JUN 28 1982

DISTRIBUTION
Docket
NRC PDR
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ORB#1 Rdg
DEisenhut
OELD
IE-2
ACRS-10
RDiggs
DBrinkman
ASLAB
TBarnhart-4
LSchneider
OPA
CParrish
JHannon
JThoma
Gray

Docket No. 50-247

Mr. John D. O'Toole, Vice President
Nuclear Engineering and Quality Assurance
Consolidated Edison Company
of New York, Inc.
4 Irving Place
New York, New York 10003

Dear Mr. O'Toole:

The Commission has issued the enclosed Amendment No. 78 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 May 5, 1982.

The amendment revises the Technical Specifications to modify the reactor coolant system heatup, cooldown, and hydrostatic test pressure/temperature limitations applicable through seven (7) effective full power years (EFPYs) of reactor operation. The present Technical Specifications are applicable only through 5 EFPYs of reactor operation.

The staff safety evaluation provides approval of your proposal with the exception of the 100 Deg F/hr heat up curve. The 100 Deg F/hr heat up curve is not approved pending submittal of further data for staff review.

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

Sincerely,

Original signed by:
S. A. Varga

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

8207150620 820628
PDR ADOCK 05000247
P PDR

Enclosures:
1. Amendment No. 78 to DPR-26
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

*Previous concurrence see next page

OFFICE	ORB#1:DL*	ORB#1:DL*	ORB#1:DL*	AD/OR:DL*	OELD*	MTEB*	
SURNAME	CParrish	JThoma:ds	SVarga	TNovak		WHazelton	
DATE	06/ /82	06/ /82	06/ /82	06/ /82	06/ /82	06/ /82	

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The staff safety evaluation provides approval of your proposal with the exception of the 100 Deg F/hr heat up curve. With the concurrence of your staff, the 100 Deg F/hr heat up curve is not approved pending submittal of further data for staff review.

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

Sincerely,

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

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See next page

OFFICE	ORB#1:DL	ORB#1:DL	ORB#1:DL	AD/OR:DL	OELD	INTEB	
SURNAME	CParrish	JThoma:ds	SVarga	TNovak	McGurren	WHAZELTON	
DATE	06/24/82	06/24/82	06/25/82	06/25/82	06/1?/82	06/25/25	

Notice only

Mr. John D. O'Toole
Consolidated Edison Company of New York, Inc.

cc: White Plains Public Library
100 Martine Avenue
White Plains, New York 10601

Joseph D. Block, Esquire
Executive Vice President
Administrative
Consolidated Edison Company
of New York, Inc.
4 Irving Place
New York, New York 10003

Louis F. Liberatori
Nuclear Licensing Engineer
Consolidated Edison Company
of New York, Inc.
4 Irving Place
New York, New York 10003

Ms. Ellyn Weiss
Sheldon, Harmon and Weiss
1725 I Street, N.W., Suite 506
Washington, D. C. 20006

Dr. Lawrence R. Quarles
Apartment 51
Kendal at Longwood
Kennett Square, Pennsylvania 19348

Theodore A. Rebelowski
Resident Inspector
U. S. Nuclear Regulatory Commission
P. O. Box 38
Buchanan, New York 10511

Brent L. Brandenburg
Assistant General Counsel
Consolidated Edison Company
of New York, Inc.
4 Irving Place - 1822
New York, New York 10003

Carl R. D'Alvia, Esquire
Attorney for the Village of
Buchanan, New York
395 South Riverside Avenue
Croton-on-Hudson, New York 10520

Ronald C. Haynes
Regional Adminstrator - Region I
U. S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, Pennsylvania 19406

Director, Technical Development
Programs
State of New York Energy Office
Agency Building 2
Empire State Plaza
Albany, New York 12223

Regional Radiation Representative
EPA Region II
26 Federal Plaza
New York, New York 10007

Thomas J. Farrelly, Esquire
Law Department
Consolidated Edison Company
of New York, Inc.
4 Irving Place
New York, New York 10003

