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Docket No. 50-247

Mr. William J. Cahill, Jr.
Vice President
Consolidated Edison Company
of New York, Inc.
4 Irving Place
New York, New York 10003

Dear Mr. Cahill:

The Commission has issued the enclosed Amendment No. 49 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2. This amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated January 9, 1979.

The amendment revises the reactor coolant system pressure and temperature heatup and cooldown curves.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original Signed By

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 49 to DPR-26
2. Safety Evaluation
3. Notice

cc w/encl:
See next page

Completes 10938

7903220243

Concur subject to additional language in FR notice as discussed with Lon Olshan

| OFFICE | DOR:ORB#1 | DOR:ORB#1 | OELD | DOR:ORB#1 | | |
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| SURNAME | CParrish:lb | LOlshan | S.H.Levin | ASchwencer | | |
| DATE | 3/2/79 | 3/1/79 | 3/1/79 | 3/2/79 | 3/1/79 | |

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C

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

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. 49
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 January 9, 1979, 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 50 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. 49 , 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

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 1, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 49

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

| Remove | Insert |
|---|---|
| Pages 3.1-4 thru 3.1-8(d) | Pages 3.1-4 thru 3.1-8(b) |
| Figure 3.1-1 | Figure 3.1-1 |
| Figure 3.1-2 | Figure 3.1-2 |
| Page 4.3-1 | Page 4.3-1 |
| Page 4.3-2 | Page 4.3-2 |
| Figure 4.3-1 | Figure 4.3-1 |

## B. 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 5 effective full-power years. The heatup or cooldown rate shall not exceed 100°F.

   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 3.1-1 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 re-calculated periodically using methods discussed in WCAP-7924A and results of surveillance specimen testing as covered in WCAP-7323.[7] 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 rate averaged over one hour shall not exceed 200°F/hr. 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.

## 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. This capsule has been tested by Southwest Research Institute (SWRI) and the results have been evaluated and reported.[8] Based on the SWRI evaluation, heatup and cooldown curves (Figures 3.1-1 and 3.1-2) were developed for up to five (5) effective full power years (EFPYs) of reactor operation.

The maximum shift in $RT_{NDT}$ after 5 EFPYs of operation is projected to be 110°F at the 1/4T and 50°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 5 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-7924.[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:

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

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 conditions (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 ID flaw is clearly the worst case.

As in the case of heatup, allowable pressure temperature relations are generated for both steady state 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 $\Delta T$ 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 E185-70, 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.

TABLE 3.1-1

Indian Point Unit No. 2

Reactor Vessel Core Region Material

| Plate | Copper Content(1) | Lowest Temperature 50 ft. lb. Charpy (Longitudinal)(2) | Lowest Temperature 50 ft. lb. Charpy (Transverse)(3) | Assumed $RT_{NDT}$(4) |
|---|---|---|---|---|
| B 2002-1 | 0.25 | 60°F | 120°F | 60°F |
| B 2002-2 | 0.14 | 62°F | 112°F | 52°F |
| B 2002-3 | 0.14 | 75°F | 120°F | 60°F |
| HAZ | -- | -45°F | 5°F | -55°F |
| Weld Material | -- | -10°F | 15°F | -45°F |

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

(2) Reference: WCAP-7323, "Con Edison Indian Point Unit No. 2 Reactor Vessel Radiation Surveillance Program", Dated May 1969.

(3) Estimated from Longitudinal Data for 77 ft. lb/54 Mil Lateral Expansion (In All Cases, Expansion Data Exceed Requirements).

