ATTACHMENT A

AMENDMENT NO. 2 TO
APPLICATION FOR AMENDMENT
TO OPERATING LICENSE

Technical Specification
Page Revisions

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
Facility Operating License No. DPR-26
February, 1985

3 LIMITING CONDITIONS FOR OPERATION

3.0.1 In the event a Limiting Condition for operation (LCO) and/or associated action requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, the unit shall be placed in at least hot shutdown within the next 7 hours, and in at least cold shutdown within the following 30 hours unless corrective measures are completed that restore compliance to the LCO within these time intervals as measured from initial discovery or until the reactor is placed in a condition in which the LCO is not applicable. Exceptions to these requirements shall be stated in the individual specifications.

3.0.2 A system, subsystem, train, component or device shall not be considered inoperable solely because its normal power source is inoperable, or solely because its emergency power source (i.e., diesel, battery) is inoperable. In such instances the equipment served by the inoperable power source shall be considered operable for purposes of compliance with their individual equipment LCOs and only the LCO for the inoperable power source shall apply.

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the Reactor Coolant System.

Objective

To specify those limiting conditions for operation of the Reactor Coolant System which must be met to ensure safe reactor operation.

A. OPERATIONAL COMPONENTS

1. Coolant Pump

   a. Except as noted in 3.1.A.1.b. below, four reactor coolant pumps shall be in operation during power operation.

   b. During power operation, one reactor coolant pump may be out of service for testing or repair purposes for a period not to exceed four hours.

   c. During shutdown conditions with fuel in the reactor, the operability requirements for reactor coolant and/or residual heat removal pumps specified in Table 3.1.A-1 shall be met.

d. When RCS temperature is less than or equal to 295°F, the requirements of Specification 3.1.A.4 regarding startup of a reactor coolant pump with no other reactor coolant pumps operating shall be adhered to.

2. Steam Generator

Two steam generators shall be capable of performing their heat transfer function whenever the reactor coolant system is above 350°F.

3. Safety Valves

a. At least one pressurizer code safety valve shall be operable, or an opening greater than or equal to the size of one code safety valve flange shall be provided to allow for pressure relief, whenever the reactor head is on the vessel except for hydrostatically testing the RCS in accordance with Section XI of the ASME Boiler and Pressure Vessel Code.

b. All pressurizer code safety valves shall be operable whenever the reactor is critical.

c. The pressurizer code safety valve lift settings shall be set at 2485 psig with ±1% allowance for error.

4. Overpressure Protection System (OPS)

a. Except as permitted by Table 3.1.A-2, the OPS shall be armed and operable when the RCS temperature is ≤ 295°F. When OPS is required to be operable, the PORV will have settings within the limits shown in Figure 3.1.A-1.

b. The requirements of 3.1.A.4.a may be modified to permit one PORV and/or its associated motor operated valve to be inoperable for a maximum of seven (7) consecutive days. If the PORV and/or its series motor operated valve is not restored to operable status within this seven (7) day period, or if both PORVs or their associated block valves are inoperable, action shall be initiated immediately to place the reactor in a condition where OPS operability is not required.

c. In the event either a PORV(s) or a RCS vent(s) is used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Nuclear Regulatory Commission within 30 days pursuant to Specification 6.9.2.f. The report shall describe the circumstances initiating the transient, the effect of the PORV(s) or vent(s) on the transient, and any corrective action necessary to prevent recurrence.

5. <u>Power Operated Relief Valves (PORVs)/Block Valves (for operation above 350°F)</u>

a. Whenever the reactor coolant system is above 350°F, the PORVs and their associated block valves shall be operable with the block valves either open or closed.

b. If a PORV becomes inoperable when above 350°F, its associated block valve shall be maintained in the closed position.

c. If a PORV block valve becomes inoperable when above 350°F, the block valve shall be closed and deenergized.

d. If the requirements of specification 3.1.A.5.a, 3.1.A.5.b or 3.1.A.5.c above cannot be satisfied, compliance shall be established within four (4) hours, or the reactor shall be placed in the hot shutdown condition within the next six (6) hours and subsequently cooled below 350°F.

6. <u>Pressurizer Heaters</u>

a. Whenever the reactor coolant system is above 350°F, the pressurizer shall be operable with at least 150kw of pressurizer heaters.

b. If the requirements of specification 3.1.A.6.a cannot be met, restore the required pressurizer heater capacity to operable status within 72 hours or the reactor shall be placed in the hot shutdown within the next six(6) hours and subsequently cooled below 350°F.

<u>Basis</u>

When the boron concentration of the Reactor Coolant System (RCS) is to be reduced, the process must be uniform to prevent sudden reactivity changes in the reactor. The requirement for at least one reactor coolant pump or one residual heat removal pump to be in operation is to provide flow to ensure mixing, prevent stratification, and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. Below 350°F, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be operable. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The residual heat removal pump will circulate the primary system volume in approximately one half hour. The pressurizer is of no concern because of the low pressurizer volume and because the pressurizer boron concentration will be higher than that of the rest of the reactor coolant system.