Joan Holt, Project Director
New York Public Interest
Reserach Group, Inc.
5 Beekman Street
New York, New York 10038

Mr. John D. O'Toole
Consolidated Edison Company of New York, Inc.

cc: Mr. Charles W. Jackson
Vice President, Nuclear Power
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenues
Buchanan, New York 10511

Mr. Kevin Burke
Director, Regulatory Affairs
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenues
Buchanan, New York 10511

Mr. Frank Matra
Resident Construction Manager
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenues
Buchanan, New York 10511

Ezra I. Bialik
Assistant Attorney General
Environmental Protection Bureau
New York State Department of Law
2 World Trade Center
New York, New York 10047

Mayor, Village of Buchanan
188 Westchester Avenue
Buchanan, New York 10511

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. 78
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 May 5, 1982, 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:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 78, 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: June 28, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 78

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
3.1-6	3.1-6
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 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 cool-down 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.

*Pending NRC approval of 100°F/hr heat up curves, reactor coolant system heat up rate shall not exceed 60°F/hr.

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. 78

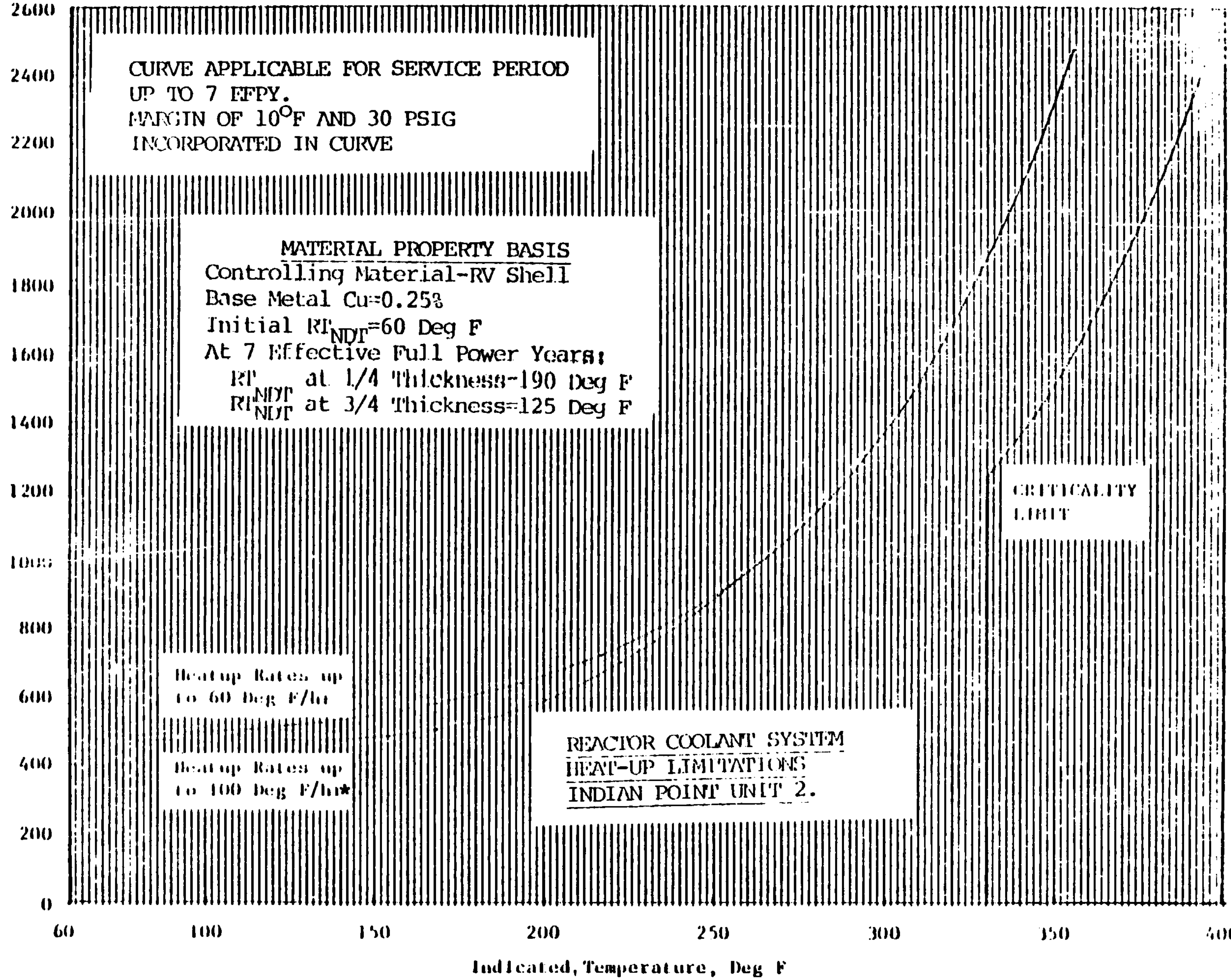

Figure 3.1-1

*Pending NRC approval of 100°F/hr heat up curve, reactor coolant system heat up rate shall not exceed 60°F/hr.

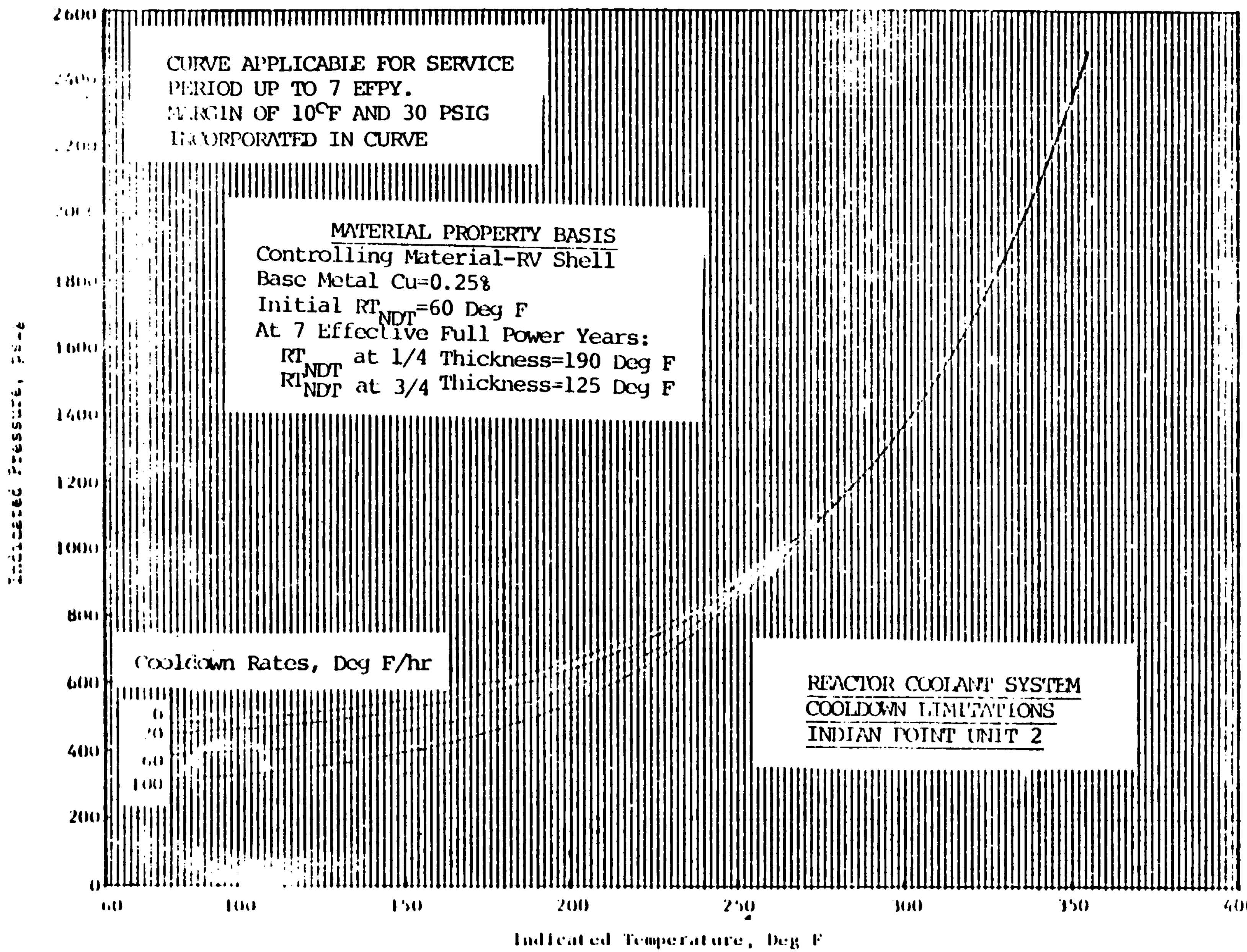
CURVE APPLICABLE FOR SERVICE
PERIOD UP TO 7 EFPY.
MARGIN OF 10°F AND 30 PSIG
INCORPORATED IN CURVE
MATERIAL PROPERTY BASIS
Controlling Material-RV Shell
Base Metal Cu=0.25%
Initial RTNDT=60 Deg F
At 7 Effective Full Power Years:
RTNDT at 1/4 Thickness=190 Deg F
RTNDT at 3/4 Thickness=125 Deg F
Cooldown Rates, Deg F/hr
0
20
60
100
REACTOR COOLANT SYSTEM
COOLDOWN LIMITATIONS
INDIAN POINT UNIT 2
2600
1800
1600
1400
1200
1000
800
600
400
200
0
60
100
150
200
250
300
350
400
Indicated Temperature, Deg F

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^{o}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.

Amendment No. 78

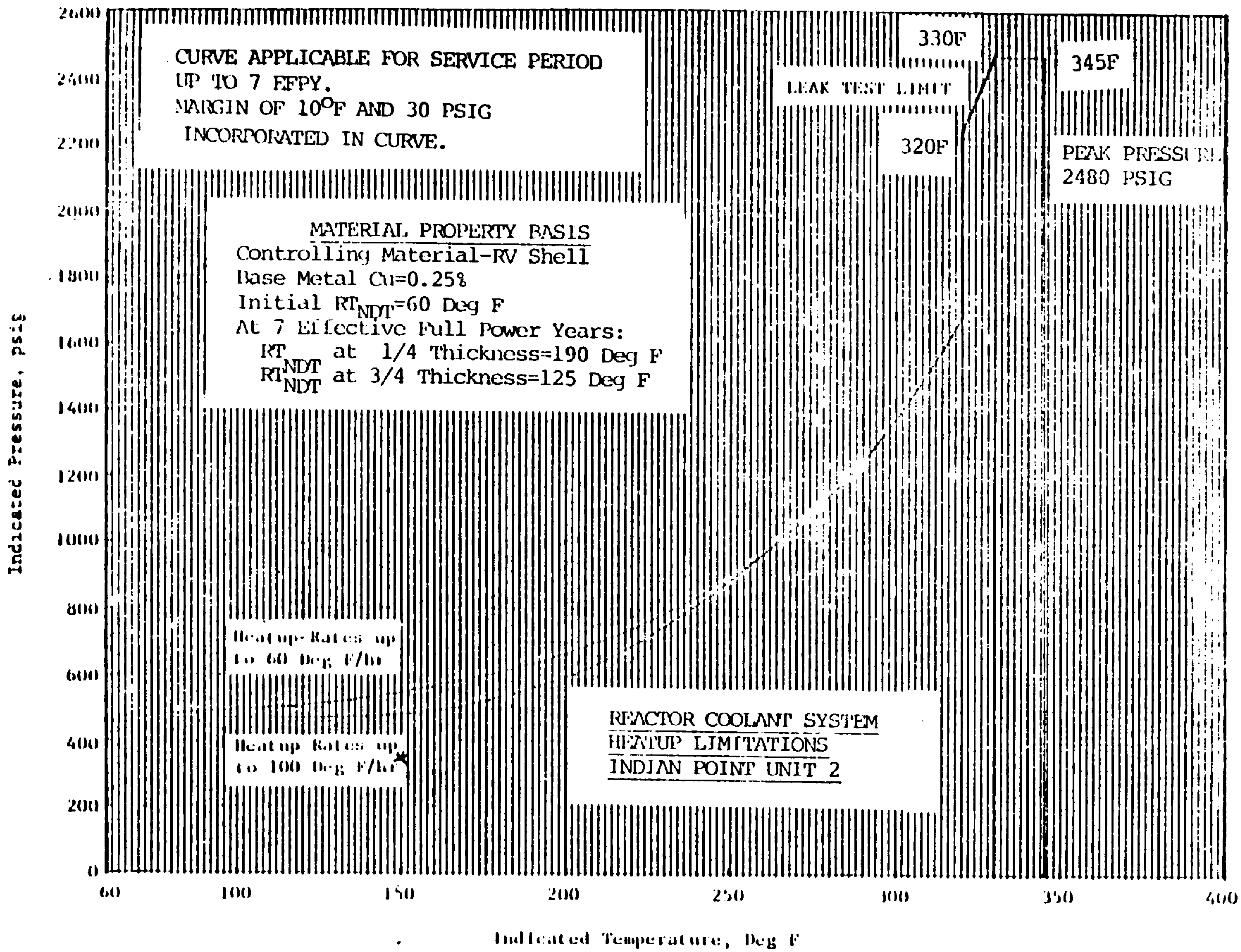

Figure 4.3-1

*Pending NRC approval of 100°F/hr heat up curve, reactor coolant system heat up rate shall not exceed 60°F.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 78 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 May 5, 1982, Consolidated Edison Company (the licensee) requested changes to the Appendix A Technical Specifications appended to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2. The proposed amendment would modify the reactor coolant system pressure-temperature curves for heatup, cooldown, core operations, inservice hydrostatic and inservice leak tests. The revised curves would be applicable for seven effective full power years (EPFY). The proposed curves were based on test results obtained from the "Reactor Vessel Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule Y" test report (SWRI Project No. 02-5212).

Discussion

The basis for review of these curves was Appendix G, 10 CFR Part 50 which references Appendix G, Section III of the ASME Code. We have utilized the test results of the "Reactor Vessel Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule Y" test report to evaluate the proposed curves. We have projected the material embrittlement to seven effective full power years using the trend curves in Regulatory Guide 1.99 Rev. 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials."

Our review indicates that all the limit curves, except for the 100°F/hr. heat up curves in the IP-2 proposed Technical Specifications meet the requirements of our regulations and are acceptable for seven effective full power years (EPFY). Our calculations indicate that the licensee must shift the 100°F/hr heat up curve 29°F to meet our regulatory requirements.

The difference in the 100°F per hour heat up curves calculated by IP-2 and the staff appears to result from a difference in the calculated thermal stress intensity factor and the metal/coolant water temperature relationship. Our evaluation of the information submitted by the licensee indicates that the thermal stress intensity factor and the metal/coolant water temperature relationship used by the applicant in the calculation of the 100°F/hr heat curves are applicable for 60°F/hr heat up, but not 100°F/hr heat up. Accordingly, the applicant's amendment request with respect to the 100°F per hour curve has been deferred and is not approved by this amendment action.

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 amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of an accident of a type different from any evaluated previously, and does not involve a significant reduction in a margin of safety, the amendment 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: June 28, 1982

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. 78 to Facility Operating License No. DPR-26 issued to the 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 Technical Specifications to modify the reactor coolant system heatup, cooldown, and hydrostatic test pressure/temperature limitations applicable through seven (7) effective full power years of reactor operation. The maximum heatup rate will be limited to 60 Deg F/hr while further review is being conducted on the proposed 100 Deg F/hr limitations.

The application for 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 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.

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 futher details with respect to this action, see (1) the application for amendment dated May 5, 1982, (2) Amendment No. 78 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, N.W., 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 Licensing.

Dated at Bethesda, Maryland, this 28th day of June, 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

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