(4) Reference: ASME Boiler and Pressure Vessel Code, Section III, 1974 Edition Appendix G, $RT_{NDT} = T_{cv} - 60°F$

$T_{cv}$ = Transfer Charpy Temperature at 50 ft. lb energy

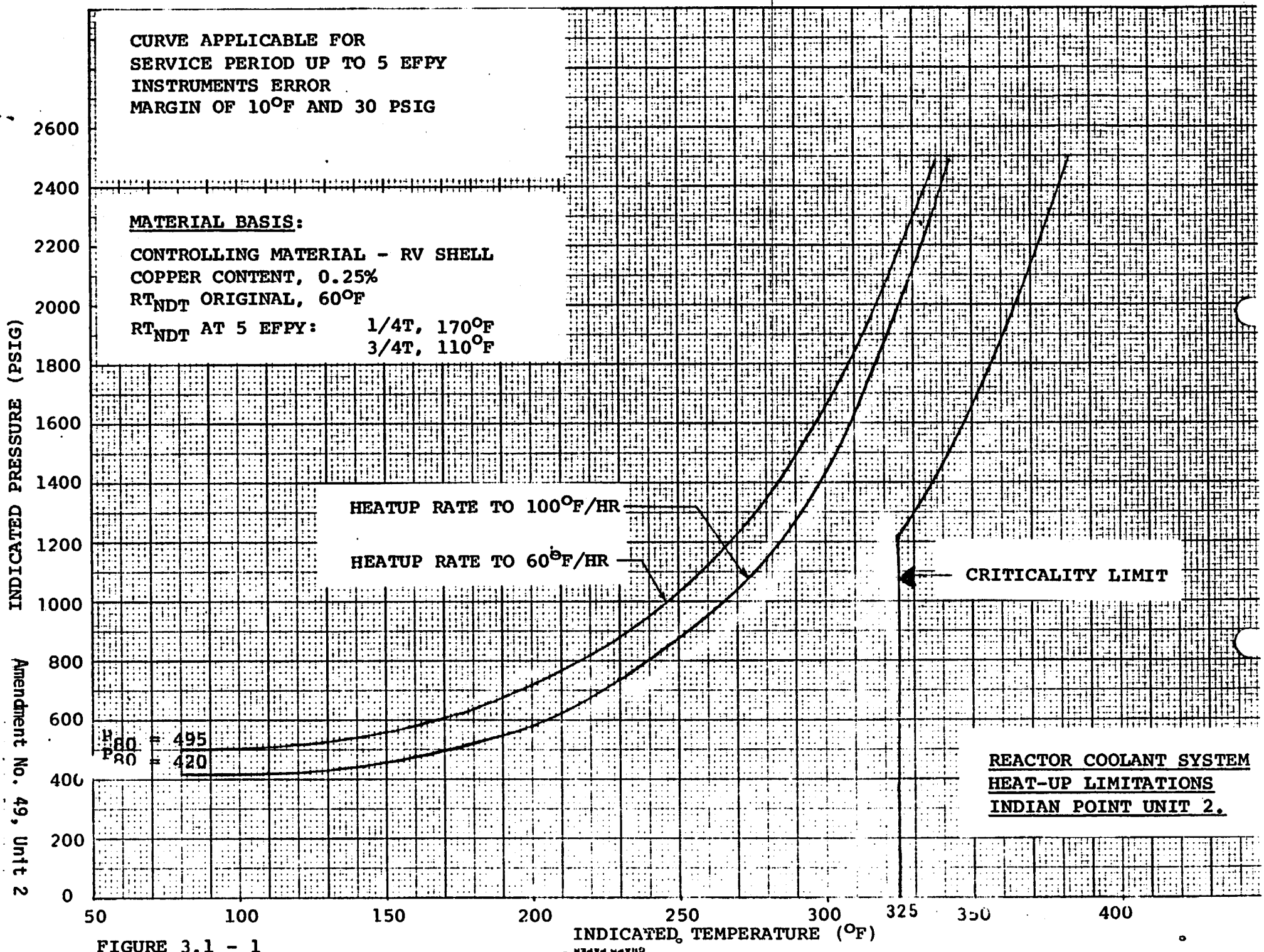
CURVE APPLICABLE FOR
SERVICE PERIOD UP TO 5 EFPY
INSTRUMENTS ERROR
MARGIN OF 10°F AND 30 PSIG
MATERIAL BASIS:
CONTROLLING MATERIAL - RV SHELL
COPPER CONTENT, 0.25%
RTNDT ORIGINAL, 60°F
RTNDT AT 5 EFPY: 1/4T, 170°F
3/4T, 110°F
HEATUP RATE TO 100°F/HR
HEATUP RATE TO 60°F/HR
CRITICALITY LIMIT
H80 = 495
P80 = 420
REACTOR COOLANT SYSTEM
HEAT-UP LIMITATIONS
INDIAN POINT UNIT 2.
INDICATED PRESSURE (PSIG)
INDICATED TEMPERATURE (°F)
0
200
400
600
800
1000
1200
1400
1600
1800
2000
2200
2400
2600
50
100
150
200
250
300
325
350
400

