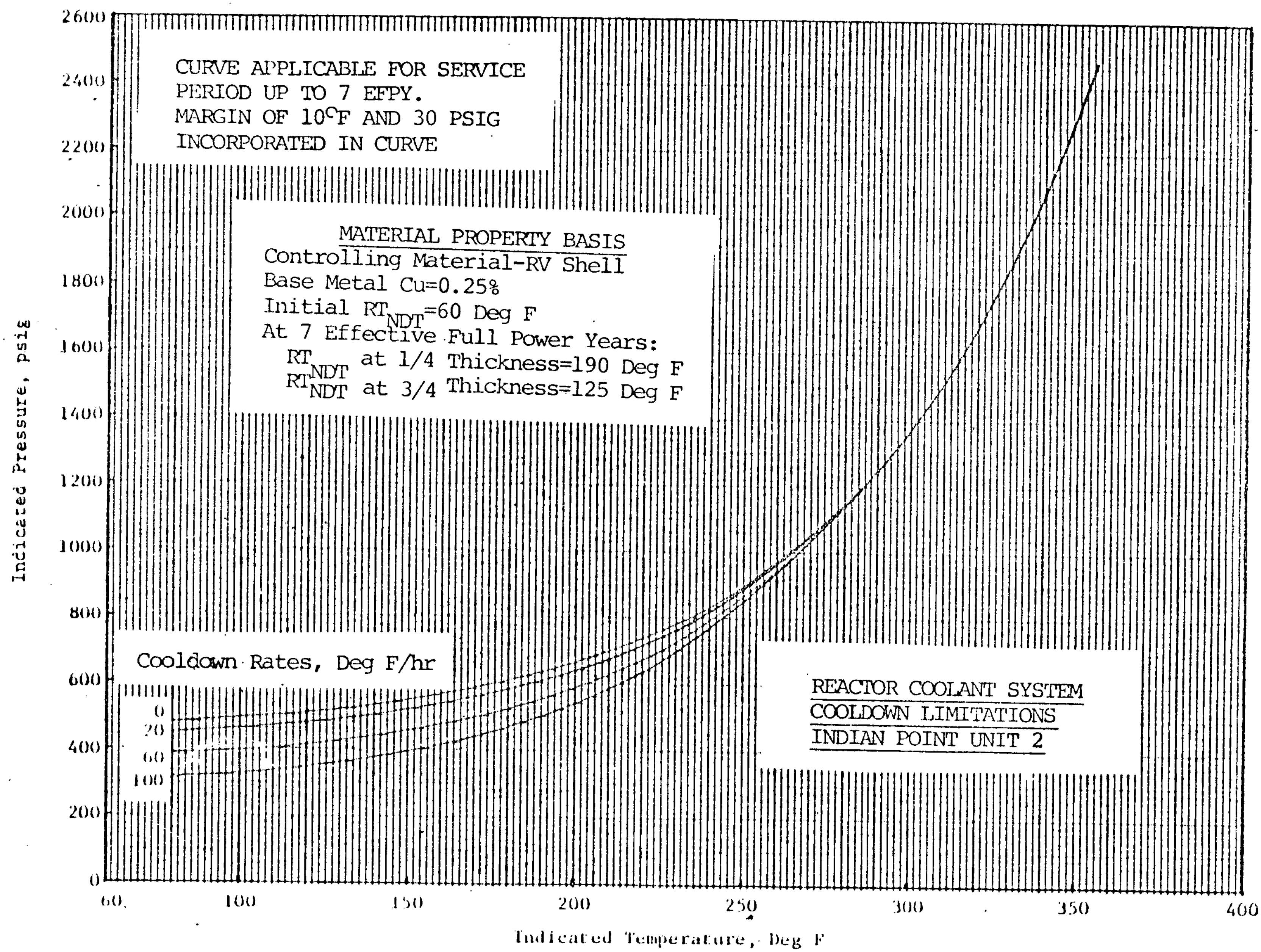

Figure 3.1-2

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.

Specification

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 modifications 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 seven (7) effective full-power yrs. of operation. Figure 4.3-1 will be recalculated periodically. Allowable pressures during cooldown for the leak test temperature shall be in accordance with Figure 3.1-2.

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: $\pm$ 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. 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 seven (7) effective full-power years, it is predicted that the highest RT_{NDT} in the core region taken at the 1/4 thickness will be $190^{\circ}F$. The minimum inservice leak test temperature requirements for periods up to seven (7) 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 3.1-2 must not be exceeded. Figures 4.3-1 and 3.1-2 are recalculated periodically, using methods discussed in WCAP-7924A and results of surveillance specimen testing, as covered in WCAP-7323.

