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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.



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## 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 $\Delta RT_{NDT}$ 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 $\Delta RT_{NDT}$ may be obtained from Figure 2-3 of WCAP-7924A[4].

Values of $\Delta RT_{NDT}$ 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 $\Delta RT_{NDT}$ determined from the surveillance capsule is different from the predicted $\Delta RT_{NDT}$.

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## 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 $\Delta T$ 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 compatability 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 $RT_{NDT}$ ( |
|---|---|---|---|---|
| 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

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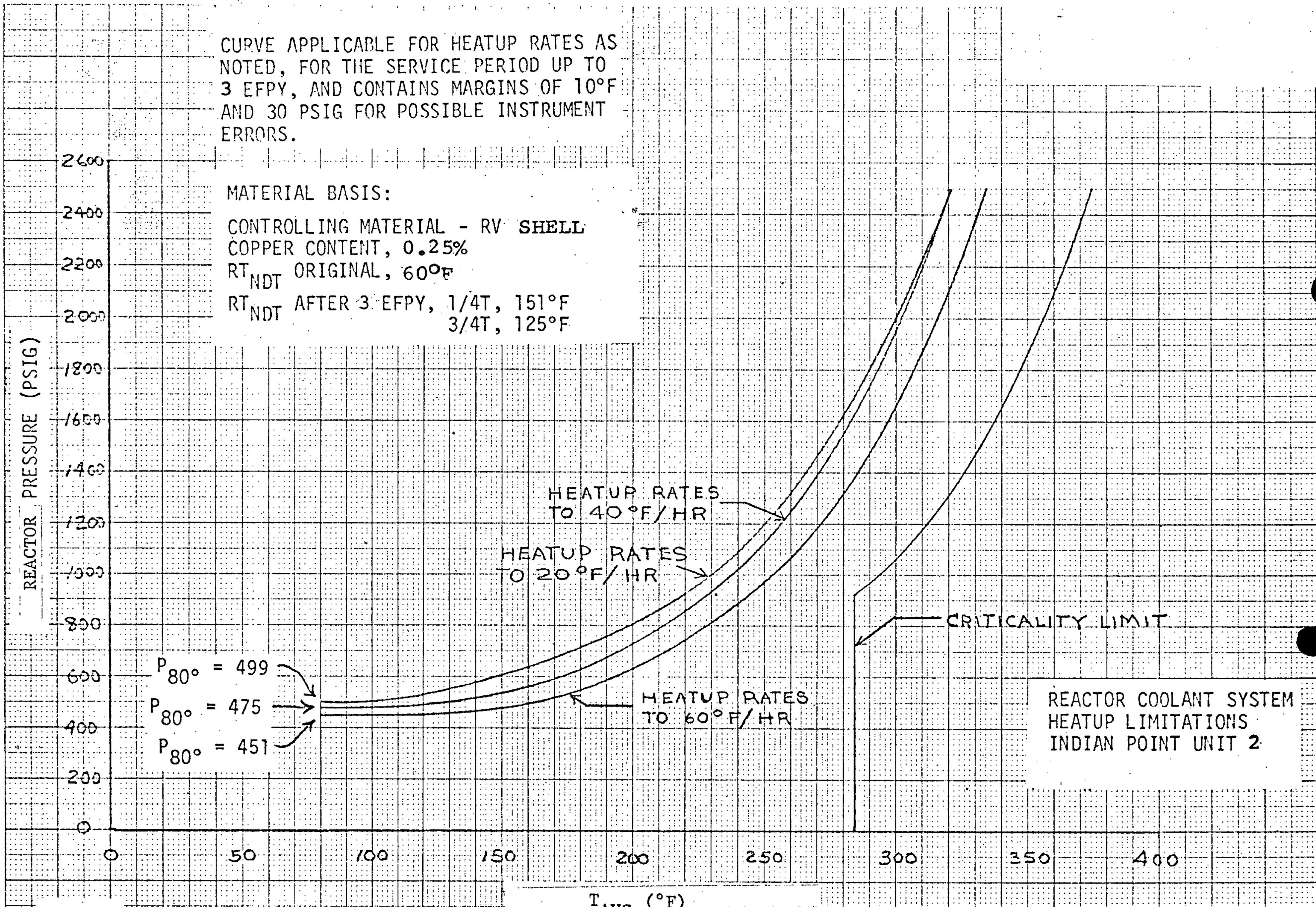


Figure 3.1-1

Amendment No.

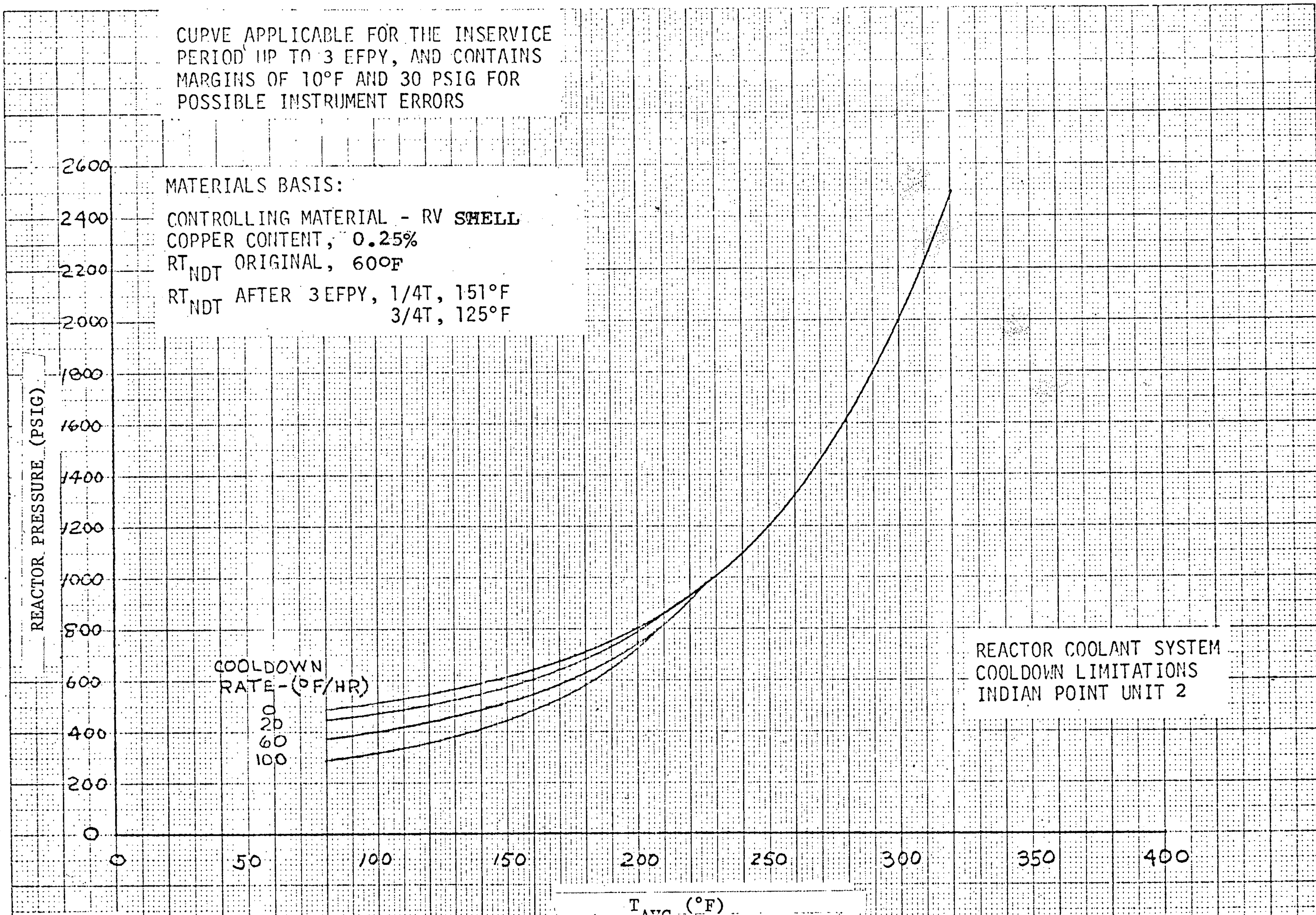


Figure 3.1-2

## 4.3 REACTOR COOLANT SYSTEM INTEGRITY TESTING

### Applicability

Applies to test requirements for Reactor Coolant System integrity.

### Objective

To specify tests for Reactor Coolant System integrity after the system is closed following normal opening, modification or repair.

### Specification

a) When the Reactor Coolant System is closed after it has been opened, the system will be leak tested at not less than 2335 psig at NDT requirements for temperature.

b) When Reactor Coolant System modifications or repairs have been made which involve new strength welds on components, the new welds 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: $\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 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.

Reference

1. FSAR, Section 4.

Amendment No.

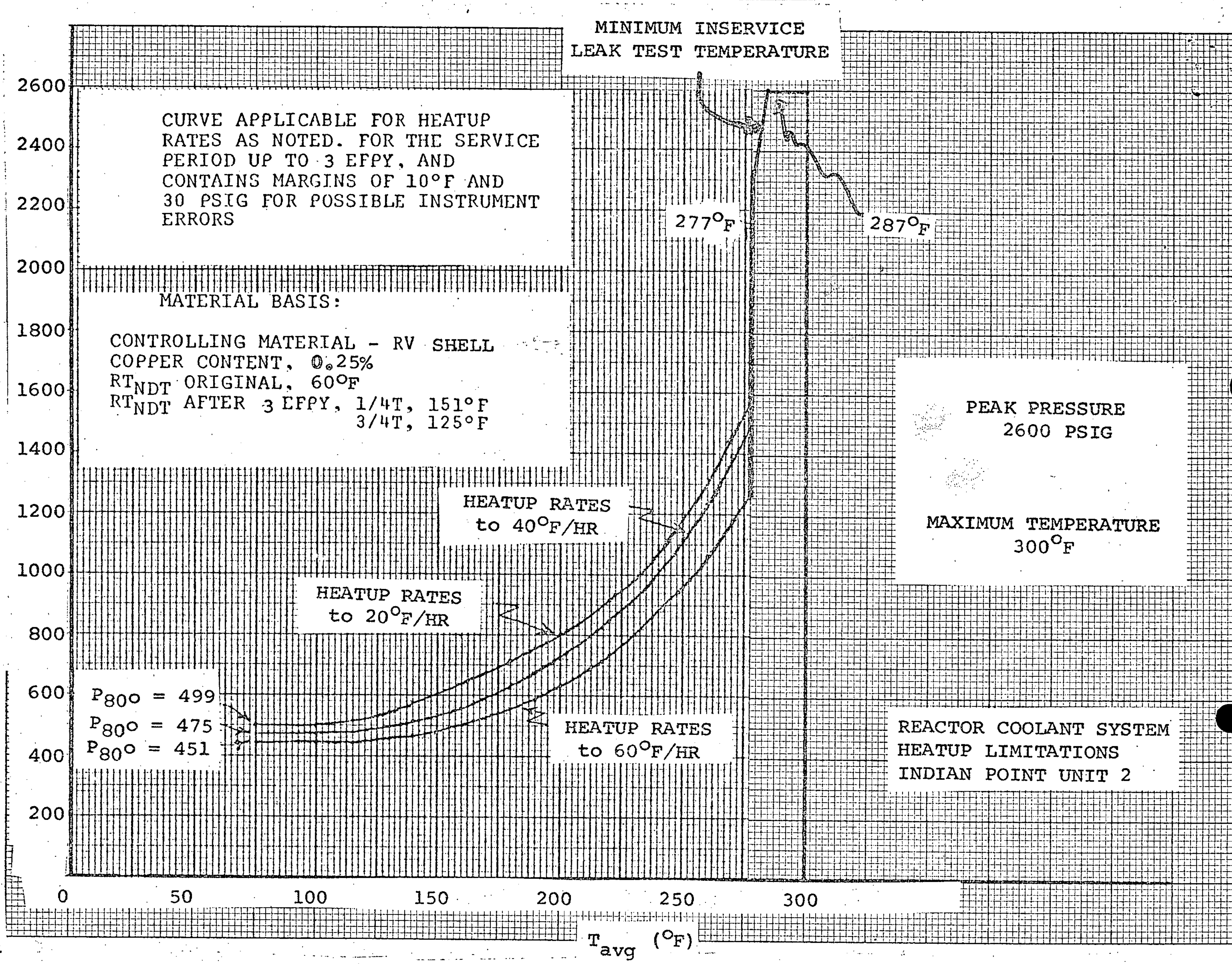


FIGURE 4.3-1 PRESSURE/TEMPERATURE LIMITATIONS FOR HYDROSTATIC LEAK TEST

ATTACHMENT B

APPLICATION FOR AMENDMENT TO
OPERATING LICENSE

SAFETY EVALUATION

Consolidated Edison Company of New York, Inc.

Indian Point Unit No. 2
Docket No. 50-247

Facility Operating License No. DPR-26
April 22, 1976

# Safety Evaluation

This Application for Amendment is submitted in response to a letter dated February 17, 1976 from Mr. Robert W. Reid to Mr. William J. Cahill, Jr. In that letter, the NRC requested that Con Edison review the reactor coolant system pressure-temperature limits contained in the Indian Point Unit No. 2 Technical Specifications. The conformance of these specifications with the fracture toughness requirements specified in 10 CFR Part 50, Appendix G was to be determined. The review found that some modifications were necessary to bring the Technical Specifications into conformance with the Appendix G requirements.

Accordingly, the proposed changes to Sections 3.1 and 4.3 of the Technical Specifications, contained in Attachment A to this Application would conform to the Indian Point Unit No. 2 reactor coolant system pressure-temperature limits with the requirements of 10CFR Part 50, Appendix G. These proposed Technical Specifications have been modeled after the corresponding sections contained in the recently-finalized Indian Point Unit No. 3 Technical Specifications. There are no facility modifications required as a result of the proposed change.

The proposed changes do not in any way alter the safety analyses performed for Indian Point Unit No. 2. The proposed changes have been reviewed by the Station Nuclear Safety Committee and by the Consolidated Edison Nuclear Facilities Safety Committee. Both committees concur that these changes do not represent a significant hazards consideration and will not cause any change in the authorized power level of the facility.

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BEFORE THE UNITED STATES
NUCLEAR REGULATORY COMMISSION

| | | |
|---|---|---|
| In the Matter of | ) | |
| | ) | |
| CONSOLIDATED EDISON COMPANY OF NEW YORK, INC. | ) | Docket No. 50-247 |
| (Indian Point Station, Unit No. 2) | ) | |

CERTIFICATE OF SERVICE

I certify that I have this 26th day of April, 1976, served the foregoing document entitled "Application for Amendment of Operating License" sworn to on April 22, 1976, including Attachments A and B thereto, by mailing copies thereof, first-class postage prepaid and properly addressed, to the following persons:

Hon. George V. Begany
Mayor, Village of Buchanan
Buchanan, New York 10511

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31 Albany Post Road
Montrose, New York 10548

Eugene R. Fidell
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