Heat transfer analyses show that reactor heat equivalent to 10% of rated power can be removed with natural circulation only [1]; hence, the specified upper limit of 2% rated power without operating pumps provides a substantial safety factor.

The specification that all reactor coolant pumps be operational during power operation is to assure that adequate core cooling will be provided. This flow will keep the minimum departure from nucleate boiling ratio above 1.30; therefore, cladding damage and release of fission products will not occur.

The Overpressure Protection System (OPS) is designed to relieve the RCS pressure for certain unlikely overpressure transients to prevent these incidents from causing the peak RCS pressure from exceeding 10CFR50, Appendix G limits. When the OPS is "armed" MOVs 535 and 536 are in the open position, and the PORVs will open upon receipt of the appropriate signal. This OPS arming can be accomplished either automatically by the OPS when the RCS is below a prescribed temperature or manually by the operator.

The OPS will be set to cause the PORVs to open at a pressure sufficiently low to prevent exceeding the Appendix G limits for the following events:

1. Startup of a reactor coolant pump with no other reactor coolant pumps running and the steam generator secondary side water temperature hotter than the RCS water temperature.

2. Letdown isolation with three charging pumps operating.

3. Startup of one safety injection pump.

4. Loss of residual heat removal causing pressure rise from heat additions from core decay heat or reactor coolant pump heat.

5. Inadvertant activation of the pressurizer heaters.

Consideration of the above events provides bounding PORV setpoints for other potential overpressure conditions caused by heat or mass additions at low temperature.

The RCS is protected against overpressure transients when RCS temperature is less than or equal to 295^{o}F by: (1) restricting the number of charging and safety injection pumps that can be energized to that which can be accommodated by the PORV's or the gas space in the pressurizer, (2) providing administrative controls on starting of a reactor coolant pump when the primary water temperature is less than the secondary water temperature, or (3) providing vent area from the RCS to containment for those situations where neither the PORV's nor the available pressurizer gas space are sufficient to preclude the pressure resulting from postulated transients from exceeding the limits of 10 CFR 50, Appendix G.

The restrictions on starting a reactor coolant pump with the secondary side water temperature higher than the primary side will prevent RCS overpressurizations from the resultant volumetric swell into the pressurizer that is caused by potential heat additions from the startup of a reactor coolant pump without any other reactor coolant pumps operating. When pressurizer level is between 30 and 85% of span, protection is provided through the use of the PORV's. When pressurizer level is less than 30% of span additional restrictions on pressurizer pressure make reliance on the PORV's unnecessary since the gas compression resulting from the insurge of liquid from the RCS pump start is insufficient to cause RCS pressure to exceed the Appendix G limits. The same method, i.e., control of pressurizer pressure and level, is used to accommodate the mass insurge into the pressurizer from safety injection and charging pump starts when the PORV's are not operational.

An additional restriction is put on the reactor coolant pump start when the secondary system water temperature is less than or equal to 40^{o}F higher than the primary system water temperature and the pressurizer level is greater than 30%. This restriction is to prohibit starting the first reactor coolant pump when the RCS temperature is between 267^{o}F and 295^{o}F. The purpose of the restriction is to assure that the temperature rise resulting from the transient will not be outside the temperature limits for OPS actuation.

When comparison to the Appendix G limits is made, the comparison is to the isothermal Appendix G curve. Other than the delay time associated with opening the PORVs, and the error caused by non-uniform RCS metal and water temperatures during heat addition transients, the analysis does not make any allowance for instrument error. Instrument error will be taken into account when the OPS is set; i.e., the instrumentation will be set so that the PORVs will open at less than the required setpoint including allowance for instrument errors.

The determination of reactor coolant temperature may be made from the Control Room instrumentation. The determination of the steam generator water temperature may be made in the following ways:

(a) Assuming that the secondary side water temperature is at the saturation temperature corresponding to the secondary side steam pressure indicated on the Control Room instrumention, or

(b) Conservatively assuming that the secondary side water temperature is at the reactor coolant temperature at which the last RCP was stopped during cooldown, or

(c) Actual or inferred measurement of the secondary side steam generator water temperature at those times it can be measured (such as return from a refueling outage).

Each of the pressurizer code safety valves is designed to relieve 408,000 lbs. per hr. of the saturated steam at the valve set point. Below approximately 350°F and 450 psig in the Reactor Coolant System, the Residual Heat Removal System can remove decay heat and thereby control system temperatures and pressure.(2)

If not residual heat were removed by the Residual Heat Removal System the amount of steam which could be generated at safety valve relief pressure would be less than half the capacity of a single valve. One valve therefore provides adequate protection for overpressurization.

The combined capacity of the three pressurizer safety valves is greater than the maximum surge rate resulting from complete loss of load (3) without a direct trip or any other control.

Two steam generators capable of performing their heat transfer function will provide sufficient heat removal capability to remove decay heat after a reactor shutdown.