Reference

1. FSAR, Section 4.

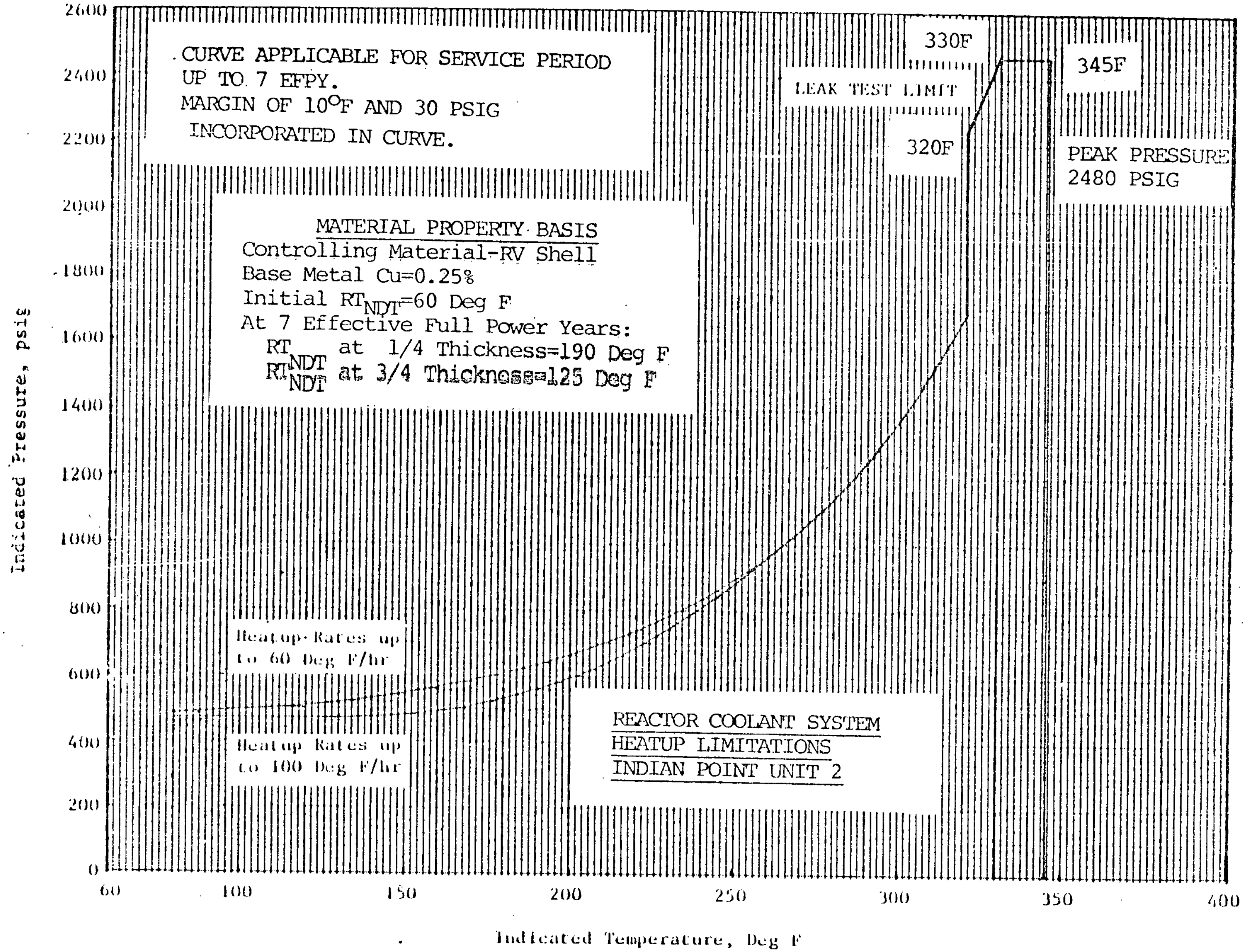
CURVE APPLICABLE FOR SERVICE PERIOD UP TO 7 EFPY.
MARGIN OF 10°F AND 30 PSIG INCORPORATED IN CURVE.
MATERIAL PROPERTY BASIS
Controlling Material-RV Shell
Base Metal Cu=0.25%
Initial RTNDT=60 Deg F
At 7 Effective Full Power Years:
RTNDT at 1/4 Thickness=190 Deg F
RTNDT at 3/4 Thickness=125 Deg F
330F
345F
LEAK TEST LIMIT
320F
PEAK PRESSURE 2480 PSIG
Heatup Rates up to 60 Deg F/hr
Heatup Rates up to 100 Deg F/hr
REACTOR COOLANT SYSTEM
HEATUP LIMITATIONS
INDIAN POINT UNIT 2
Indicated Pressure, psig
0
200
400
600
800
1000
1200
1400
1600
1800
2000
2200
2400
2600
60
100
150
200
250
300
350
400
Indicated Temperature, Deg F

Figure 4.3-1

ATTACHMENT B

Safety Evaluation

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
May 1982

Safety Evaluation

The proposed changes to the technical specifications, contained in Attachment A to this Application, would incorporate the reactor coolant system heatup, cooldown and hydrostatic leak test limitations applicable through seven (7) effective full power years (EFPYs) of reactor operation. The basis for these changes is provided in the Southwest Research Institute (SWRI) Report entitled, "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule Y," dated November 1980.

The proposed changes have been reviewed by both the Station Nuclear Safety Committee and the Consolidated Edison Nuclear Facilities Safety Committee. Both committees concur that the proposed changes do not represent a significant hazards consideration and will not cause any change in the types or an increase in the amounts of efflents or any change in the authorized power level of the facility.

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

STATE OF NEW YORK)
)
COUNTY OF NEW YORK) ss:

AFFIDAVIT OF SERVICE

Jeffrey H. Lomm, being sworn, states:

That he is an Engineer employed by Consolidated Edison Company of New York, Inc., and that he has served the foregoing document, sworn to on May 5, 1982, entitled, "Application for Amendment to Operating License" by mailing a copy thereof, first class postage prepaid and properly addressed to the following person:

Hon. F. Webster Pierce
Mayor, Village of Buchanan
235 Tate Avenue
Buchanan, New York 10511

Jeffrey H. Lomm
Jeffrey H. Lomm

Subscribed and sworn to before me this 5 day of May, 1982.

Thomas Love
Notary Public

THOMAS LOVE
Notary Public State of New York
No. 31-2409638
Qualified in New York County
Commission Expires March 30, 1983