FIGURE 3.1 - 1

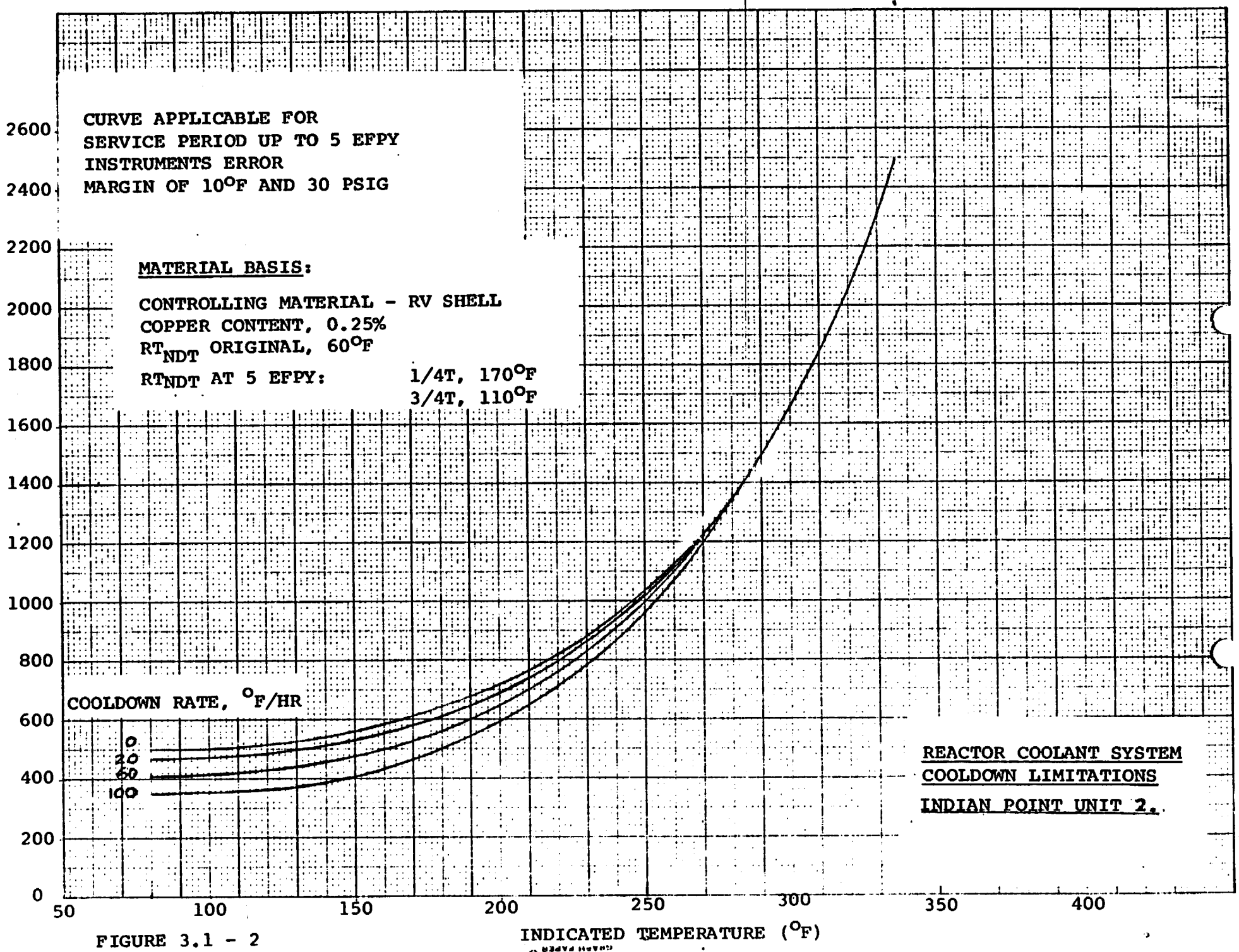
CURVE APPLICABLE FOR
SERVICE PERIOD UP TO 5 EFPY
INSTRUMENTS ERROR
MARGIN OF 10°F AND 30 PSIG
MATERIAL BASIS:
CONTROLLING MATERIAL - RV SHELL
COPPER CONTENT, 0.25%
RTNDT ORIGINAL, 60°F
RTNDT AT 5 EFPY: 1/4T, 170°F
3/4T, 110°F
COOLDOWN RATE, °F/HR
0
20
60
100
REACTOR COOLANT SYSTEM
COOLDOWN LIMITATIONS
INDIAN POINT UNIT 2.
INDICATED PRESSURE (PSIG)
INDICATED TEMPERATURE (°F)
0
200
400
600
800
1000
1200
1400
1600
1800
2000
2200
2400
2600
50
100
150
200
250
300
350
400