All pressurizer heaters are supplied electrical power from an emergency bus. The requirement that 150kw of pressurizer heaters and their associated controls be operable when the reactor coolant system is above 350°F provides assurance that these heaters will be available and can be energized during a loss of offsite power condition to assist in maintaining natural circulation at hot shutdown.

The power operated relief valves (PORVs) can operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to provide a relief path when desirable and to ensure the ability to seal off possible RCS leakage paths. Both the PORVs and the PORV block valves are subject to periodic valve testing for operability in accordance with the ASME Code Section XI as specified in the Indian Point Unit No. 2 Inservice Inspection and Testing Program.

Reference

1) FSAR Section 14.1.6
2) FSAR Section 9.3.1
3) FSAR Section 14.1.10

Table 3.1.A-1

Reactor Coolant (RC) Pumps/Residual Heat Removal (RHR) Pump(s) Operability/Operating Requirements for Decay Heat Removal and Core Mixing

(1) Reactor Condition	(2) Required No. of Pumps Operating	(3) Required No. of Pumps Operable (including operating pump)	(4) Action Required if Condition of Column (2) or (3) is not met
Hot shutdown Tavg > 350°F (Excluding loss of offsite power)	Two RCPs	Two RCPs	With less than two reactor coolant pumps operating, maintain the reactor trip breakers open. With no reactor coolant pumps operating, Tavg may be maintained above 350°F for up to one hour provided: (1) no operations are permitted that would cause dilution of the reactor coolant system, and (2) RCS temperature is maintained at least 10°F below saturation temperature. If a RCP has not been restored to operating status within the one hour permitted, take action as listed below for no operable pumps. With only one RCP operable, restore a second RCP to operable status within 72 hours or bring the RCS temperature to 350°F. Except for testing, with no RCPs operable, immediately initiate action to bring RCS temperature to 350°F.

Table 3.1.A-1

Reactor Coolant (RC) Pumps/Residual Heat Removal (RHR) and Pump(s) Operability/Operating Requirements for Decay Heat Removal and Core Mixing

(1) Reactor Condition	(2) Required No. of Pumps Operating	(3) Required No. of Pumps Operable (including operating pump)	(4) Action Required if Condition of Column (2) or (3) is not met
Hot shutdown $T_{avg} \leq 350^{o}F$	One RCP or one RHR pump	Two RCPs or Two RHR pumps or one RCP and one RHR pump	The requirement to have at least one RCP or RHR pump in operation may be suspended for up to one hour provided: (1) no operations are permitted that would cause dilution of the reactor coolant system, and (2) RCS temperature is maintained at least $10^{o}F$ below saturation temperature. If a pump has not been restored to operating status within the one hour permitted, take action as listed below for no operable pumps. With only one pump (RHR or RCP) operable, either restore a second pump to operable status or be in cold shutdown within 20 hours. With no pumps operable, suspend all operations involving a reduction in boron concentration and immediately initiate action to restore at least one pump to operable status.

Table 3.1.A-2
OPS Operability Requirements

Reactor Coolant Pumps

With OPS operable at or below 295^{O}F, a reactor coolant pump can be started (or jogged) with no other reactor coolant pumps operating if:

(1) The temperature of all steam generators is less than or equal to the RCS temperature; or,

(2) The temperature of all steam generators is less than or equal to 40^{O}F higher than the RCS temperature and:

- o RCS temperature is less than or equal to 267^{O}F,
- o Pressurizer level is between 30 - 85% of span; or

(3) The temperature of all steam generators is less than or equal to 100^{O}F higher than RCS temperature and:

- o RCS pressure less than or equal to 450 psig,
- o RCS temperature is greater than or equal to 145^{O}F,
- o Pressurizer level is less than or equal to 30% of span.

With OPS inoperable at or below 295^{O}F, a reactor coolant pump can be started (or jogged) with no other reactor coolant pumps operating if:

(1) The temperature of all steam generators is less than or equal to the RCS temperature; or,

(2) The temperature of all steam generators is less than or equal to 100^{O}F higher than RCS temperature and:

- o RCS pressure is less than or equal to 450 psig,
- o RCS temperature is greater than or equal to 145^{O}F,
- o Pressurizer level is less than or equal to 30% of span.

Table 3.1.A-2

OPS Operability Requirements

Safety Injection and Charging Pumps

With OPS operable at or below 295°F, no more than one (1) safety injection (SI) and three (3) charging pumps may be energized.

