ATTACHMENT I TO IPN-90-046

PROPOSED TECHNICAL SPECIFICATION CHANGES

RELATED TO

PRESSURE-TEMPERATURE LIMITS

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

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 9.00 effective full-power years (EFPYs). The heatup and cooldown rates shall not exceed 60°F/hr and 100°F/hr respectively.

   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 the Basis and results of surveillance specimens as covered in Specification 4.2. The order of specimen removal may be modified based on the results of testing of previously removed specimens.

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.

Basis

Fracture Toughness Properties

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the Summer 1965 Section III of the ASME Boiler and Pressure Vessel Code [6] and ASTM E185 [5] and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1972 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code [1], and the calculation methods described in WCAP-7924 [2].

The first reactor vessel material surveillance capsule was removed during the 1978 refueling outage. This capsule has been tested by Westinghouse Corporation and the results have been evaluated and reported [7]. Based on the Westinghouse evaluation, heatup and cooldown curves (Figures 3.1-1 and 3.1-2) were developed for up to 9.26 EFPYs of reactor operation.

Generic Letter 88-11 requested that licensees use the methodology of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials", to predict the effect of neutron radiation on reactor vessel materials as required by paragraph V.A. of 10 CFR part 50, Appendix G. Capsule Z was analyzed [8] and new pressure-temperature curves were developed using this methodology.

The maximum shift in RT_{NDT} after 9.00 EFPYs of operation is projected to be 194°F at the 1/4 T and 157°F at the 3/4 T vessel wall locations for Plate B2803-3,the controlling plate. Plate B2803-3 was also the controlling plate for the first operating period of 2 EFPYs.

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} at the end of 9.00 years of service life. The 9.00 year service life period is chosen such at 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 Q4.2-1 [3].

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 of the ASME Boiler and Pressure Vessel Code and discussed in detail in WCAP-7924 [2].

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 [1] 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:

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

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 time, 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-7924 [2].

Pressure 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. ASME Boiler and Pressure Vessel Code, Section III, 1972 Summer Addenda.

2. WCAP-7924, "Basis for Heatup and Cooldown Limit Curves", W. S. Hazelton, S. L. Anderson, S. E. Yanichko, July 1972.

3. FSAR Volume 5, Response to Question Q4.2.

4. Intentionally deleted.

5. ASTM E185-70, Surveillance Tests on Structural Materials in Nuclear Reactors.

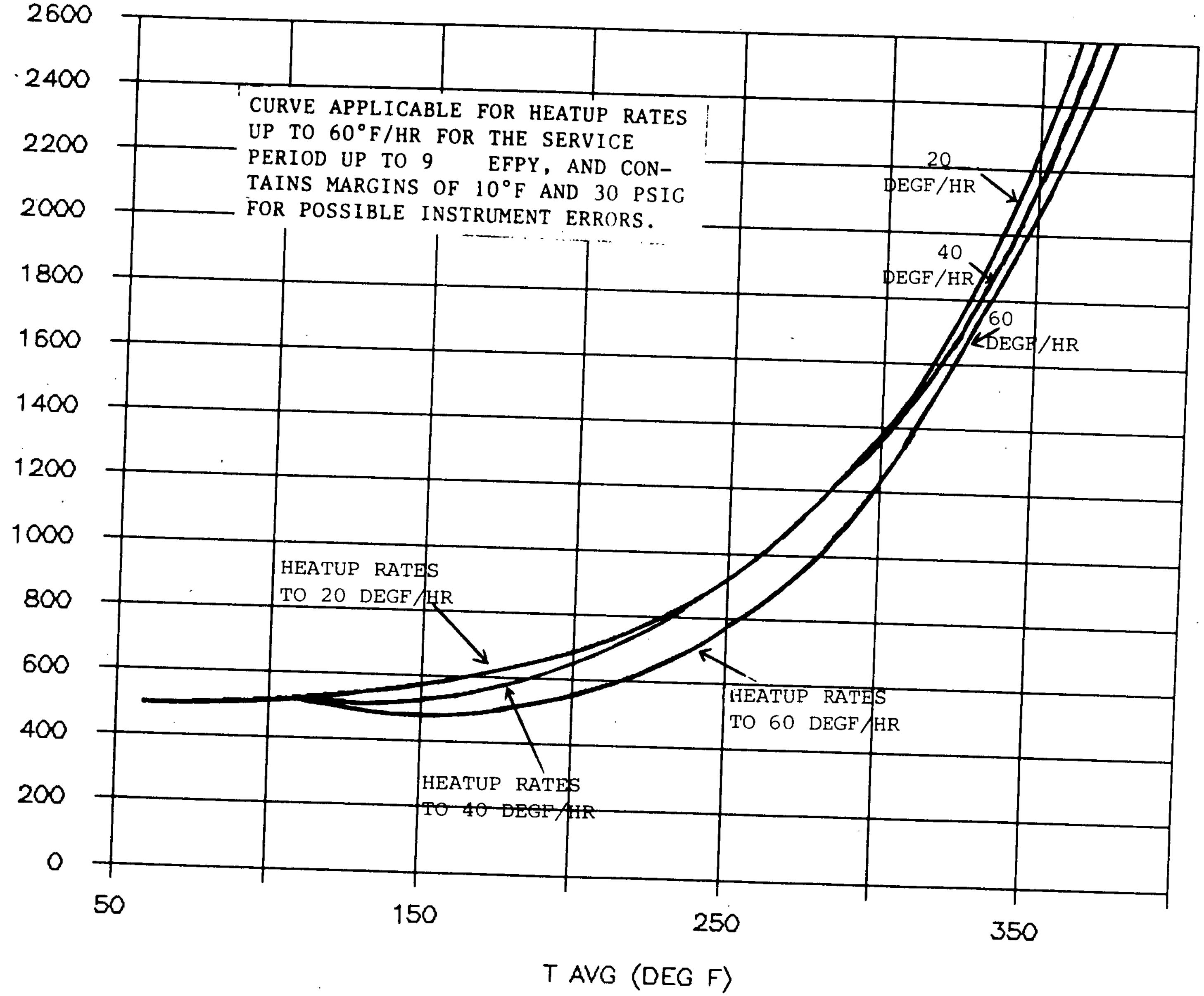
6. ASME Boiler and Pressure Vessel Code, Section III, Summer 1965.

7. WCAP-9491, "Analysis of Capsule T from the Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program", J. A. Davidson, S. L. Anderson, W. T. Kaiser, April 1979.

8. WCAP-11815, "Analysis of Capsule Z from the New York Power Authority Indian Point Unit 3 Reactor Vessel Radiation Surveillance Program, S. E. Yanichko, S. L. Anderson, L. Albertin, March 1988.

FIGURE 3.1-1

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS - INDIAN POINT 3



MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: PLATE METAL

COPPER CONTENT: 0.24 WT%

PHOSPHORUS CONTENT: 0.012 WT%

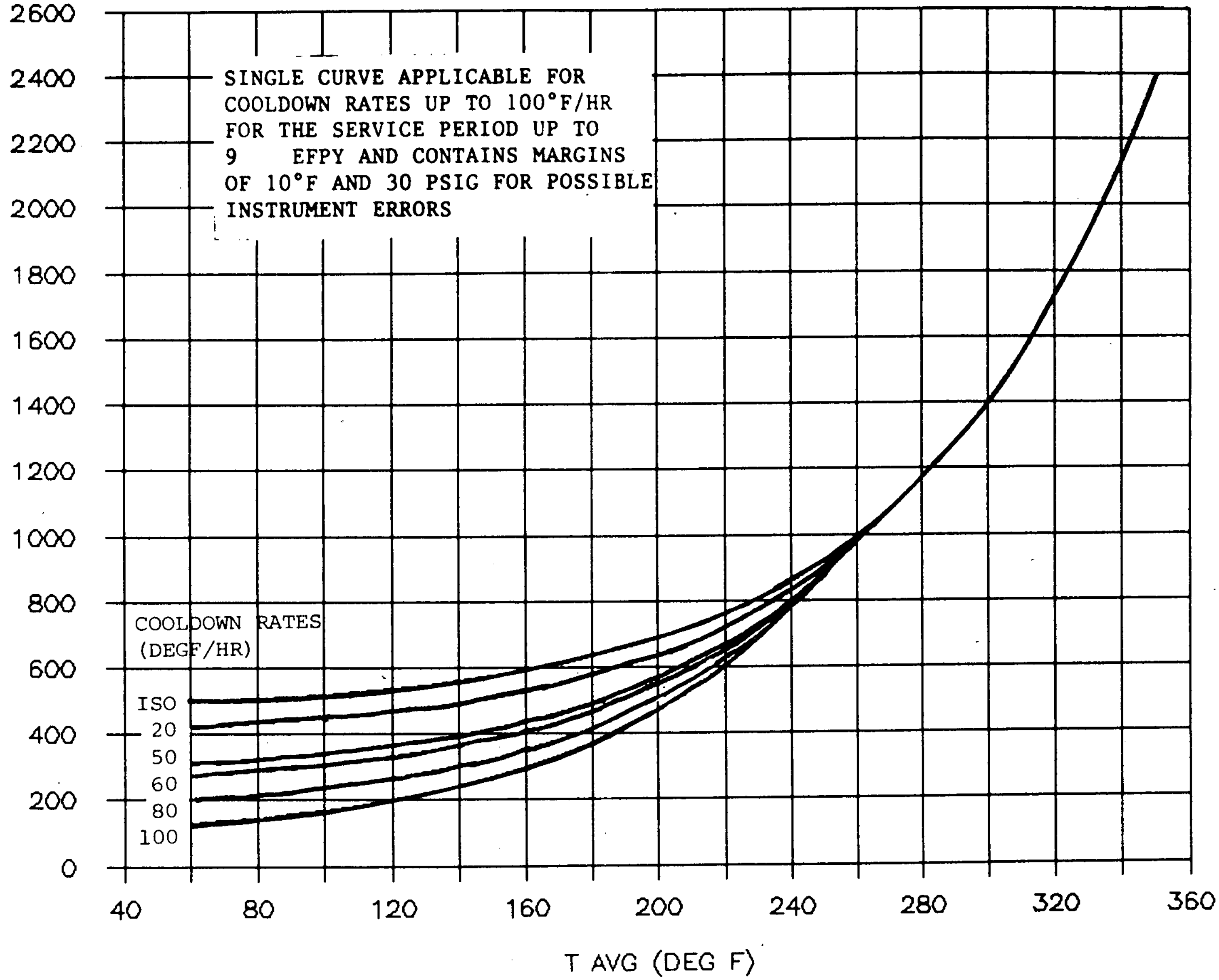
RT(NDT) INITIAL: 74 DEG F

RT(NDT) AFTER 9 EFPY: 1/4T = 194 DEG F

3/4T = 157 DEG F

FIGURE 3.1-2

REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS - INDIAN POINT 3



MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: PLATE METAL

COPPER CONTENT: 0.24 WT%

PHOSPHORUS CONTENT: 0.012 WT%

RT(NDT) INITIAL: 74 DEG F

RT(NDT) AFTER 9 EFPY: 1/4T = 194 DEG F

3/4T = 157 DEG F

C. MINIMUM CONDITIONS FOR CRITICALITY

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

2. This section intentionally deleted.

3. At all times during critical operation, T_{avg} should be no lower than 450°F.

4. The reactor shall be maintained subcritical by at least 1% $\frac{\Delta k}{k}$ 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 concentration in the coolant will be lower and the moderator coefficient 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 coefficient 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 an increase in moderator temperature. 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 except when T_{avg} is $\geq$ 450°F provides increased assurance that an overpressure event will not occur whenever the reactor vessel is in the nil-ductility temperature range. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.

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

References:

1. FSAR Table 3.2.1-1

2. FSAR Figure 3.2.1-9

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 and in accordance with 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.

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 9.00 EFPYs 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 test temperatures are shown on Figure 4.3-1. The temperatures are calculated in accordance with ASME Code Section III, 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 9.00 effective full power years, it is predicted that the highest RT_{NDT} in the core region taken at the 1/4 thickness will be 194°F. The temperature determined by methods of ASME Code Section III for 2335 psig is 135°F above this RT_{NDT} and for 2510 psig (maximum) is 145°F above this RT_{NDT}. The minimum inservice leak test temperature requirements for periods up to nine effective full power years are shown on Figure 4.3-1.

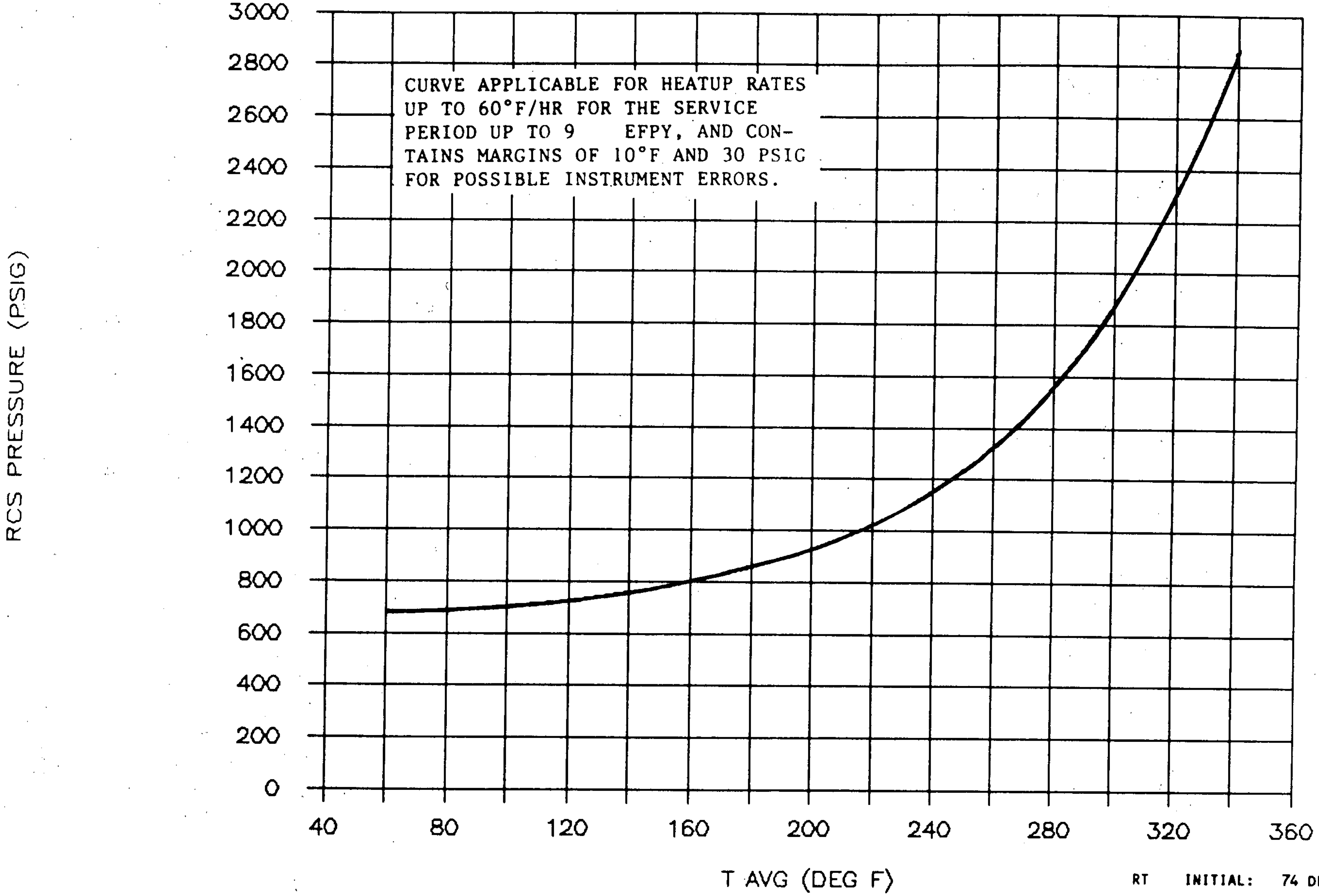
The heatup limits specified on the heatup curve, Figure 4.3-1, must not be exceeded while the reactor coolant system is being heated to the inservice leak test temperature. For cooldown from the leak test temperature, the limitations of Figure 3.1-2 must not be exceeded. Figures 4.3-1 and 3.1-2 are recalculated periodically, using methods discussed in the Basis for Specification 3.1.B and results of surveillance specimens, as covered in Specification 4.2.

<u>Reference</u>

1. FSAR, Section 4

FIGURE 4.3-1, PRESSURE/TEMPERATURE

LIMITATIONS FOR HYDROSTATIC LEAK TEST



RT_{NDT} INITIAL: 74 DEG F

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: PLATE METAL

COPPER CONTENT: 0.24 WT%

PHOSPHORUS CONTENT: 0.012 WT%

RT_{NDT} AFTER 9 EFPY: 1/4T = 194 DEG F

3/4T = 157 DEG F

Amendment No. ~~28~~

ATTACHMENT II TO IPN-90-046

SAFETY EVALUATION

RELATED TO

PRESSURE-TEMPERATURE LIMITS

TECHNICAL SPECIFICATION CHANGES

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

Section I - Description of Changes

This application for amendment to the Indian Point 3 Technical Specifications seeks to amend Section 3.1.B (Heatup and Cooldown), Section 4.3 (Reactor Coolant System Integrity Testing), and Section 3.1.C (Minimum Conditions for Criticality). Sections 3.1.B and 4.3 are being amended to incorporate revised pressure-temperature limits. These revisions are being made in accordance with Reference 1, which requested that licensees use the methodology of Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," to predict the effect of neutron radiation on reactor vessel materials. Section 3.1.C is being amended to delete Section 3.1.C.2 which establishes pressure-temperature requirements on the reactor coolant system when the reactor is critical.

Section II - Evaluation of Changes

Generic Letter 88-11 requested that licensees use the methodology of RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," to predict the effect of neutron radiation on reactor vessel material. In accordance with the requirements of RG 1.99, Revision 2, and in keeping with the methodologies described therein, a series of heatup and cooldown curves have been developed for Indian Point 3. The heatup curves cover a range of heatup rates from 20 $^{\circ}$F/hr through 60 $^{\circ}$F/hr, and the cooldown curves a range from 0 $^{\circ}$F/hr (isothermal) to 100 $^{\circ}$F/hr.

The calculations supporting these curves incorporate data from analysis of Indian Point 3 surveillance capsules T, Y and Z performed in 1979, 1983 and 1988 respectively. Analysis of capsule Z was performed using the guidance provided in RG 1.99, Revision 2, and is described in Reference 2. Reference 3, (attached) provides a detailed description of the methods used to prepare these curves.

Section 3.1.C.2 is being removed because it provides pressure-temperature limits for critical operation below 450 $^{\circ}$F. Since such operation is specifically prohibited by specification 3.1.C.3., specification 3.1.C.2 is unnecessary and may be misleading.

Section III - No Significant Hazards Evaluation

Consistent with the requirements of 10 CFR 50.92, the enclosed application involves no significant hazards based on the following information:

(1) Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response:

Neither the probability nor the consequences of a previously analyzed accident is increased due to the proposed changes. The adjusted reference temperature of the limiting beltline material was used to correct the pressure-temperature curves to account for irradiation effects. Thus, the operating limits are adjusted to incorporate the initial fracture toughness conservatism present when the reactor vessel was new. The adjusted reference temperature calculations were performed utilizing the guidance contained in RG 1.99, Revision 2. The updated curves provide assurance that brittle fracture of the reactor vessel is prevented.

Removal of the pressure-temperature limits for criticality does not increase the consequences or probability of any accident because these limits are conservatively encompassed and are bounded by the requirements of specification 3.1.C.3.

(2) Does the proposed license amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response:

The updated P-T limits will not create the possibility of a new or different kind of accident. The revised operating limits merely update the existing limits by taking into account the effects of radiation embrittlement, utilizing criteria defined in RG 1.99, Revision 2. The updated P-T curves are conservatively adjusted to account for the effect of irradiation on the limiting reactor vessel material.

No change is being made to the way the pressure-temperature limits provide plant protection. No new modes of operation are involved. Incorporating this amendment does not necessitate physical alteration of the plant.

(3) Does the proposed amendment involve a significant reduction in a margin of safety?

Response:

The proposed amendment does not involve a significant reduction in the margin of safety. The pressure-temperature operating limits are designed to provide a margin of safety. The required margin is specified in ASME Boiler and Pressure Vessel Code, Section III, Appendix G and 10 CFR 50 Appendix G. The revised curves are based on the latest NRC guidelines along with actual neutron flux/fluence data for the reactor vessel. The new limits retain a margin of safety equivalent to the original margin when the vessel was new and the fracture toughness was slightly greater. The new operating limits account for irradiation embrittlement effects, thereby maintaining a conservative margin to safety.

The removal of the pressure-temperature limits for criticality does not reduce the plant safety margin because these limits are conservatively encompassed and bounded by the requirement of specification 3.1.C.3.

Section IV - Impact of Changes

These changes will not adversely impact the following:

ALARA Program
Security and Fire Protection Programs
Emergency Plan
FSAR or SER Conclusions
Overall Plant Operations and the Environment

Section V - Conclusions

The incorporation of these changes: a) will not increase the probability nor the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report; b) will not increase the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report; c) will not reduce the margin of safety as defined in the bases for any Technical Specification; d) does not constitute an unreviewed safety question; and e) involves no significant hazards considerations as defined in 10 CFR 50.92.

Section VI - References

a) IP-3 Final Safety Analysis Report
b) IP-3 Safety Evaluation Report