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 will meet the requirements of ASME Section XI, 1970 Edition IS400 and IS500.

c) The Reactor Coolant System leak test temperature-pressure relationship shall be in accordance with the limits of Figure 4.3-1 for heat-up for the first five (5) effective full-power yrs. of operation. Figure 4.3-1 will be recalculated periodically. Allowable pressures during 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: ±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 five (5) effective full-power years, it is predicted that the highest $RT_{NDT}$ in the core region taken at the 1/4 thickness will be 170°F. The minimum inservice leak test temperature requirements for periods up to five (5) 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 the WCAP-7924A and results of surveillance specimen testing, as covered in WCAP-7323.

Reference

1. FSAR, Section 4.

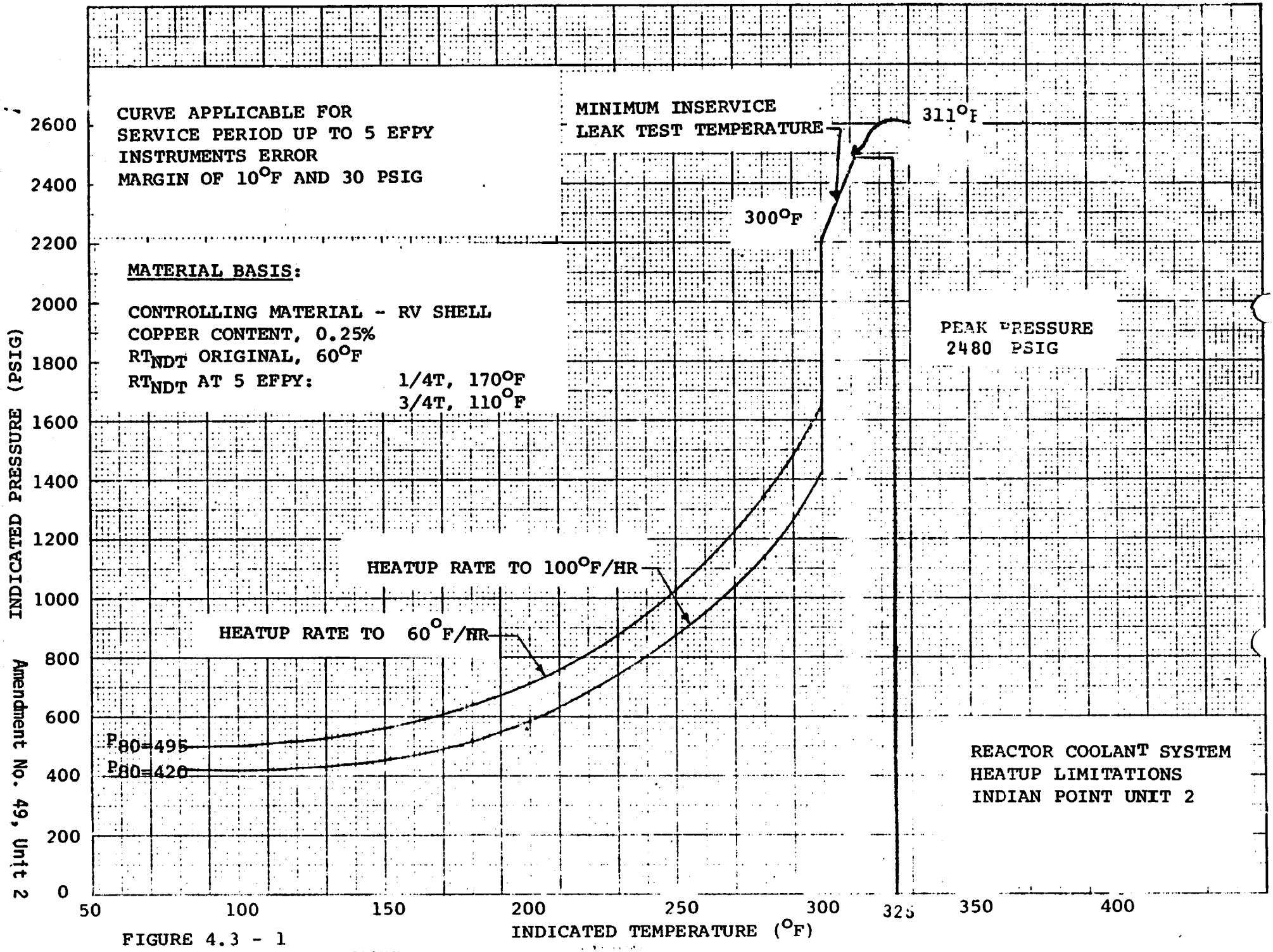
CURVE APPLICABLE FOR
SERVICE PERIOD UP TO 5 EFPY
INSTRUMENTS ERROR
MARGIN OF 10°F AND 30 PSIG
MINIMUM INSERVICE
LEAK TEST TEMPERATURE
311°F
300°F
MATERIAL BASIS:
CONTROLLING MATERIAL - RV SHELL
COPPER CONTENT, 0.25%
RTNDT ORIGINAL, 60°F
RTNDT AT 5 EFPY: 1/4T, 170°F
3/4T, 110°F
PEAK PRESSURE
2480 PSIG
HEATUP RATE TO 100°F/HR
HEATUP RATE TO 60°F/HR
P80=495
P80=420
REACTOR COOLANT SYSTEM
HEATUP LIMITATIONS
INDIAN POINT UNIT 2
INDICATED PRESSURE (PSIG)
INDICATED TEMPERATURE (°F)
0
200
400
600
800
1000
1200
1400
1600
1800
2000
2200
2400
2600
50
100
150
200
250
300
325
350
400

FIGURE 4.3 - 1

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 49 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

By letter dated January 9, 1979, Consolidated Edison Company of New York submitted an application to amend the Technical Specifications appended to Facility Operating License DPR-26 for Indian Point, Unit 2. The requested changes would modify the reactor coolant system pressure-temperature limits to account for increases in the reactor vessel metal reference nil-ductility temperature ($RT_{NDT}$) due to irradiation. Also enclosed in this submittal are copies of SwRI Final Report 02-4531 dated June 30, 1977, entitled, "Reactor Vessel Material Surveillance Program for Indian Point 2 Analysis of Capsule T." This document was used in the development of the operating limit curves.

## Discussion

10 CFR Part 50, Appendix G "Fracture Toughness Requirements", requires that pressure-temperature limits be established for reactor coolant system heatup and cooldown operations, inservice leak and hydrostatic tests, and reactor core operation. These limits are required to ensure that the stresses in the reactor vessel remain within acceptable limits. They are intended to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences.

