

ATTACHMENT I

REVISED TECHNICAL SPECIFICATION BASES PAGES

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
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B. HEATUP AND COOLDOWN

Specifications

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) averaged over one hour shall be limited in accordance with Figure 3.1.B-1 and Figure 3.1.B-2 for the service period up to 21.63 effective full-power years. The heatup or cooldown rate shall not exceed 100°F/hr.
 - 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 present may be obtained by interpolation.
 - b. Figure 3.1.B-1 and Figure 3.1.B-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.B-1 and Figure 3.1.B-2 shall be recalculated periodically using methods discussed in WCAP-7924A and WCAP-12796 and results of surveillance specimen testing as covered in WCAP-7323⁽⁷⁾ and as specified in Specification 3.1.B.3 below. 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 six months prior to scheduled specimen removal.
3. The reactor vessel surveillance program* includes six specimen capsules to evaluate radiation damage based on pre-irradiation and post-irradiation tensile and Charpy V notch (wedge open loading) testing of specimens.

* Refer to UFSAR Section 4.5, WCAP-7323, and Indian Point Unit No. 2, "Application for Amendment to Operating License," sworn to on February 3, 1981.

The specimens will be removed and examined at the following intervals:

Capsule 1	End of Cycle 1 operation
Capsule 2	End of Cycle 2 operation
Capsule 3	End of Cycle 5 operation
Capsule 4	End of Cycle 8 operation
Capsule 5	End of Cycle 16 operation
Capsule 6	Spare

4. 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.
5. 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.
6. 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 UFSAR. 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 UFSAR. 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.

The current heatup and cooldown curves are based upon a maximum fluence of 0.98×10^{19} n/cm² at the inner reactor vessel surface (45° angle, vessel belt line). This fluence is based upon plant operation for a nominal period of 21.63 EFPYs (Operation up to Cycle 9 for 9.63 EFPYs at 2758 MWt power level and beyond Cycle 9 for 12 EFPYs at 3071.4 MWt power level and T average of 579.7°F). Any changes in the operating conditions could result in an extension of the allowable EFPYs, since the fluence (or ΔRT_{NDT} due to irradiation) is the controlling factor in the generation of these curves.

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 and Pressure Vessel Code, 1995 Edition as amended in 1996 (A96), Section XI, Appendix G, and the calculation methods described in WCAP-7924A⁽⁴⁾ and WCAP-12796⁽¹³⁾ and MSE-REME-0076⁽¹⁴⁾.

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^(8,9). The second surveillance capsule was removed during the 1978 refueling outage. That capsule has been tested by SWRI and the results have been evaluated and reported⁽¹⁰⁾. The third vessel material surveillance capsule was removed during the 1982

refueling outage. This capsule has been tested by SWRI and the results have been evaluated and reported⁽¹¹⁾. The fourth surveillance capsule was removed during the 1987 refueling outage. This capsule has been tested by SWRI and the results have been evaluated and reported⁽¹²⁾. Heatup and cooldown curves (Figures 3.1.B-1 and 3.1.B-2) were developed by Westinghouse⁽¹³⁾. These curves are essentially identical to those obtained using the new Appendix G methods⁽¹⁴⁾.

The maximum shift in RT_{NDT} at a fluence of 0.98×10^{19} n/cm², (nominal 21.63 EFPYs of operation) is projected to be 155.5°F at the 1/4 T and 105°F at the 3/4 T vessel wall locations, per Plate B2002-3 the controlling plate. The initial value of RT_{NDT} for this plate of the IP2 reactor vessel was 21°F. The heatup and cooldown curves 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 for the above fluence⁽¹²⁾.

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 XI 1995 Edition as amended in 1996 (A96), of the ASME Boiler and Pressure Vessel Code and are discussed in detail in WCAP-7924A⁽⁴⁾ and WCAP-12796⁽¹³⁾ and MSE-REME-0076⁽¹⁴⁾.

The approach specifies that the allowable total stress intensity factor (K_t), 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:

$$2 K_{im} + K_{it} \leq K_{IR} \quad (1)$$

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 condition (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 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 I.D. flaw is clearly the worst case.

As in the case of heatup, allowable pressure-temperature relations are generated for both steady and finite cooldown rate situations. Composite limit curves are then constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because system control

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.B-2 represent a composite curve consisting of the more conservative values calculated for steady state and the specific cooling rate shown.

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 UFSAR, Section 4.1.5.
- (2) ASME Boiler & Pressure Vessel Code, Section III, Summer 1965, N-415.
- (3) Indian Point Unit No. 2 UFSAR, 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 No. 02-4531 - "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule T," E.B. Norris, June 30, 1977.
- (9) Supplement to Final Report - SWRI Project No. 02-4531 - "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule T," E.B. Norris, December 1980.
- (10) Final Report - SWRI Project No. 02-5212 - "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule Y," E.B. Norris, November 1980.

- (11) Final Report - SWRI Project No. 06-7379 - "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule Z," E.B. Norris, April 1984.
- (12) Final Report - SWRI Project No. 17-2108 (Revised)- "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule V," F.A. Iddings - SWRI, March, 1990.
- (13) WCAP-12796, "Heatup and Cooldown Limit Curves for the Consolidated Edison Company Indian Point Unit 2 Reactor Vessel," N.K. Ray, Westinghouse Electric Corporation.
- (14) MSE-REME-0076, "Indian Point Unit 2 Heatup and Cooldown Limit Curves for Normal Operation," P.A. Grendys, Westinghouse Electric Corporation.

ATTACHMENT II

DOCUMENTATION FOR WITHHOLDING OF
WESTINGHOUSE PROPRIETARY INFORMATION

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247
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