OPS is not required to be operable at or below 295°F, if either the conditions of Column I or the conditions of Column II below are met for the specified conditions:

Maximum Number of Energized Pumps (SI and/or charging)		I	II
		Operating Restrictions (pressurizer pressure, pressurizer level, and RCS temperature)	Vent Area to Containment Atmosphere (square inches)
SI	Charging		
0	1	See Figure 3.1.A-2	2.00
1	3	See Figure 3.1.A-3	2.00
3	3	-------	5.00

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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.B-1 and Figure 3.1.B-2 for the service period up to 15 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 heat up 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 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. The specimens will be removed and examined at the following intervals:

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

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 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 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. These samples are evaluated according to ASTM E185.[6] To compensate for any increase in the RT_{NDT} caused by irradiation, the limits on the pressure-temperature relationship are periodically changed to stay within the stress limits during heatup and cooldown, in accordance with the requirements of the ASME Boiler & Pressure Vessel Code, 1974 Edition, Section III, Appendix G, and the calculation methods described in WCAP-7924A[4].

The first reactor vessel material surveillance capsule was removed during the 1976 refueling outage. That capsule was tested by Southwest Research Institute (SWRI) and the results were evaluated and reported.[8] [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] Based on the SWRI evaluation, heatup and cooldown curves (Figures 3.1.B-1 and 3.1.B-2) were developed for up to fifteen (15) effective full power years (EFPYs) of reactor operation.

The maximum shift in RT_{NDT} after 15 EFPYs of operation is projected to be 142°F at the 1/4T and 71°F at the 3/4T vessel wall locations, per Plate B2002-3 the controlling plate. The initial value of RT_{NDT} for the IP2 reactor vessel was 34°F as described in Table 3.1.B-1. The heatup and cooldown curves for 15 EFPYs have been computed on the basis of the RT_{NDT} of Plate B2002-3 because it is anticipated that the RT_{NDT} of the reactor vessel beltline material will be highest for Plate B2002-3 at least through that time period.[11]

Heatup and Cooldown Curves

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Non Mandatory Appendix G in Section III 1974 Edition of the ASME Boiler and Pressure Vessel Code and discussed in detail in WCAP-7924A.[4]

The approach specifies that the allowable total stress intensity factor (K_I) at any time during heatup or cooldown cannot be greater than that shown on the

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:

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

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 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 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 1/4T. The thermal gradients induced during cooldown tend to produce tensile stresses at the 1/4T 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 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 calcuated 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.

Details of these calculations are provided in WCAP-7924A[4].

Pressurizer Limits

Although the pressurizer operates at temperature ranges above those for which there is reason for concern about brittle fracture, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, 1965 Edition and associated Code Addenda through the Summer 1966 Addendum.

References

(1) Indian Point Unit No. 2 FSAR, Section 4.1.5.

(2) ASME Boiler & Pressure Vessel Code, Section III, Summer 1965, N-415.

(3) Indian Point Unit No. 3 FSAR, Section 4.2.5.

(4) WCAP-7924A, "Basis for Heatup and Cooldown Limit Curves," W.S. Hazelton, S. L. Anderson, S. E. Yanichko, April 1975.

(5) ASME Boiler and Pressure Vessel Code, Section III, 1974 Edition, Appendix G.

(6) ASTM E185-79, Surveillance Tests on Structural Materials in Nuclear Reactors.

(7) WCAP-7323, "Consolidated Edison Company, Indian Point Unit No. 2 Reactor Vessel Radiation Surveillance Program", S.E. Yanichko, May 1969.

(8) Final Report - SWRI Project 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.

TABLE 3.1.B-1

Indian Point Unit No. 2
Reactor Vessel Core Region Material

Plate	Copper Content	Initial RT NDT
B 2002-1	0.25	34°F
B 2002-2	0.14	21°F
B 2002-3*	0.14	21°F
HAZ	-	0°F
Weld Material	-	0°F

References:

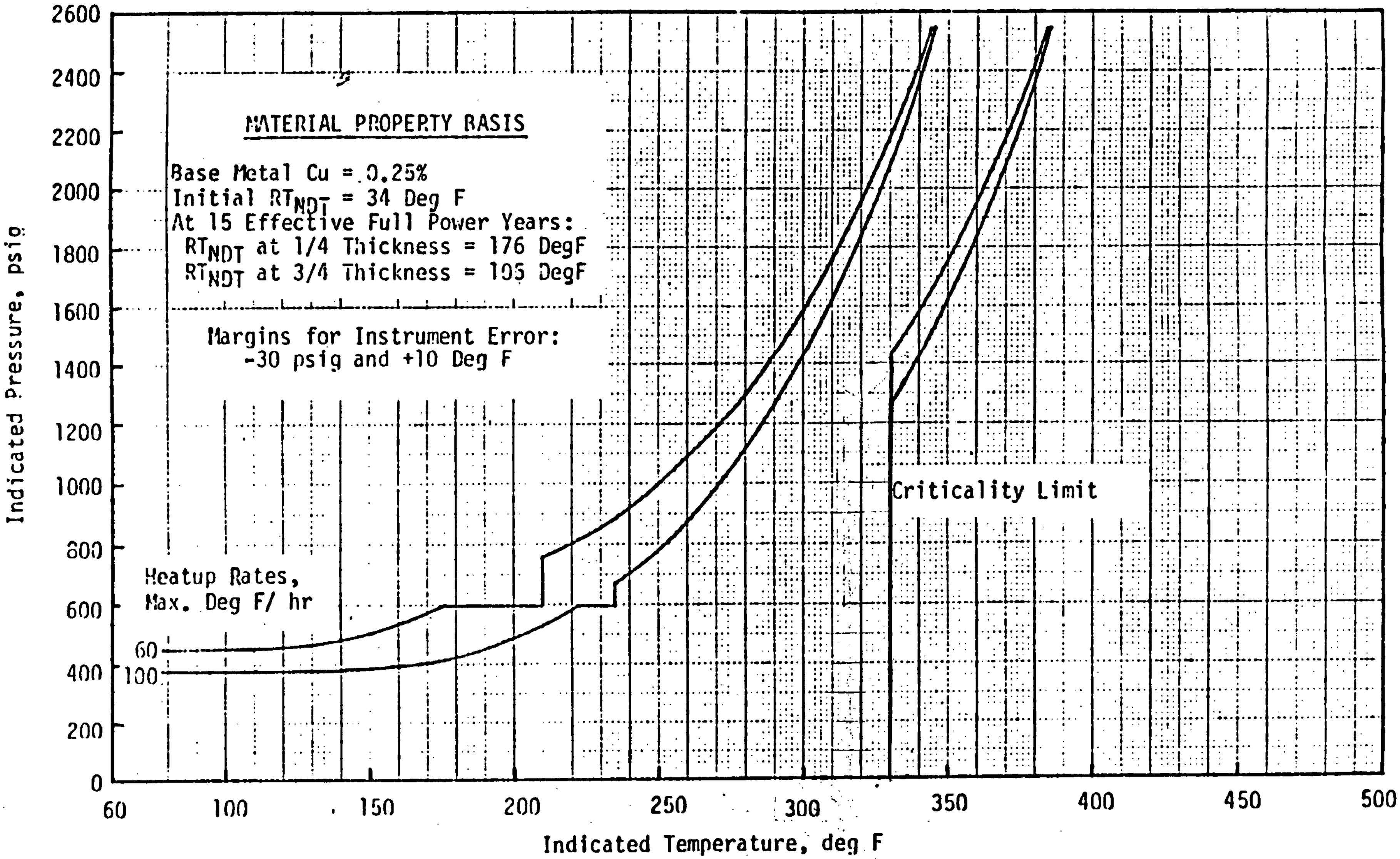
(1) Letter No. IPP-75-50, Westinghouse to Con Edison Dated May 16, 1975

(2) Letter dated March 29, 1978 from W. J. Cahill, Jr. (Consolidated Edison) to R. W. Reid (NRC), "Indian Point Unit No. 2 Reactor Vessel Material Surveillance Program."

(3) 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.

Notes:

* Based on Reference (3) above, the bounding values for copper (0.25%) and initial RT_{NDT} (34°F) are applied to the controlling plate (B2002-3) for the purpose of generating the heatup and cooldown limitations.



INDIAN POINT UNIT NO. 2 COOLANT HEATUP LIMITATIONS
APPLICABLE FOR PERIODS UP TO 15 EFFECTIVE FULL POWER YEARS

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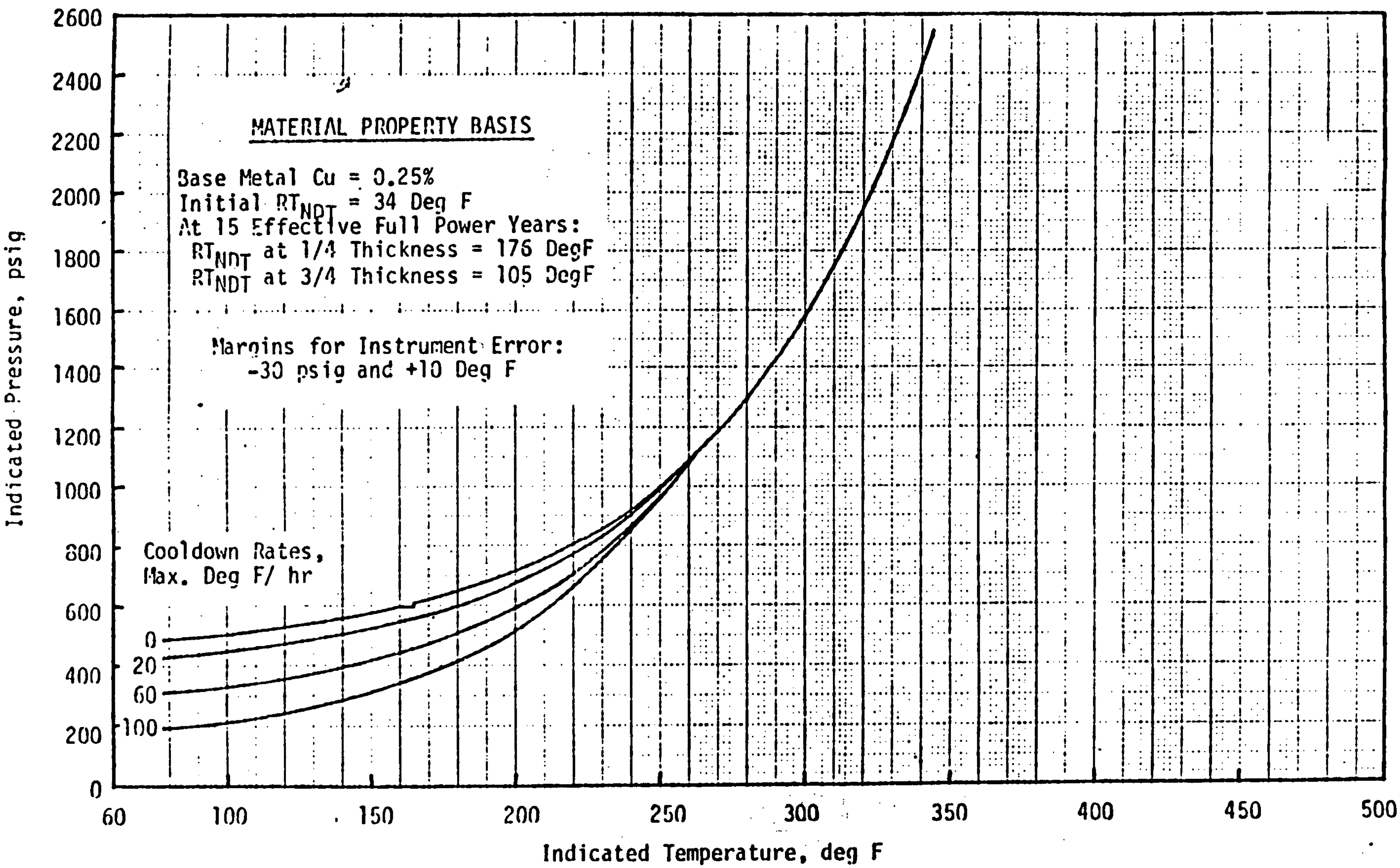
Figure 3.1.B-1



INDIAN POINT UNIT NO. 2 COOLANT COOLDOWN LIMITATIONS
APPLICABLE FOR PERIODS UP TO 15 EFFECTIVE FULL POWER YEARS

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Figure 3.1.B-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 modification or repairs have been made which involve new strength welds on components, the new welds shall meet the requirements of the applicable version of ASME Section XI as specified in the Con Edison Inservice Inspection and Testing Program in effect at the time.

c) The Reactor Coolant System leak test temperature-pressure relationship shall be in accordance with the limits of Figure 4.3-1 for heatup for the first fifteen (15) effective full-power yrs. of operation. Figure 4.3-1 will be recalculated periodically. Allowable pressure during cooldown for the leak test temperature shall be in accordance with Figure 3.1.B-2.

Basis

For normal opening, the integrity of the system, in terms of strength, is unchanged. If the system doe 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-desctructive 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 operaton.

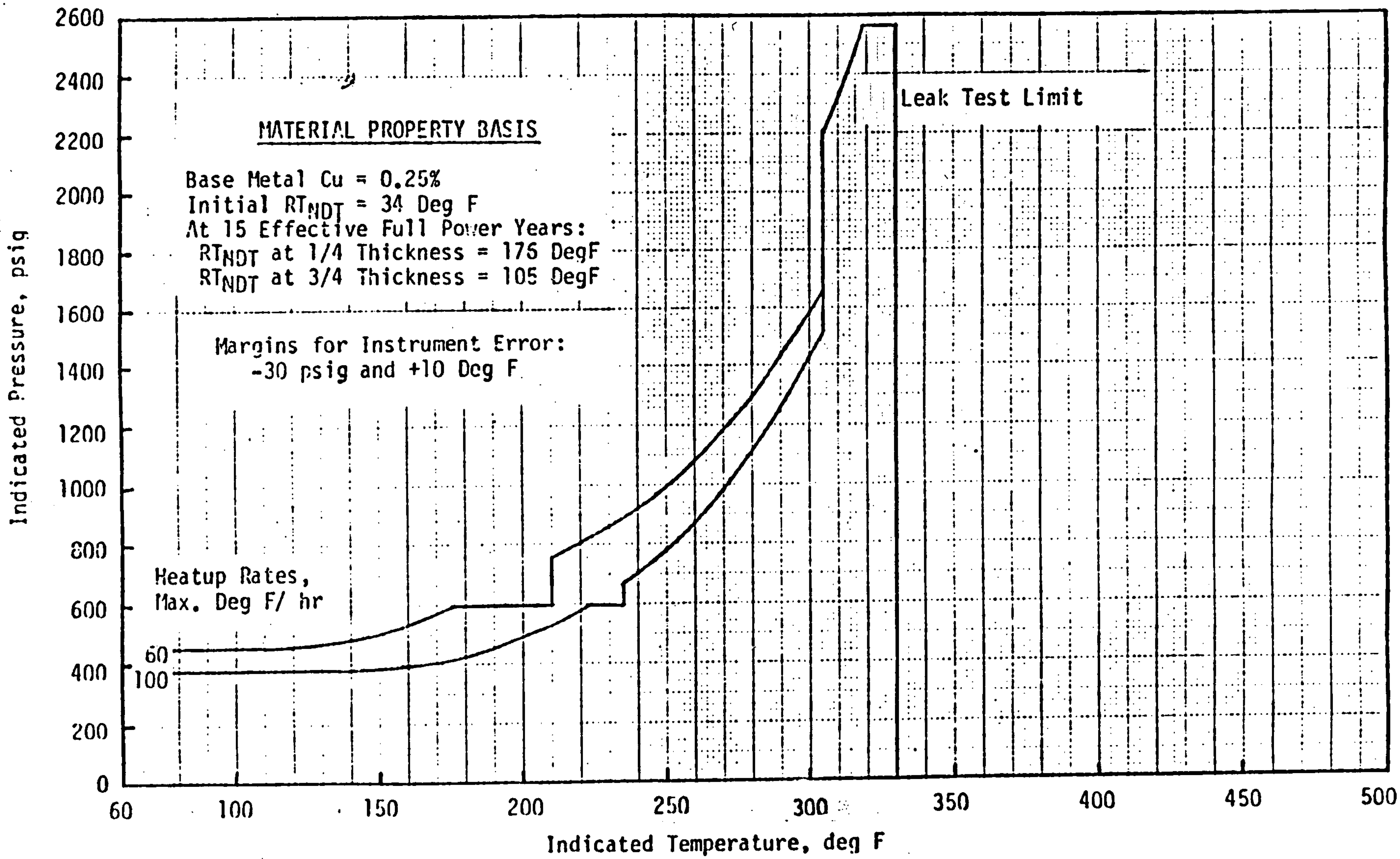
The inservice leak temperatures are shown on Figure 4.3-1. The temperatures are calculated in accordance with ASME Code Section III, 1974 Edition, Appendix G. This Code requires that a safety factor of 1.5 times the stress intensity factor caused by pressure be applied to the calculation.

For the first fifteen (15) effective full-power years, it is predicted that the highest RT_{NDT} in the core region taken at the 1/4 thickness will be 176°F. The minimum inservice leak test temperature requirements for periods up to fifteen (15) 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.B-2 must not be exceeded. Figures 4.3-1 and 3.1.B-2 are recalculated periodically, using methods discussed in WCAP-7924A and results of surveillance specimen testing, as covered in WCAP-7323.

Reference

1. FSAR, Section 4



INDIAN POINT UNIT NO. 2 VESSEL LEAK TEST LIMITATIONS
APPLICABLE FOR PERIODS UP TO 15 EFFECTIVE FULL POWER YEARS

Figure 4.3-1

Amendment No.

ATTACHMENT B

AMENDMENT NO. 2 TO
APPLICATION FOR AMENDMENT
TO OPERATING LICENSE

Safety Assessment

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
Facility Operating License No. DPR-26
Feburary, 1985

SAFETY ASSESSMENT

Discussion:

The proposed changes, contained in Attachment A to this Application, would modify portions of the present Technical Specifications and portions of Consolidated Edison's previous February 14, 1983 license amendment application to reflect revised pressure/temperature limitations for reactor coolant system heatup, cooldown, hydrostatic test and low temperature overpressure protection applicable through fifteen (15) effective full power years (EPFYs). In addition, a revision is requested to changes proposed in the February 14, 1983 application regarding the number of reactor coolant pumps required to be operating when the reactor coolant system is at hot shutdown and above 350°F.

Sections 3.1.B and 4.3 of the present Indian Point Unit No. 2 Technical Specifications contain limitations for reactor coolant system heatup, cooldown and hydrostatic testing applicable through 7 EFPYs of reactor operation. Based on present projections, the Indian Point reactor is expected to exceed 7 EFPYs in July, 1985. The changes proposed to technical specification sections 3.1.B and 4.3 in this application will incorporate revised requirements applicable through 15 EFPYs. These new requirements are based on the testing and analysis of the most recent reactor vessel surveillance capsule (i.e., Capsule Z-removed during the 1982 refueling/maintenance outage). The final report on the capsule analysis entitled, "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule Z", dated April, 1984, was submitted to the NRC by Consolidated Edison letter dated May 7, 1984. For demonstrative purposes, that report contained heatup and cooldown curves and predicted shifts in RT_{NDT} for projected operation up to 32 EFPYs. The 15 EFPY limitations proposed in this application are based on the same data and analysis contained in the Capsule Z report for the 32 EFPY limitations. The revisions to Table 3.1-1 of the present technical specifications (contained in renumbered Table 3.1.B-1 of this application) regarding reactor vessel core region material are based on the detailed information provided to the NRC Staff by Consolidated Edison letter dated March 29, 1978 as applied in the Capsule Z report provided to the Staff on May 7, 1984. The net effect is that all heatup and cooldown limitations are now based on an initial RT_{NDT} of 34°F (vs. 60°F) for the controlling vessel plate material.

With regard to low temperature overpressure protection, proposed technical specifications were provided in Consolidated Edison's February 14, 1983 license amendment application but have not to date been issued by the NRC Staff. However, a Staff safety evaluation report documenting the acceptability of the Indian Point Unit No. 2 overpressure protection system was issued by letter dated April 24, 1984. Since overpressure protection limitations are keyed to the reactor coolant system heatup, cooldown and hydrostatic testing limitations, such limitations must also be revised as appropriate for applicability through 15 EFPYs.

With respect to the low temperature overpressure protection pressure limitations over the applicable temperature range, a comparison of the present 7 EFPY isothermal Appendix G curve and the proposed 15 EPFY isothermal curve was made. The 7 EFPY isothermal curve provides slightly more conservative pressure/temperature limitations than does the 15 EFPY isothermal curve primarily due to the revised initial RT_{NDT}. Therefore, with the exception of the OPS arming temperature and maximum starting temperature for the first RCP as discussed below, the proposed parametric and operational limitations contained in our previous February 14,1983 Application remain unchanged.

The new minimum temperature for the reactor coolant system hydrostatic testing window per Figure 4.3-1 is approximately 305°F. Thus, the overpressure protection system arming temperature has been accordingly revised to 295°F (i.e., 305°F-10°F (error) = 295°F) and the maximum temperature for starting the first reactor coolant pump has been revised to 267°F (i.e., 295°F - 28°F (heat addition transient overshoot) = 267°F). These changes have been reflected in this Application as revisions to previously proposed (i.e., February 14, 1983) technical specifications 3.1.A.1, 3.1.A.4, 3.1.A bases, Table 3.1.A-2 and Figures 3.1.A-1, 3.1.A-2 and 3.1.A-3. The theory and analysis supporting these revised limits is the same as previously provided to the NRC Staff. Only the specific limiting values have changed to reflect the shift in the heatup and cooldown limitation curves.

Finally, we are also requesting a change to the previously proposed technical specification Table 3.1.A-1 (not yet issued by NRC Staff) contained in the February 14, 1983 Application. This change would revise from one to two the number of reactor coolant pumps required to be operating when the reactor coolant system is at hot shutdown and greater than 350°F and revise the associated action statement accordingly. This change is necessary to achieve consistency between the proposed technical specification and the assumptions of the safety analysis for the Uncontrolled Rod Withdrawal From Subcritical transient. In addition, an action statement is being added to sheets 1 of 4 of Table 3.1.A-1 to address required actions if no pumps have been operating for more than the one hour permitted.

Basis for no significant hazard consideration determination

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870). Example (vi) of those involving no significant hazards considerations discusses a change which may reduce a safety margin but where the results are clearly within all acceptable criteria with respect to the system or component. The proposed revision to the heatup, cooldown, hydrotest and overpresure protection limitations are in a less restrictive direction and would appear to reduce a safety margin. However, consistent with the Commissions criteria for determining whether a proposed amendment to an operating license involves no significant hazards considerations, 10 CFR 50.92 (48FR14871), we have determined that the proposed changes to the heatup, cooldown, hydrostatic test pressure-temperature limitations and low temperature overpressure protection requirements will not involve a significant increase in the probability or consequences of an accident previously evaluated; or create the possibility of a new or different kind of accident from any previously evaluated; or involve a significant reduction in a margin of safety. The proposed revision reflects conservative values of RT_{NDT} for the reactor vessel and provides a margin of safety which complies with the fracture toughness requirements of 10 CFR 50 Appendix G.

The proposed change regarding the minimum number of reactor coolant pumps required to be operating when at hot shutdown but above 350°F is consistent with example (ii) of the Commission's guidance concerning the application of standards for determining whether a significant hazards consideration exists (48 FR 14870). The proposed change increases the minimum number of reactor coolant pumps required to be operating during specified plant conditions constituting an additional limitation, restriction, or control not presently included in the technical specifications. Likewise, the new action statement added to sheets 1 of 4 and 2 of 4 of Table 3.1.A-1 which would provide new explicit action to be taken if no pumps are operating for longer than the one hour permitted, is also consistent with example (ii) and thus does not constitute a significant hazards consideration.

Therefore, since this application for amendment involves proposed changes that satisfy the Commission's criteria for a determination of no significant hazards consideration or are otherwise consistent with examples of changes for which no significant hazards consideration exists, we have determined that this application involves no significant hazards consideration.

The proposed changes have been reviewed by the Station Nuclear Safety Committee and the Consolidated Edison Nuclear Facilities Safety Committees. Both committees concur that these changes do not represent a significant hazards consideration and will not cause any change in the types or increase in the amounts of effluents or any change in the authorized power level of the facility.