July 22, 1985

Docket No. 50-247

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Mr. John D. O'Toole
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New York, New York 10003

Dear Mr. O'Toole:

The Commission has issued the enclosed Amendment No. 96 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 in response to your application transmitted by letter dated February 28, 1985 as supplemented July 5, 1985.

The amendment revises the Technical Specifications to reflect revised pressure temperatures limitations for reactor coolant system heat up, cooldown and hydrostatic test through fifteen effective full power years.

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

Sincerely,

/s/JDNeighbors

Joseph D. Neighbors, Project Manager
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 96 to DPR-26
2. Safety Evaluation

cc: w/enclosures
See next page

ORB#1:DL	ORB#1:DL	BC-ORB#1:DL	OELD	AD:OR:DL
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7/11/85

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P PDR

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

CONSOLIDATED EDISION COMPANY OF NEW YORK, INC.

DOCKET NO. 50-247

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 96
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 February 28, 1985, as supplemented July 5, 1985, 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. 96 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 22, 1985

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 96 TO FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.1-4	3.1-4
3.1-4a	3.1-5
3.1-6	3.1-6
3.1-7	3.1-7
3.1-8	3.1-8
3.1-8a	3.1-8a
3.1-8b	3.1-8b
Figure 3.1-1	Figure 3.1-1
Figure 3.1-2	Figure 3.1-2
4.3-1	4.3-1
4.3-2	4.3-2
Figure 4.3-1	Figure 4.3-1

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 15 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 3.1-1 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 heat up 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 3.1-3 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. The specimens will be removed and examined at the following intervals:

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

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 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 FSAR. 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 FSAR. 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.

An approximation of the maximum integrated fast neutron ($E > 1$ Mev) 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] [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] Based on the SWRI evaluation, heatup and cooldown curves (Figures 3.1-1 and 3.1-2) were developed for up to fifteen (15) effective full power years (EFPYs) of reactor operation.

The maximum shift in RT_{NDT} after 15 EFPYs of operation is projected to be 142°F at the 1/4T and 71°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 34°F as described in Table 3.1-1. The heatup and cooldown curves for 15 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.[11]

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

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:

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

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 1/4T. The thermal gradients induced during cooldown tend to produce tensile stresses at the 1/4T location and compressive stresses at the 3/4 T position. Thus, the ID 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 calcuated 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 3.1-2 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 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.

TABLE 3.1.B-1

Indian Point Unit No. 2
Reactor Vessel Core Region Material

Plate	Copper Content	Initial RT NDT
B 2002-1	0.25	34°F
B 2002-2	0.14	21°F
B 2002-3*	0.14	21°F
HAZ	-	0°F
Weld Material	-	0°F

References:

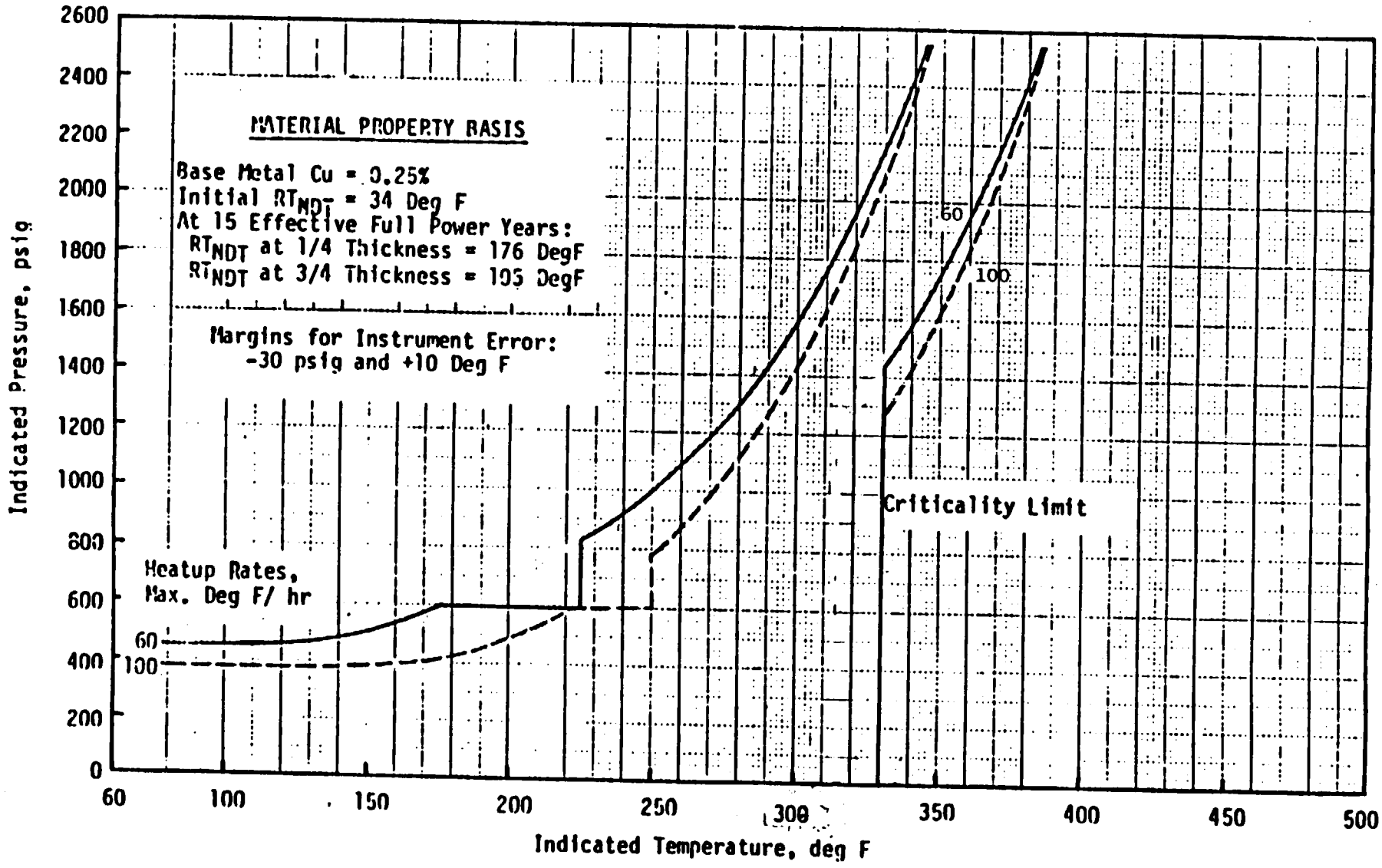
(1) Letter No. IPP-75-50, Westinghouse to Con Edison Dated May 16, 1975

(2) Letter dated March 29, 1978 from W. J. Cahill, Jr. (Consolidated Edison) to R. W. Reid (NRC), "Indian Point Unit No. 2 Reactor Vessel Material Surveillance Program."

(3) 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.

Notes:

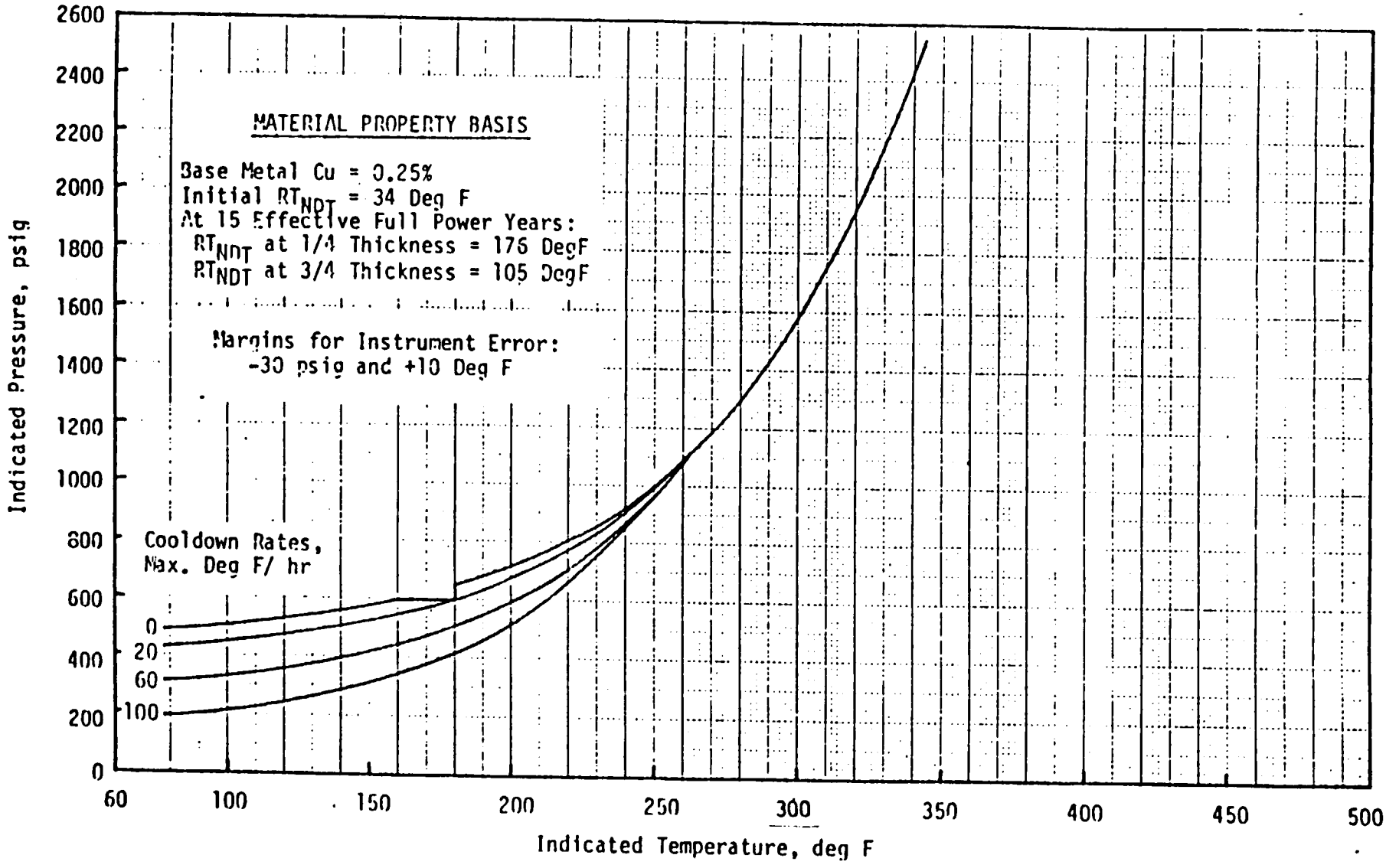
* Based on Reference (3) above, the bounding values for copper (0.25%) and initial RT_{NDT} (34°F) are applied to the controlling plate (B2002-3) for the purpose of generating the heatup and cooldown limitations.



INDIAN POINT UNIT NO. 2 COOLANT HEATUP LIMITATIONS
APPLICABLE FOR PERIODS UP TO 15 EFFECTIVE FULL POWER YEARS

Amendment No. 96

Figure 3.1-1



INDIAN POINT UNIT NO. 2 COOLANT COOLDOWN LIMITATIONS
APPLICABLE FOR PERIODS UP TO 15 EFFECTIVE FULL POWER YEARS

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 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 fifteen (15) effective full-power yrs. 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 3.1-2.

Basis

For normal opening, the integrity of the system, in terms of strength, is unchanged. If the system doe 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-desctructive 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 operaton.

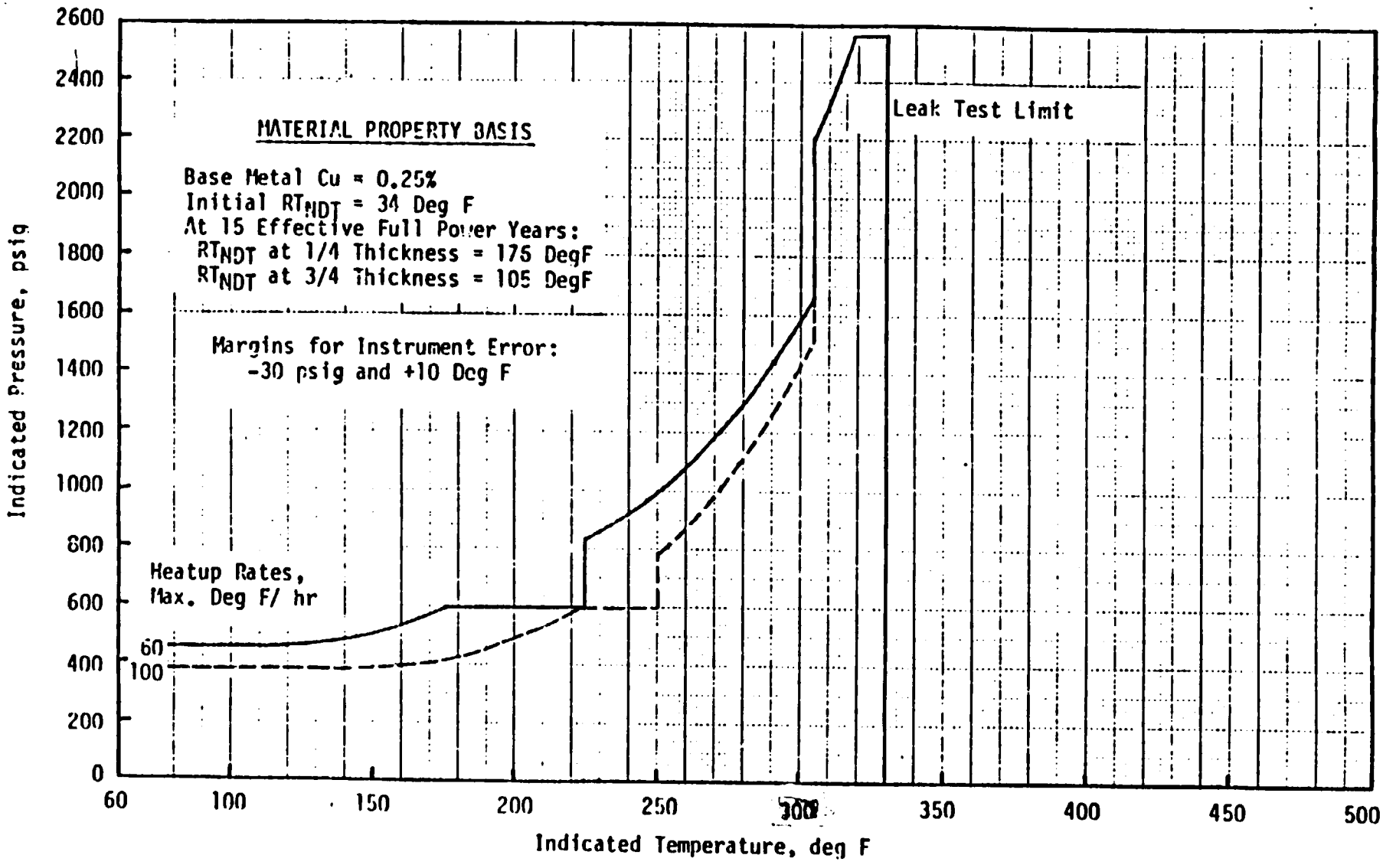
The inservice leak temperatures are shown on Figure 4.3-1. The temperatures are calculated in accordance with ASME Code Section III, 1974 Edition, Appendix G. This Code requires that a safety factor of 1.5 times the stress intensity factor caused by pressure be applied to the calculation.

For the first fifteen (15) effective full-power years, it is predicted that the highest RT_{NDT} in the core region taken at the 1/4 thickness will be 176°F. The minimum inservice leak test temperature requirements for periods up to fifteen (15) 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



INDIAN POINT UNIT NO. 2 VESSEL LEAK TEST LIMITATIONS
APPLICABLE FOR PERIODS UP TO 15 EFFECTIVE FULL POWER YEARS

Figure 4.3-1

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 96 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

Introduction

In letters from J.D. O'Toole to S.A. Varga, dated February 28 and July 5, 1985, the Consolidated Edison Company of New York, Inc. requested an amendment to the Technical Specifications for Indian Point Unit 2 (IP-2). The amendment proposes to incorporate into the IP-2 Technical Specifications revised reactor coolant pressure-temperature limits, which will be applicable through fifteen (15) effective full power years (EFPY) of reactor operation. In support of this amendment, the licensee referenced a Southwest Research Institute report entitled "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule Z," dated April 1984. This document was transmitted to the Staff in a letter from J.D. Toole to S.A. Varga dated May 7, 1984.

Discussion and Evaluation

Pressure-temperature limits must be calculated in accordance with the requirements of Appendix G, 10 CFR 50, which became effective on July 26, 1983. Pressure-temperature limits that are calculated in accordance with the requirements of Appendix G, 10 CFR 50 are dependent upon the initial RT_{NDT} for the limiting materials in the beltline, and closure flange regions of the reactor vessel and the increase in RT_{NDT} resulting from neutron irradiation damage to the limiting beltline material. The IP-2 reactor vessel was procured to ASME Code requirements, which did not specify fracture toughness testing to determine the initial RT_{NDT} for each vessel material. The licensee indicates that the initial RT_{NDT} for materials in the beltline regions of the IP-2 vessel were

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estimated using the method recommended by the staff in Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements." This method results in an initial RT_{NDT} for the limiting beltline weld metal of 0°F and an initial RT_{NDT} for the limiting beltline plate material of 34°F.

The licensee indicates that the limiting closure flange region materials are the closure flange forgings, which were fabricated to ASME Code SA 336 requirements and were heat treated to the quenched and tempered condition. Based on their chemical composition, these forgings are similar to that of ASME Code SA 508 Class 2 material. A conservative estimate of the initial RT_{NDT} of the licensee's closure flange base material may be based upon a conservative estimate of RT_{NDT} for quenched and tempered SA 508 Class 2 material. According to Table 4.4 of NUREG 0577, "Potential for Low Fracture Toughness and Lamellar Tearing of PWR Steam Generator and Reactor Coolant Pump Supports," the upper bound RT_{NDT} for quenched and tempered ASME SA 508 Class 2 material is 40°F. Thus a conservative estimate of the RT_{NDT} for the closure flange region forgings is 40°F.

The increase in RT_{NDT} resulting from neutron irradiation damage was estimated by the licensee using the empirical relationship documented in Regulatory Guide 1.99, Rev. 1, April 1977, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." This method of predicting neutron irradiation damage is dependent upon the predicted amount of neutron fluence and the amounts of copper and phosphorus in the beltline material. Using flux wire measurements and a two-dimensional discrete ordinate transport calculation, the Capsule Z Test Report indicates that the peak inside surface flux was 1.88×10^{10} n/cm^2 (E> 1MeV). This results in a predicted neutron fluence for 15 EFPY of 3.8×10^{18} n/cm^2 (E> 1MeV) at the 1/4 T beltline location and 9.5×10^{17} n/cm^2 (E> 1MeV) at the 3/4 T beltline location.

Capsule Z Test Report indicates that the limiting beltline plate, heat number B2002-3, has .25 copper and .010 phosphorus. In a letter from W. J. Cahill, Jr. to R. W. Reid, dated March 29, 1978 the licensee indicates that the IP-2 beltline was fabricated using RACO 3 weld wire heat numbers W5214 and 34 B009. RACO 3 weld wire heat number W5214 was used as surveillance material in Indian Point Unit 3. The amount of copper for this heat of weld wire was reported in WCAP-9491, Analysis of Capsule T from the Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program" as .15 percent. RACO 3 weld wire heat number 34 B009 was used in the fabrication of the Robinson Unit 2 reactor vessel beltline. The amount of copper for this heat of weld wire was reported in a letter from Carolina Power & Light Company to H. R. Denton, dated June 29, 1984 as 0.187 percent. Using the Regulatory Guide 1.99, Rev. 1 method of predicting neutron irradiation damage and these reported amounts of copper, the limiting beltline material would be heat number B2002-3 plate material.

Material from heat number B2002-3 plate was contained in the IP-2 surveillance capsule. In Table 1 we have compared the amount of increase in RT_{NDT} resulting from neutron irradiation damage observed on the capsule material to that predicted by Regulatory Guide 1.99, Rev. 1. The regulatory guide provides a conservative estimate of the amount of neutron irradiation damage to heat number B2002-3 plate material, since the predicted values exceed the observed values at two of the three neutron fluences.

Summary

The staff has used the method of calculating pressure-temperature limits in USNRC Standard Review Plan 5.3.2, NUREG-0800, Rev. 1, July 1981 to evaluate the proposed pressure-temperature limits. The amount of neutron

irradiation damage to the limiting beltline material, heat number B2002-3 plate was estimated using Regulatory Guide 1.99, Rev. 1. Our conclusion is that the proposed pressure-temperature limits meet the safety margins of Appendix G, 10 CFR 50 for (15) fifteen EFPY and may be incorporated into the plant's technical specifications.

Environmental Consideration

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 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 this amendment.

Conclusion

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: July 22, 1985

Principal Contributor:

B. Elliot

TABLE I

INCREASE IN REFERENCE TEMPERATURE, RT_{NDT} FOR SURVEILLANCE MATERIAL HEAT NO. B2002-3

Neutron Fluence (n/cm^2, E> 1MeV)	Increase in RT_{NDT} (°F) Observed from Capsule Test Data	Predicted by Regulatory Guide 1.99, Rev. 1
2.5×10^{18}	115	110
4.7×10^{18}	145	150
9.6×10^{18}	180	215