The pressure-temperature limits depend upon the metallurgical properties of the reactor vessel materials. The properties of materials in the vessel beltline region vary over the lifetime of the vessel because of the effects of neutron irradiation. One principle effect of the neutron irradiation is that it causes the vessel material nil-ductility temperature ($RT_{NDT}$) to increase with time. The pressure-temperature operating limits must be modified periodically to account for this radiation induced increase in $RT_{NDT}$ by increasing the temperature required for a given pressure. The operating limits for a particular operating period are based on the material properties at the end of the operating period. By periodically revising the pressure-temperature limits to account for radiation damage, the stresses and stress intensities in the reactor vessel are maintained within acceptable limits.

7903220 247

The magnitude of the shift in $RT_{NDT}$ is proportional to the neutron fluence that the materials are subjected to. The shift in $RT_{NDT}$ can be predicted from Regulatory Guide 1.99. To check the validity of the predicted shift in $RT_{NDT}$, a reactor vessel material surveillance program is required. Surveillance specimens are periodically removed from the vessel and tested. The results of these tests are compared to the predicted shifts in $RT_{NDT}$, and the pressure-temperature operating limits are revised accordingly.

Evaluation

The revised operating limits are based on the calculation methods contained in Appendix G to ASME Code, Section III and the test data from material surveillance capsule T. The test data from capsule T is presented in SwRI Final Report 02-4531. The revised operating limit curves are proposed for operation through five EFPY. The present operating limit curves in the Technical Specifications are applicable for three EFPY.

We have reviewed the proposed changes to the operating limits for Indian Point 2 and have performed independent calculations to verify compliance with Appendix G, 10 CFR 50. We have also reviewed the test results on material surveillance capsule T. This capsule contained copper, nickel and cobalt wire dosimeters. Based on analysis of these dosimeters, it is concluded that the maximum neutron fluence received by the vessel wall in 1.42 EFPY was $6.97 \times 10^{17}$ n/cm$^2$. There may be about a 20% uncertainty in the fluence. However, from our review, we conclude the conservatisms in the calculational methods and the determination of the unradiated valve of $RT_{NDT}$ will compensate for this fluence uncertainty. Capsule T is a Type I capsule and contained test specimens from three vessel beltline plates. From the test data we conclude that plate material will be the limiting material for at least five EFPY. The limiting plate material has an initial $RT_{NDT}$ of 60°F. Its $RT_{NDT}$ increased by 130°F at a fluence of $2 \times 10^{18}$ n/cm$^2$.

Based on the test results from Capsule T we calculate that the $RT_{NDT}$ of the limiting vessel material at the 1/4 T* location will be 170°F at five EFPY. Based on this value of $RT_{NDT}$ we calculated pressure-temperature operating limits in accordance with the procedures given in Appendix G to ASME Code, Section III. These calculations showed that the operating curves proposed by the licensee are acceptable for operation through five EFPY and for this operating period are in accordance with Appendix G, 10 CFR 50. Comformance with Appendix G to 10 CFR Part 50 in establishing safe operating limitations will ensure adequate safety margins during operation, testing, maintenance and postulated accident conditions and constitutes an acceptable basis for satisfying the requifements to NRC General Design Criterion 31, Appendix A, 10 CFR Part 50.

*1/4 T is one-fourth the thickness of the vessel wall, measured from the inside.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the change does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the change does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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.

Date: March 1, 1979

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-247

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 49 to Facility Operating License No. DPR-26, issued to Consolidated Edison Company of New York, Inc. (the licensee), which revised Technical Specifications for operation of the Indian Point Nuclear Generating Unit No. 2 (the facility) located in Buchanan, Westchester County, New York . The amendment is effective as of the date of issuance.

The amendment revises the reactor coolant system pressure and temperature heatup and cooldown curves based on data from a material surveillance capsule.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

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The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated January 9, 1979, (2) Amendment No. 49 to License No. DPR-26, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, D.C. and at the White Plains Public Library, 100 Martine Avenue, White Plains, New York. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 1st day of March, 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors