DISTRIBUTION:
Docket
NRC PDR
Local PDR
ORB#4 Rdg.
VStello
KRGoller/TJCarter
RIngram
PErickson
Attorney, OELD
OI&E (5)
BJones (4)
BScharf (15)
JMMcGough
BHarless
DEisenhut
ACRS (16)
OPA, Clare Miles
DRoss
TBAbernathy
JRBuchanan
Gray File
Xtra Cy File - 4

February 18, 1977

Docket No.: 50-247

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

Gentlemen:

The Commission has issued the enclosed Amendment No. 28 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 April 26, 1976.

The amendment revises the Technical Specifications to change requirements for the Indian Point Unit No. 2 pressure-temperature limits during reactor heatup and cooldown.

Copies of the Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

Original Signed by

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosures:
1. Amendment No. 28 to License No. DPR-26
2. Safety Evaluation
3. Federal Register Notice

cc w/enclosures:
See next page

subject to additions to p. 1 of FR notice

247
Const. 1

8111060222 770218
PDR ADOCK 05000247
P PDR

ORB#4:DOR	ORB#4:DOR	OELD	C-ORB#4:DOR		
RIngram	PErickson:dr	S H Lewis	RWReid		
2/18/77	2/18/77	2/18/77	2/18/77		

Consolidated Edison Company
of New York, Inc.

cc w/enclosure(s):
Mrs. Kay Winter, Librarian
Hendrick Hudson Free Library
31 Albany Post Road
Montrose, New York 10548

Leonard M. Trosten, Esq.
LeBoeuf, Lamb, Leiby & MacRae
1757 N Street, N.W.
Washington, D.C. 20036

Anthony Z. Roisman, Esq.
Berlin, Roisman & Kessler
1025 15th Street, N.W., 5th Floor
Washington, D.C. 20005

Paul S. Shemin, Esq.
Assistant Attorney General
State of New York
Department of Law
Two World Trade Center
New York, New York 10047

Sarah Chasis, Esq.
Richard M. Hall, Esq.
Natural Resources Defense
Council
15 West 44th Street
New York, New York 10036

Carl R. D'Alvia, Esq.
Box 306
Croton-on-Hudson, New York 10520

Michael Curley, Esq.
Office of the General Counsel
New York State Atomic Energy Council
New York State Department of Commerce
99 Washington Avenue
Albany, New York 12210

Honorable George Begany
Mayor, Village of Buchanan
188 Westchester Avenue
Buchanan, New York 10511

Director, Technical Development
Programs
State of New York
Energy Office
Swan Street Building
CORE 1 - Second Floor
Empire State Plaza
Albany, New York 12223

Chief, Energy Systems
Analyses Branch (AW-459)
Office of Radiation Programs
U. S. Environmental Protection Agency
Room 645, East Tower
401 M Street, S.W.
Washington, D.C. 20460

U. S. Environmental Protection
Agency
Region II Office
ATTN: EIS COORDINATOR
26 Federal Plaza
New York, New York 10007

Edward J. Sack, Esq.
Law Department
Consolidated Edison Company
of New York, Inc.
4 Irving Place
New York, New York 10003

Richard C. King, Esq.
New York State Energy Office
Swan Street Building
Core 1, 2nd Floor
Empire State Plaza
Albany, New York 12223

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. 28
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) sworn to April 22, 1976, 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 DPR-26 is hereby amended to read as follows:

8111060228 770218
PDR ADOCK 05000247
P PDR

(2) Technical Specifications

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

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 18, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 28

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Pages
iv	iv
v	v
3.1-4	3.1-4
3.1-5	3.1-5
3.1-6	3.1-6
3.1-7	3.1-7
3.1-8	3.1-8(a)
-	3.1-8(b)
-	3.1-8(c)
-	3.1-8(d)
-	Table 3.1-1
3.1-9	3.1-9
3.1-10	3.1-10
Fig. 3.1-1	Fig. 3.1-1
Fig. 3.1-2	Fig. 3.1-2
4.3-1	4.3-1
4.3-2	4.3-2
-	Fig. 4.3-1

Changes on the revised pages are shown by marginal lines.

LIST OF TABLES

IP2 Reactor Vessel Core Region Material	3.1-1
Engineered Safety Features Initiation Instrument Setting Limits	3-1
Reactor Trip Instrumentation Limiting Operating Conditions	3-2
Instrumentation Operating Condition for Engineered Safety Features	3-3
Instrument Operating Conditions for Isolation Functions	3-4
Safety Related Shock Suppressors (Snubbers)	3.12-1
Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channel	4.1-1
Frequencies for Sampling Tests	4.1-2
Frequencies for Equipment Tests	4.1-3
Inservice Inspection Requirements for Indian Point No. 2	4.2-1
Minimum Shift Crew Composition	6.2-1
Protection Factors for Respirators	6.12-1

Amendment No. ~~21~~, 28

LIST OF FIGURES

Safety Limits Four Loop Operation 100% Flow	2.1-1
Safety Limits Three Loop Operation 73% Flow	2.1-2
Reactor Coolant System Heatup Limitations	3.1-1
Reactor Coolant System Cooldown Limitations	3.1-2
Required Hot Shutdown Margin vs Reactor Coolant Boron Concentration	3.10-1
Hot Channel Factor Normalized Operating Envelope	3.10-2
Insertion Limits, 100 Step Overlap Four Loop Operation	3.10-3
Insertion Limits, 100 Step Overlap Three Loop Operation	3.10-4
Target Band on Indicated Flux Difference as a Function of Operating Power Level	3.10-5
Permissible Operating Band on Indicated Flux Difference as a Function of Burnup	3.10-6
Reactor Coolant System Heatup Limitation	4.3-1
Facility Management and Technical Support Organization	6.2-1
Facility Organization	6.2-2

Amendment No. ~~20~~, 28

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 3 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 recalculated periodically using methods discussed in WCAP-7924A and results of surveillance specimen testing as covered in WCAP-7323.(7)

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. 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].

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} at the end of three years of service life.

The three-year service life period is chosen such that the limiting RT_{NDT} at the 1/4 T location in the core region is higher than the RT_{NDT} of the limiting unirradiated material. This service period assures that all components in the Reactor Coolant System will be operated conservatively in accordance with Code recommendations.

The highest RT_{NDT} of the core region material is determined by adding the radiation induced ΔRT_{NDT} for the applicable time period to the original RT_{NDT} shown in Table 3.1-1. An approximation of the fast neutron (E >1 Mev) fluence at 1/4 thickness and 3/4 thickness vessel locations is given as a function of full-power service life in Figure 2-4 of WCAP-7924A[4]. Exposure to the Indian Point Unit No. 2 vessel will be less than that indicated by that figure. Using the applicable fluence at the end of the three-year period and the copper content of the material in question, the ΔRT_{NDT} may be obtained from Figure 2-3 of WCAP-7924A[4].

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, when evaluated according to ASTM E185[6], are available. The first capsule will be removed early in the service life of the reactor vessel, note FSAR Section 4.5.1. The heatup and cooldown curves will be re-evaluated if the ΔRT_{NDT} determined from the surveillance capsule is different from the predicted ΔRT_{NDT}.

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 on 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 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 E185-70, 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.

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)	Assume 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

C. MINIMUM CONDITIONS FOR CRITICALITY

1. Except during low power physics tests, the reactor shall not be made critical at any temperature above which the moderator temperature coefficient is positive.

2. In no case shall the reactor be made critical below the temperature and pressure limits shown in Figure 3.1-1.

3. When the reactor coolant temperature is below the minimum temperature specified in 1. above, the reactor shall be subcritical by an amount greater than the potential reactivity insertion due to depressurization.

4. The reactor shall be maintained subcritical by at least 1% until normal water level is established in the pressurizer.

Basis:

During the early part of the initial fuel cycle, the moderator temperature coefficient is calculated to be slightly positive at coolant temperatures below the power operating range.[(1)(2)] The moderator coefficient at low temperatures will be most positive at the beginning of life of the fuel cycle, when the boron concentration in the coolant is the greatest. Later in the life of the fuel cycle, the boron concentrations in the coolant will be lower and the moderator coefficients will be either less positive or will be negative. At all times, the moderator coefficient is negative in the power operating range.[(1)(2)] Suitable physics measurements of moderator coefficients of reactivity will be made as part of the startup program to verify analytic predictions.

The requirement that the reactor is not to be made critical when the moderator coefficient is positive has been imposed to prevent any unexpected power excursion during normal operations as a result of either an increase of moderator temperature or decrease of coolant pressure. This requirement is waived during low power physics tests to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special operating precautions will be taken.

The requirement that the reactor is not to be made critical below the temperature and pressure limits shown in Figure 3.1-1 provides increased assurance that the proper relationship between reactor coolant pressure and temperature will be maintained during system heatup and pressurization in accordance with the requirements of 10CFR50 Appendix G, as amended February 2, 1976. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.

If the shutdown margin specified in 3.1.C-3 is maintained, there is no possibility of an accidental criticality as a result of a decrease of coolant pressure.

The requirement for bubble formation in the pressurizer when the reactor has passed the threshold of 1% subcriticality will assure that the Reactor Coolant System will not be solid when criticality is achieved.

References

1. FSAR Table 3.2.1-1

2. FSAR Figure 3.2.1-9

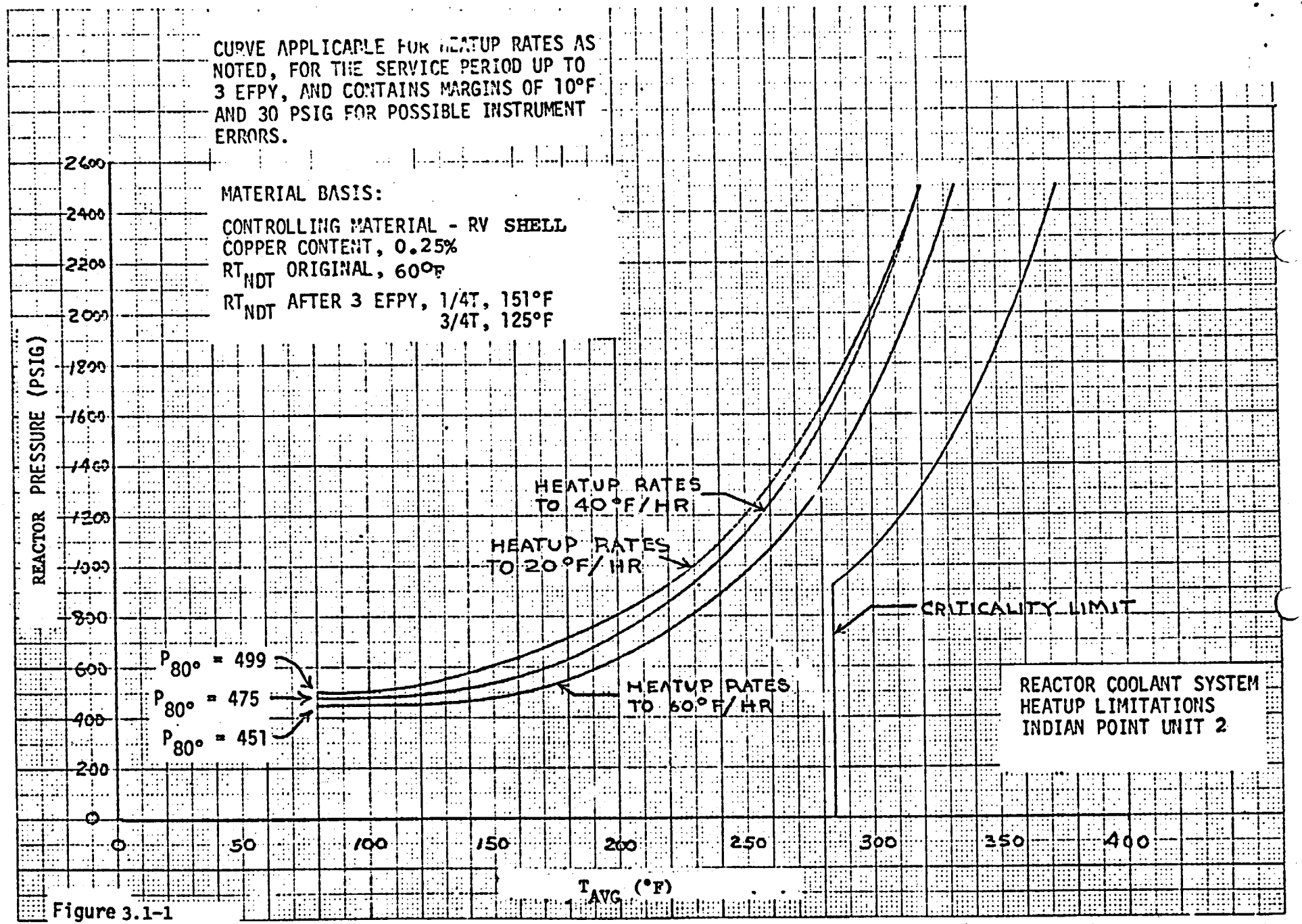
CURVE APPLICABLE FOR HEATUP RATES AS NOTED, FOR THE SERVICE PERIOD UP TO 3 EFPY, AND CONTAINS MARGINS OF 10°F AND 30 PSIG FOR POSSIBLE INSTRUMENT ERRORS.
MATERIAL BASIS:
CONTROLLING MATERIAL - RV SHELL
COPPER CONTENT, 0.25%
RTNDT ORIGINAL, 60°F
RTNDT AFTER 3 EFPY, 1/4T, 151°F
3/4T, 125°F
REACTOR PRESSURE (PSIG)
2600
2400
2200
2000
1800
1600
1400
1200
1000
800
600
400
200
0
HEATUP RATES TO 40°F/HR
HEATUP RATES TO 20°F/HR
HEATUP RATES TO 60°F/HR
CRITICALITY LIMIT
P80° = 499
P80° = 475
P80° = 451
REACTOR COOLANT SYSTEM
HEATUP LIMITATIONS
INDIAN POINT UNIT 2
0
50
100
150
200
250
300
350
400
TAVG (°F)

Figure 3.1-1

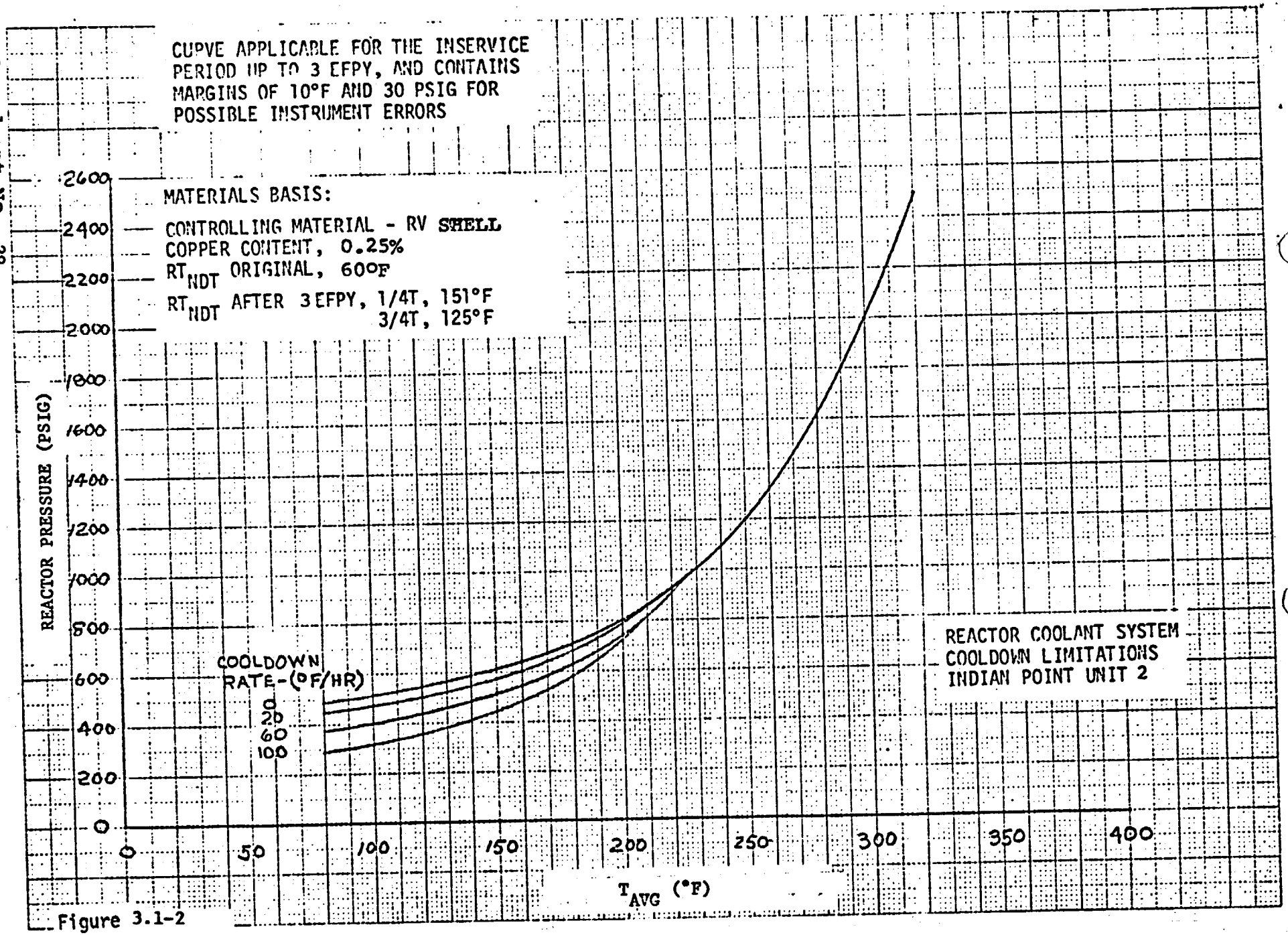
CURVE APPLICABLE FOR THE INSERVICE PERIOD UP TO 3 EFPY, AND CONTAINS MARGINS OF 10°F AND 30 PSIG FOR POSSIBLE INSTRUMENT ERRORS
MATERIALS BASIS:
CONTROLLING MATERIAL - RV SHELL
COPPER CONTENT, 0.25%
RTNDT ORIGINAL, 60°F
RTNDT AFTER 3 EFPY, 1/4T, 151°F
3/4T, 125°F
REACTOR PRESSURE (PSIG)
2600
2400
2200
2000
1800
1600
1400
1200
1000
800
600
400
200
0
COOLDOWN RATE-(°F/HR)
0
20
60
100
0
50
100
150
200
250
300
350
400
TAVG (°F)
REACTOR COOLANT SYSTEM
COOLDOWN LIMITATIONS
INDIAN POINT UNIT 2

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 heatup for the first three effective full-power yrs. of operation. Figure 4.3-1 will be recalculated periodically. Allowable pressures during cooldown from 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 three effective full-power years, it is predicted that the highest RT_{NDT} in the core region taken at the 1/4 thickness will be 151°F. The minimum inservice leak test temperature requirements for periods up to three 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.

<u>Reference</u>

1. FSAR, Section 4.

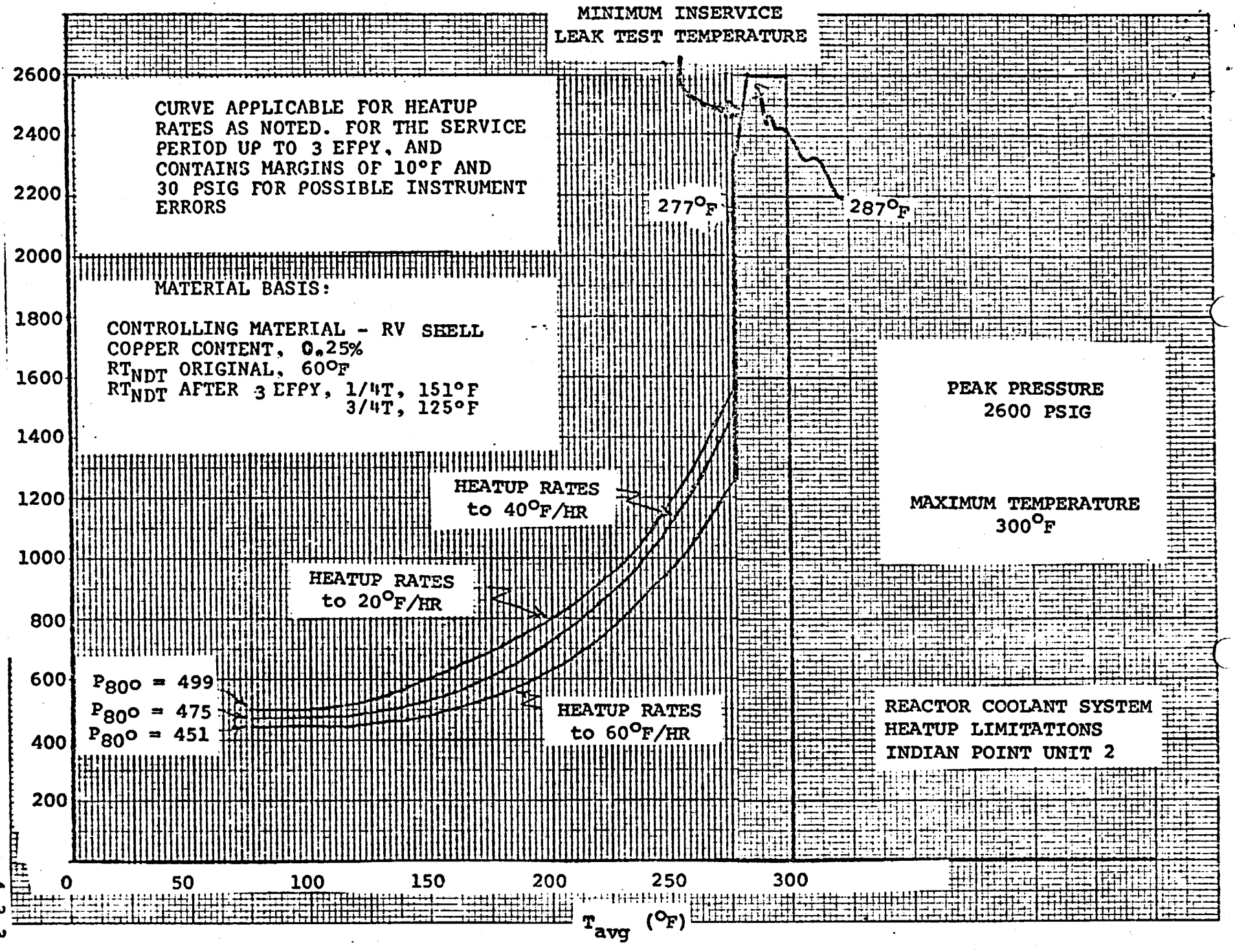

FIGURE 4.3-1 PRESSURE/TEMPERATURE LIMITATIONS FOR HYDROSTATIC LEAK TEST

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 28 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 April 26, 1976, Consolidated Edison Company of New York, Inc. (Con Ed) requested changes to the Technical Specifications appended to Facility Operating License No. DPR-26 for Indian Point Unit No. 2. The requested changes would modify the reactor coolant system pressure-temperature limits to account for neutron irradiation induced increases in reactor vessel metal nil ductility temperature (RT_{NDT})[1].

Discussion

Title 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 specific pressure-temperature limits which are initially established depend upon the metallurgical properties of the reactor vessel material and the design service conditions. However, the metallurgical properties vary over the lifetime of the reactor vessel because of the effects of neutron irradiation. One principal effect of the neutron irradiation is that it causes the reactor vessel nil ductility temperature (RT_{NDT}) to increase or shift with time. The practical results of the RT_{NDT} shift is that, for any

[1] RT_{NDT} is the temperature associated with the transition from a ductile to a brittle fracture mode of failure of a metal test specimen.

8111060229\770218
PDR ADOCK 05000247
P PDR

given value of reactor pressure, the reactor vessel metal temperature must be maintained at higher values during the heatup and cooldown process. By periodically revising the pressure-temperature limits to account for neutron irradiation induced increases in RT_{NDT}, the stresses in the reactor vessel are maintained within acceptable limits.

The magnitude of the shift in RT_{NDT} is proportional to the integrated amount of neutron irradiation experienced by the reactor vessel. In addition a reactor vessel material surveillance program has been previously established to check the validity of the predicted increases in RT_{NDT}. Surveillance specimens are periodically removed from the reactor vessel for testing and analysis. The results of the tests and analysis are compared with the predicted shifts in RT_{NDT}; then the pressure-temperature limits are revised for future operation as required.

Evaluation

The current reactor coolant system pressure-temperature limits were established to cover periods of reactor operation up to about 2 EFPY (Effective Full Power Years) for Indian Point Unit No. 2. The proposed revised limits would allow reactor operation up to 3 EFPY for Unit No. 2. Con Ed, in their letter dated April 26, 1976, requested an amendment to Technical Specifications 3.1 and 4.3 to conform the pressure-temperature operating limits with Appendix G, 10 CFR 50. The curves are based on the most limiting value of RT_{NDT} at the end of 3 EFPY. The most limiting material is the RV shell plate B 2002-1 that has an assumed initial RT_{NDT} of 60°F and a copper content of 0.25%. After 3 EFPY the RT_{NDT} at 1/4T is calculated to be 151°F. The operating limit curves were derived using the methods outlined in Appendix G to ASME Code, Section III and WCAP-7924A. We have reviewed these operating limit curves and conclude that they conform to Appendix G requirements. The proposed Technical Specifications 3.1.B, "Heatup and Cooldown," 3.1.C, "Minimum Conditions for Criticality," and 4.3.C, "Reactor Coolant System Integrity Testing," are acceptable for 3 EFPY.

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) than 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 accidents previously considered and does not involve a significant decrease in a safety margin, 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.

Dated: February 18, 1977

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. 28 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 its date of issuance.

The amendment changes the Technical Specifications to allow operation of Indian Point Unit No. 2 with revised pressure-temperature limits during reactor heatup and cooldown. The revised limits allow operation up to three effective full power years.

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.

8111060233 770218
PDR ADOCK 05000247
P PDR

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 transmitted by letter dated April 26, 1976, (2) Amendment No. 28 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 Hendrick Hudson Free Library, 31 Albany Post Road, Montrose, 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 18th day of February 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors