ATTACHMENT I

PROPOSED TECHNICAL SPECIFICATION CHANGES

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3.0 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 305°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 ≤ 305°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.i. 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. Power Operated Relief Valves (PORVs)/Block Valves (for operation above 350°F)

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.

e. With regard to the use of the PORVs/Block Valves as a reactor coolant system vent, the requirements of Specification 3.16 shall be adhered to.

6. Pressurizer Heaters

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 condition within the next six (6) hours and subsequently cooled to below 350°F.

Basis

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 the safety limit DNBRs; 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 10 CFR 50, 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. Inadvertent 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 305°F by: (1) restricting the number of charging and safety injection pumps that can be energized to that which can be accommodated by the PORVs 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 PORVs 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 PORVs. When pressurizer level is less than 30% of span, additional restrictions on pressurizer pressure make reliance on the PORVs 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 PORVs 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 30°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 275°F and 305°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 instrumentation, 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 no 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.

<u>Reference</u>

(1) UFSAR Section 14.1.12

(2) UFSAR Section 9.3.1

(3) UFSAR Section 14.1.8

(4) Revised OPS Setpoints For Indian Point Unit 2, D.M. Speyer and A.P. Ginsberg, Feburary 14, 1991.

Table 3.1.A-2

OPS Operability Requirements

Reactor Coolant Pumps

With OPS operable at or below 305°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 30°F higher than the RCS temperature and:

- o RCS temperature is less than or equal to 275°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°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°F,
- o Pressurizer level is less than or equal to 30% of span.

With OPS inoperable at or below 305°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°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°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 305°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 305°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 Operating Restrictions (pressurizer pressure, pressurizer level, and RCS temperature)	II Vent Area to Containment Atmosphere (square inches)
SI	Charging		
0	1	See Figure 3.1.A-2	2.00 (or 1 PORV fully open)
1	3	See Figure 3.1.A-3	2.00 (or 1 PORV fully open)
3	3	---------	5.00

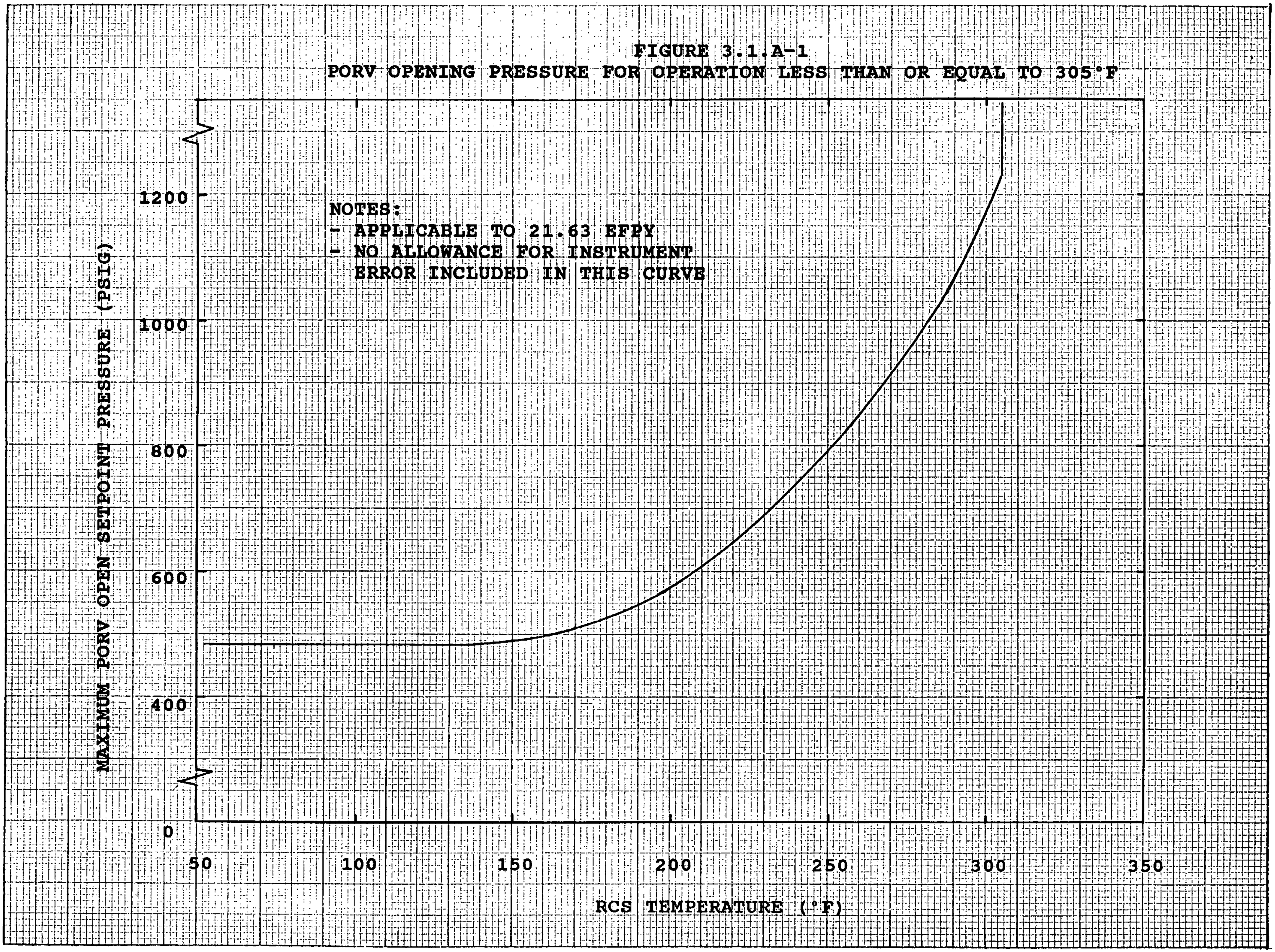


FIGURE 3.1.A-1
PORV OPENING PRESSURE FOR OPERATION LESS THAN OR EQUAL TO 305°F
NOTES:
- APPLICABLE TO 21.63 EFPY
- NO ALLOWANCE FOR INSTRUMENT
ERROR INCLUDED IN THIS CURVE
MAXIMUM PORV OPEN SETPOINT PRESSURE (PSIG)
1200
1000
800
600
400
0
50
100
150
200
250
300
350
RCS TEMPERATURE (°F)

FIGURE 3.1.A-2
MAXIMUM PRESSURIZER LEVEL WITH PORVs
INOPERABLE AND ONE CHARGING PUMP ENERGIZED

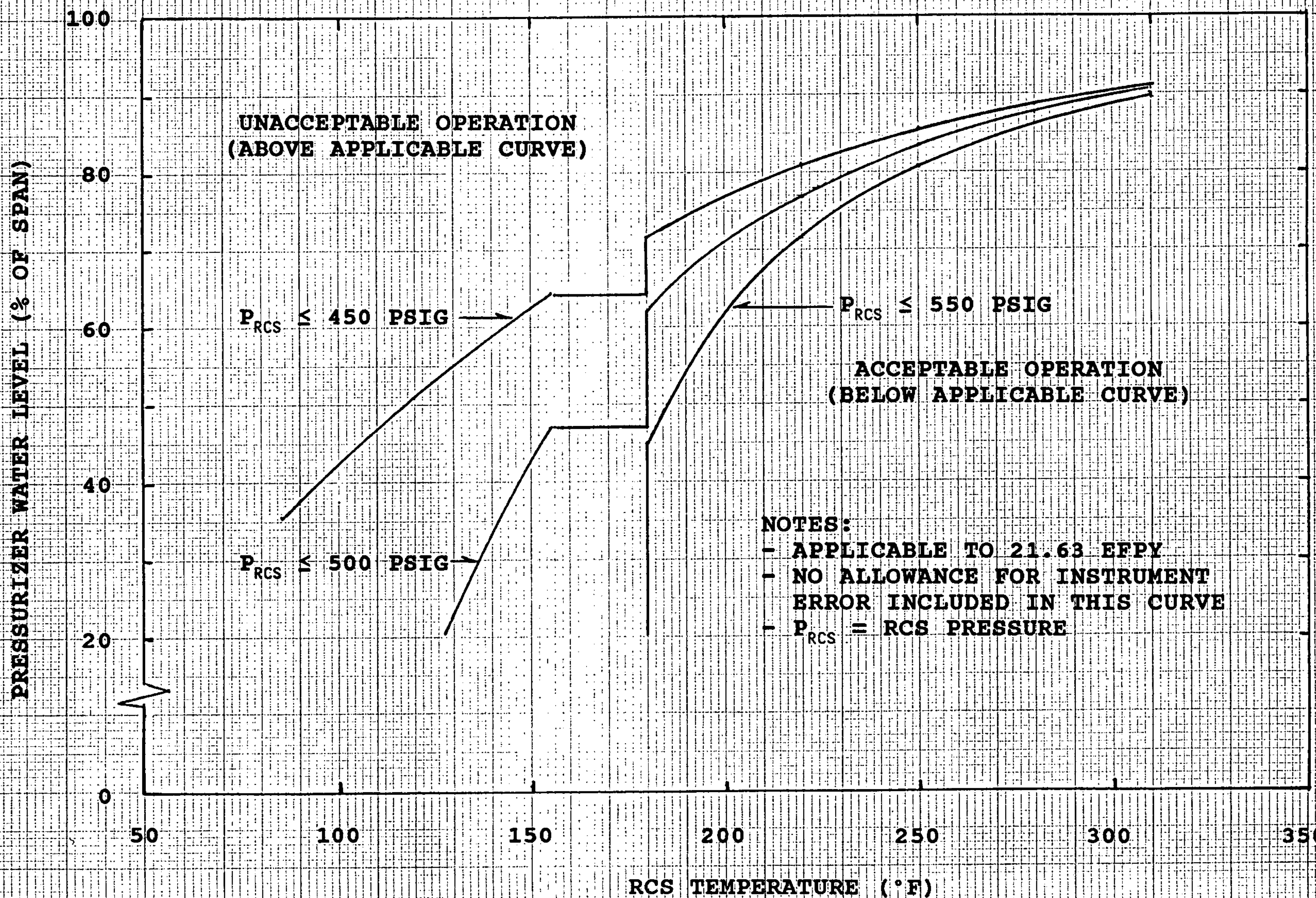
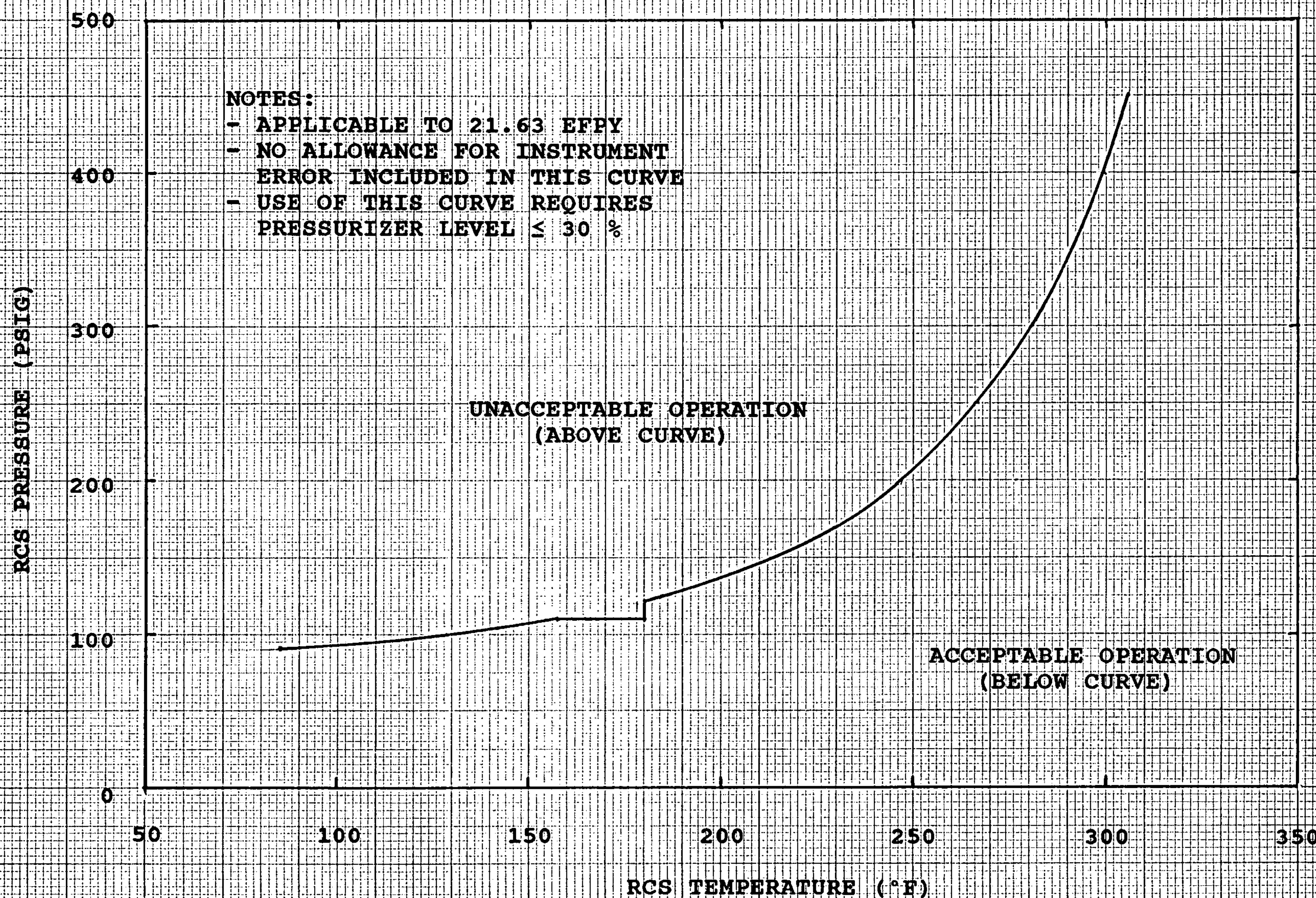


FIGURE 3.1.A-3
MAXIMUM REACTOR COOLANT SYSTEM PRESSURE FOR OPERATION WITH PORVs INOPERABLE AND ONE SAFETY INJECTION PUMP AND/OR THREE CHARGING PUMPs ENERGIZED



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 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

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* 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.

post-irradiation tensile and Charpy V notch (wedge open loading) testing of specimens. 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/cm^2 at the inner reactor vessel surface (45° angle, vessel belt line). This fluence is based upon plant operaton 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, 1974 Edition, Section III, Appendix G, and the calculation methods described in WCAP-7924A[4] and WCAP-12796.

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].

The maximum shift in RT_{NDT} at a fluence of 0.98 X 10^{19} n/cm^2, (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 III 1974 Edition of the ASME Boiler and Pressure Vessel Code and are discussed in detail in WCAP-7924A[4] and WCAP-12796.

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 I.D. 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 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 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. 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.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.

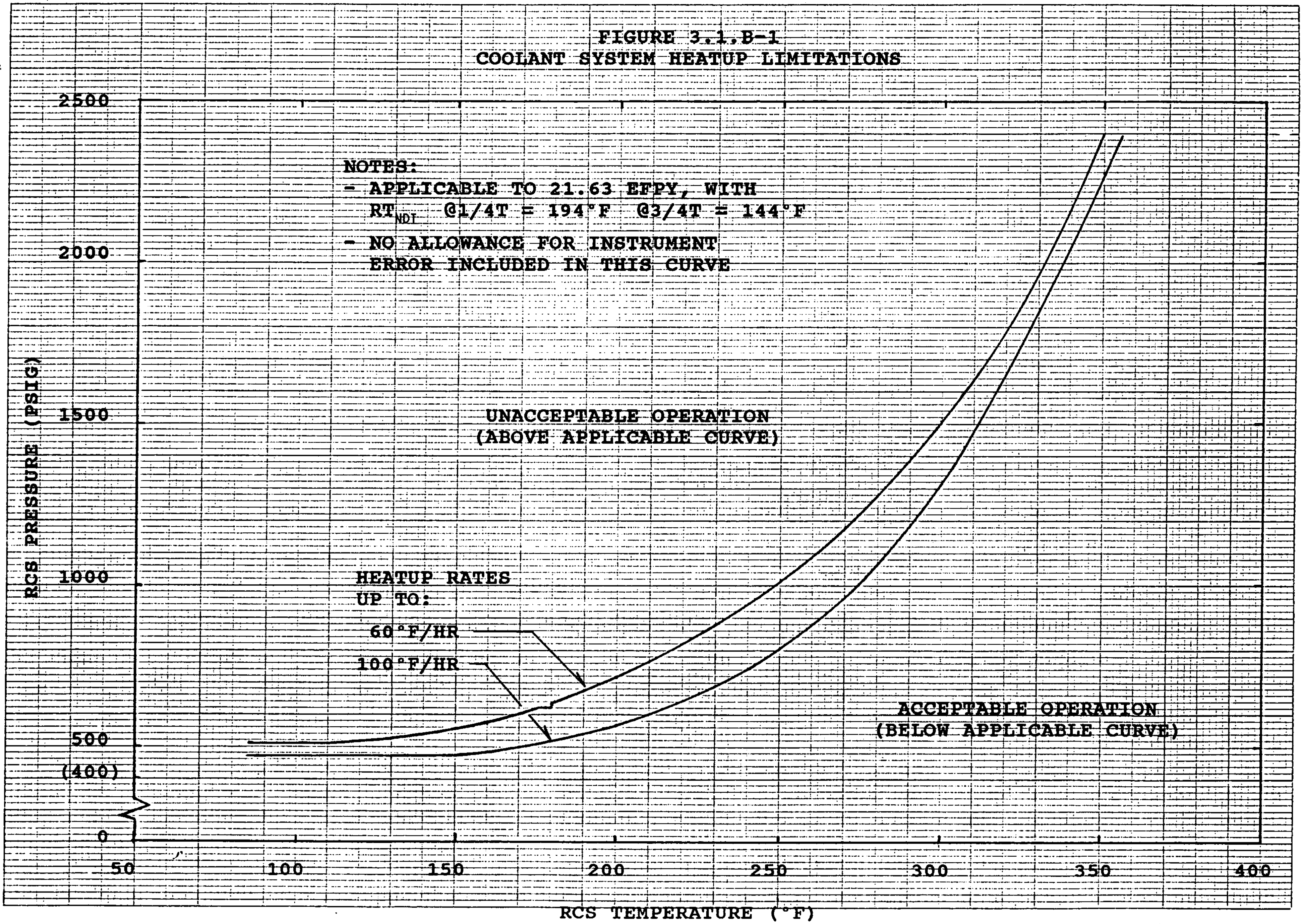


FIGURE 3.1.B-1
COOLANT SYSTEM HEATUP LIMITATIONS
NOTES:
- APPLICABLE TO 21.63 EFPY, WITH
RT_{NDT} @1/4T = 194°F @3/4T = 144°F
- NO ALLOWANCE FOR INSTRUMENT
ERROR INCLUDED IN THIS CURVE
UNACCEPTABLE OPERATION
(ABOVE APPLICABLE CURVE)
HEATUP RATES
UP TO:
60°F/HR
100°F/HR
ACCEPTABLE OPERATION
(BELOW APPLICABLE CURVE)
RCS PRESSURE (PSIG)
2500
2000
1500
1000
500
(400)
0
50
100
150
200
250
300
350
400
RCS TEMPERATURE (°F)

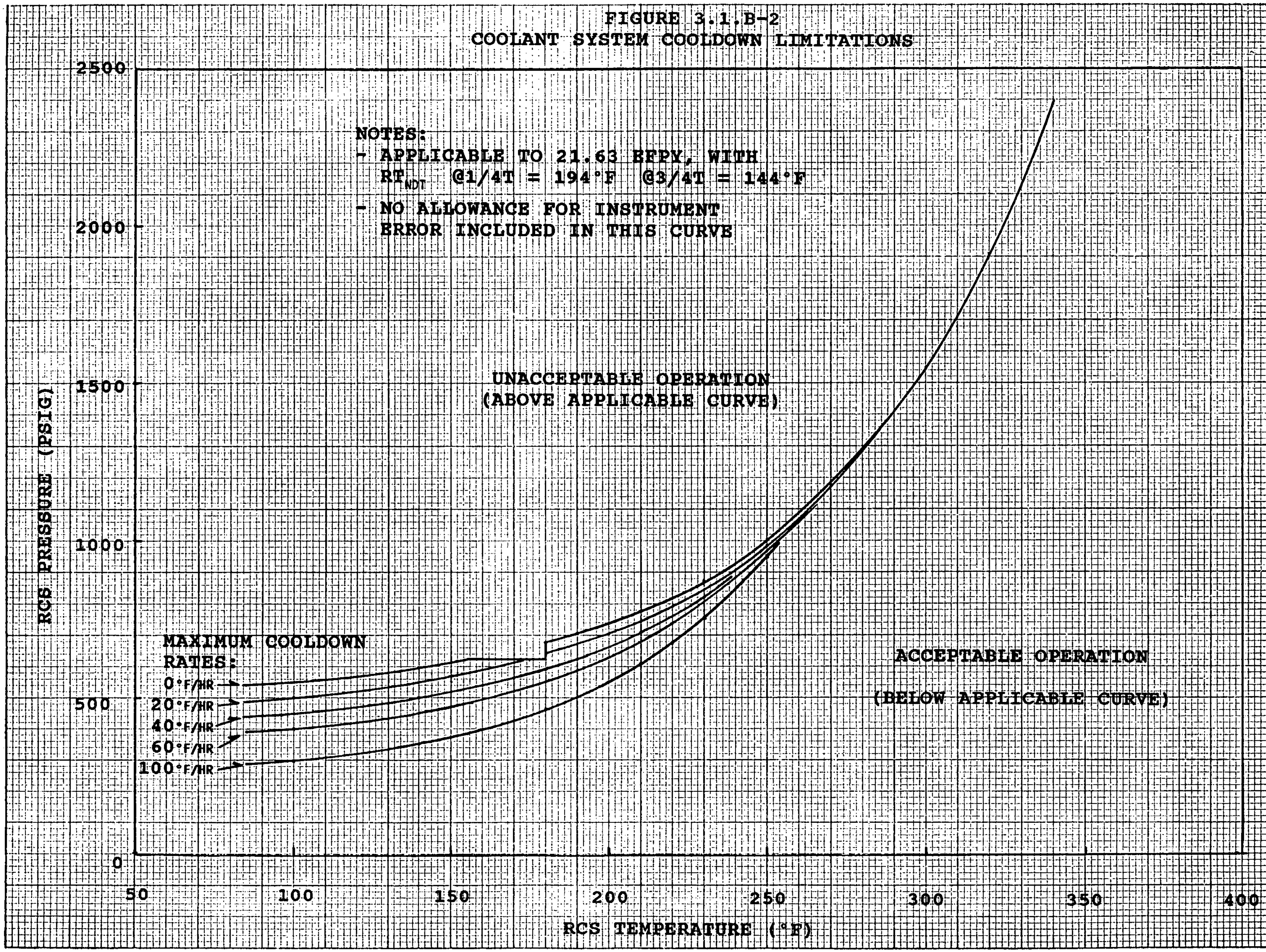


FIGURE 3.1.B-2
COOLANT SYSTEM COOLDOWN LIMITATIONS
NOTES:
- APPLICABLE TO 21.63 EFPY, WITH
RT_{NDT} @1/4T = 194°F @3/4T = 144°F
- NO ALLOWANCE FOR INSTRUMENT
ERROR INCLUDED IN THIS CURVE
UNACCEPTABLE OPERATION
(ABOVE APPLICABLE CURVE)
MAXIMUM COOLDOWN
RATES:
0°F/HR
20°F/HR
40°F/HR
60°F/HR
100°F/HR
ACCEPTABLE OPERATION
(BELOW APPLICABLE CURVE)
RCS PRESSURE (PSIG)
2500
2000
1500
1000
500
0
50
100
150
200
250
300
350
400
RCS TEMPERATURE (°F)

C. MINIMUM CONDITIONS FOR CRITICALITY

Specifications

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 450°F.

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)]. 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 negative. At all times, the moderator coefficient is negative in the power operating range[(1)]. 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 lower 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 450°F 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 10 CFR 50 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) UFSAR Section 3.2

3.2 CHEMICAL AND VOLUME CONTROL SYSTEM

Applicability

Applies to the operational status of the Chemical and Volume Control System.

Objective

To define those conditions of the Chemical and Volume Control System necessary to ensure safe reactor operation.

Specifications

A. When fuel is in the reactor there shall be at least one flow path to the core for boric acid injection.

B. The reactor shall not be made critical unless the following Chemical and Volume Control System conditions are met:

1. Two charging pumps shall be operable.

2. The boric acid storage system shall contain a minimum of 6000 gallons of 11 1/2% to 13% by weight (20,000 ppm to 22,500 ppm of boron) boric acid solution at a temperature of at least 145°F, and at least one boric acid transfer pump shall be operable.

3. System piping and valves shall be operable to the extent of establishing one flow path from the boric acid storage system and one flow path from the refueling water storage tank (RWST) to the Reactor Coolant System.

4. Two channels of heat tracing shall be operable for the flow path from the boric acid storage system.

C. During power operation, the requirements of 3.2.B may be modified to allow any one of the following components to be inoperable. If the system is not restored to meet the requirements of 3.2.B within the time period specified, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the requirements of 3.2.B are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

1. One of the two operable charging pumps may be removed from service provided a second charging pump is restored to operable status within 24 hours.

2. The boric acid storage system (including the boric acid transfer pumps) may be inoperable provided the RWST is operable and provided that the boric acid storage system and at least one boric acid transfer pump is restored to operable status within 48 hours.

3. One channel of heat tracing for the flow path from the boric acid storage system to the Reactor Coolant System may be out of service provided the failed channel is restored to an operable status within 7 days and the redundant channel is demonstrated to be operable daily during that period.

4. Both channels of heat tracing for the flow path from the boric acid storage system to the Reactor Coolant System may be out of service provided at least one channel is restored to operable status within 48 hours, the required flow path is shown to be clear of blockage, and the second channel is restored to operable status within 7 days.

D. When RCS temperature is less than or equal to 305°F, the requirements of Table 3.1.A-2 regarding the number of charging pumps allowed to be energized shall be adhered to.

3.3 ENGINEERED SAFETY FEATURES

Applicability

Applies to the operating status of the Engineered Safety Features.

Objective

To define those limiting conditions for operation that are necessary (1) to remove decay heat from the core in emergency or normal shutdown situations, (2) to remove heat from containment in normal operating and emergency situations, (3) to remove airborne iodine from the containment atmosphere following a Design Basis Accident, (4) to minimize containment leakage to the environment subsequent to a Design Basis Accident.

Specifications

The following specifications apply except during low-temperature physics tests.

A. SAFETY INJECTION AND RESIDUAL HEAT REMOVAL SYSTEMS

1. The reactor shall not be made critical except for low-temperature physics tests, unless the following conditions are met:

    a. The refueling water storage tank contains not less than 345,000 gallons of water with a boron concentration of at least 2000 ppm.

    b. Deleted

    c. The four accumulators are pressurized to at least 615 psig and each contains a minimum of 787.5 ft^3 and a maximum of 802.5 ft^3 of water with a boron concentration of at least 2000 ppm. None of these four accumulators may be isolated.

d. Three safety injection pumps together with their associated piping and valves are operable.

e. Two residual heat removal pumps and heat exchangers together with their associated piping and valves are operable.

f. Two recirculation pumps together with the associated piping and valves are operable.

g. Valves 842 and 843 in the mini-flow return line from the discharge of the safety injection pumps to the RWST are de-energized in the open position.

h. Valves 856A, C, D and E, in the discharge header of the safety injection header, are in the open position. Valves 856B and F, in the discharge header of the safety injection header, are in the closed position. The hot-leg valves (856B and F) shall be closed with their motor operators de-energized by locking out the circuit breakers at the Motor Control Centers.

i. The four accumulator isolation valves shall be open with their motor operators de-energized by locking out the circuit breakers at the Motor Control Centers.

j. Valve 1810 on the suction line of the high-head SI pumps and valves 882 and 744, respectively on the suction and discharge line of the residual heat removal pumps, shall be blocked open by de-energizing the valve-motor operators.

k. The refueling water storage tank low-level alarms are operable and set to alarm between 74,200 gallons and 99,000 gallons of water in the tank.

2. During power operation, the requirements of 3.3.A.1 may be modified to allow any one of the following components to be inoperable at any one time. If the system is not restored to meet the requirements of 3.3.A.1

within the time period specified, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the requirements of 3.3.A.1 are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

a. One safety injection pump may be out of service, provided the pump is restored to operable status within 24 hours and the remaining two pumps are demonstrated to be operable.

b. One residual heat removal pump may be out of service, provided the pump is restored to operable status within 24 hours and the other residual heat removal pump is demonstrated to be operable.

c. One residual heat removal heat exchanger may be out of service provided that it is restored to operable status within 48 hours.

d. Any valve required for the functioning of the system during and following accident conditions may be inoperable provided that it is restored to operable status within 24 hours and all valves in the system that provide the duplicate function are demonstrated to be operable.

e. Deleted

f. One refueling water storage tank low-level alarm may be inoperable for up to 7 days provided the other low-level alarm is operable.

3. When RCS temperature is less than or equal to 305°F, the requirements of Table 3.1.A-2 regarding the number of safety injection (SI) pumps allowed to be energized shall be adhered to.

a single failure. Valve 744 will not need to be closed following the injection phase. The accumulator isolation valve motor operators are de-energized to prevent an extremely unlikely spurious closure of these valves from occurring when accumulator core cooling flow is required.

With respect to the core cooling function, there is some functional redundancy for certain ranges of break sizes(3). The measure of effectiveness of the Safety Injection System is the ability of the pumps and accumulators to keep the core flooded or to reflood the core rapidly where the core has been uncovered for postulated large area ruptures. The result of the performance is to sufficiently limit any increase in clad temperature below a value where emergency core cooling objectives are met(9). The range of core protection as a function of break diameter provided by the various components of the Safety Injection System is presented in Figure 6.2-9 of the UFSAR.

The requirement regarding the maximum number of SI pumps that can be energized when RCS temperature is less than or equal to 305°F is discussed under Specification 3.1.A.

The containment cooling and iodine removal functions are provided by two independent systems: (1) fan-coolers plus charcoal filters and (2) containment spray with sodium hydroxide addition. During normal power operation, the five fan-coolers are required to remove heat lost from equipment and piping within containment at design conditions (with a cooling water temperature of 95°F)(12). In the event of a Design Basis Accident, any one of the following combinations will provide sufficient cooling to reduce containment pressure at a rate consistent with limiting offsite doses to acceptable values: (1) five fan-cooler units, (2) two containment spray pumps, (3) three fan-cooler units and one spray pump. Also, in the event of a Design Basis Accident, three charcoal filters (and their associated recirculation fans) in operation, along with one containment spray pump and sodium hydroxide addition, will reduce airborne organic and molecular iodine activities sufficiently to limit offsite doses to acceptable values. These constitute the minimum safeguards for iodine removal, and are capable of being operated on emergency power with one diesel generator inoperable.

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.

Specifications

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 21.63 effective full-power years 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 does not leak at 2335 psig (Operating pressure + 100 psi: ± 100 psi is normal system pressure fluctuation), it will be leak-tight during normal operation.

For repairs on components, the thorough non-destructive testing gives a very high degree of confidence in the integrity of the system, and will detect any significant defects in and near the new welds. In all cases, the leak test will assure leak-tightness during normal operation.

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

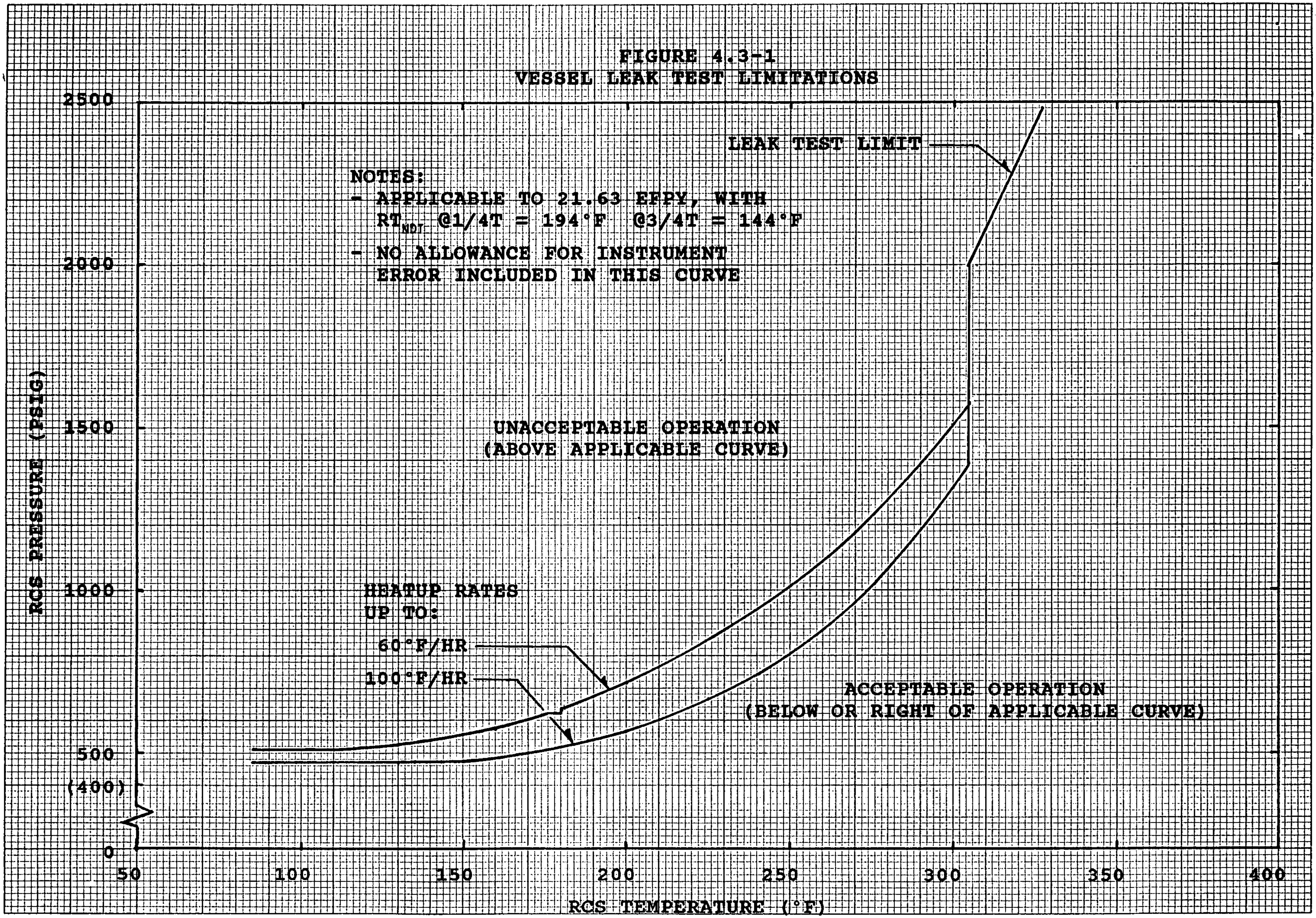
For the first 21.63 effective full-power years, it is predicted that the highest RT_{NDT} in the core region taken at the 1/4 thickness will be 194°F. The minimum inservice leak test temperature requirements for periods up to 21.63 effective full-power years are shown on Figure 4.3-1.

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 WCAP-12796 and results of surveillance specimen testing, as covered in WCAP-7323.

The current heatup and cooldown curves are based upon a maximum fluence of 0.98×10^{19} n/cm^2 at the inner reactor vessel surface (45° angle, vessel belt line). This fluence is based upon plant operaton 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.

Reference

UFSAR Section 4



20 Squares to the Inch

ATTACHMENT II

SAFETY ASSESSMENT

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247
MARCH, 1991

SECTION I - Description of Changes

This application for amendment to the Indian Point 2 Technical Specifications seeks to amend Section 3.1.A.1 (Reactor Coolant Pump), Section 3.1.A.4 (Overpressure Protection System), Section 3.1.B (Heatup and Cooldown), Section 3.1.C (Minimum Conditions for Criticality), Section 3.2.D (CVCS), Section 3.3.A.3 (Safety Injection), and Section 4.3 (Reactor Coolant System Integrity Testing). These changes are being made to incorporate revised pressure-temperature limits and Overpressure Protection System (OPS) parameters. These revisions are being made in accordance with Reference 1, which required that licensees use the methodology of Regulatory Guide (RG) 1.99 Revision 2, "Radiation Embrittlement of Reactor Vessel Material", to predict the effect of neutron radiation on reactor vessel materials. Also, Section 3.1.C is being amended to replace pressure-temperature requirements on the reactor coolant system when the reactor is critical, with a fixed temperature limit.

The proposed changes are specified in Attachment I to the Application for Amendment enclosed with this letter. Attachment III "Heatup and Cooldown Limit Curves for the Consolidated Edison Company Indian Point Unit 2 Reactor Vessel," WCAP-12796 describes the methodology used in developing the pressure-temperature curves. Attachment IV, Revised OPS Setpoints For Indian Point Unit 2 describes the methods used in developing the OPS curves and tables.

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 2. The heatup curves cover a range of heatup rates from 20°F/hr. through 100°F/hr., and the cooldown curves a range from 0°F/hr. (isothermal) to 100°F/hr.

The calculations supporting these curves incorporate data from analysis of Indian Point 2 surveillance capsules V, T, Y, and Z.

Section 3.1.C.2 of the Technical Specifications is being replaced by a fixed temperature limit which bounds the allowed pressure-temperature range for criticality.

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 an accident previously analyzed is increased due to the proposed changes. The adjusted reference temperature of the most limiting beltline material was used to correct the pressure-temperature (P-T) curves to account for irradiation effects. Thus, the operating limits are adjusted to incorporate both the initial fracture toughness conservatism present when the reactor vessel was new and the effect of fluence. The adjusted reference temperature calculations were performed utilizing the guidance contained in RG 1.99, Revision 2. Overpressure Protection System (OPS) Curves and Tables were regenerated to be consistent with the new P-T curves. 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 the proposed new specification 3.1.C.2.

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

Response

The updated P-T and OPS 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 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 the margin of safety?

Response:

The proposed amendment does not involve a significant reduction in the margin of safety. The pressure-temperature operating limits and OPS setpoints are designed to maintain an appropriate 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 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 of 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 requirements of the proposed Specification 3.1.C.2.

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 or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report; b) will not create the possibility for an accident or malfunction of a new or different kind from 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.

These proposed changes have been reviewed by the Station Nuclear Safety Committee and the Nuclear Facility Safety Committee.

ATTACHMENT III

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247
MARCH, 1991

ATTACHMENT B

SAFETY ASSESSMENTS

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247
FEBRUARY, 1994

SAFETY ASSESSMENT

ANALOG ROD POSITION INDICATION

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

DESCRIPTION OF CHANGE

Technical Specification 4.1, Table 4.1-1, item 9, requires calibration of the analog rod position instrumentation channel during each refueling interval. Currently, this calibration is performed every 18 months (+25%). It is proposed that this calibration frequency be revised to every 24 months (+25%). This change is being made in accordance with the guidance contained in Generic Letter 91-04.

The Analog Rod Position Indication (ARPI) system provides a continuous, visual indication of the control rod positions. The Bank Demand Step Counter provides a reference for the purpose of comparison, i.e., where the Rod Control System has required that the bank be placed. This reference is established by a digital step counter responding to each rod withdrawal or insertion step demand. The ARPI provides an indication via the display of the rod position as a function of change in detector output voltage generated by the insertion or withdrawal of the control rod drive shaft through the detector coil stack.

Test data was reviewed from 1986, 1987, 1988, 1989 and 1991. Overall, the channel was found and left within calibration tolerance and was therefore in an operable status throughout the surveillance interval. A review of known error contributors lead to the conclusion that none of the major contributors exhibit time dependent effects. Thus, it is concluded that an extension of the surveillance interval from 18 (+25%) to 24 (+25%) months would not result in any significant additional error for this system.

Basis for No Significant Hazards Considerations Determination

The proposed change does not involve a significant hazards consideration since:

1. A significant increase in the probability or consequences of an accident previously evaluated will not occur.

   It is proposed that the channel calibration frequency for the analog rod position indication channel be changed from every 18 months (+25%) to every 24 months (+25%).

   A statistical analysis of channel uncertainty for a 30 month operating cycle has been performed. Based upon this analysis it has been concluded that none of the major error contributors are time dependent and that it can be reasonably expected that the channel will remain within calibration tolerance over a possible 30 month operating cycle. In addition, the rod bottom bistable is subject to monthly testing which would detect any abnormalities in an extended operating cycle. Due to this monthly test and the acceptable past test history, it is concluded that the channel will continue to operate within tolerance over an extended operating cycle and will not contribute to a significant increase in the probability or consequences of an accident previously evaluated.

2. The possibility of a new or different kind of accident from any accident previously evaluated has not been created.

The proposed change in operating cycle length due to an increased surveillance interval is not expected to affect the ability of the instrument channel to remain within calibration tolerance. Furthermore, the rod position indicator is used in normal operation only as an aid in control rod movement. Normally, very little control rod movement occurs during normal operation. Furthermore, it is not relied upon for accident prevention or accident mitigation. In accordance with existing Technical Specifications, normal operation can continue even if one channel is inoperable because alternate means (core instrumentation) exists to monitor rod position. The frequent monthly test tends to minimize the effect of a longer operating cycle for the rod position indication channel as any malfunction induced by time would be detected. Thus, it is concluded that the possibility of a new or different kind of accident from any accident previously evaluated has not been created.

3. A significant reduction in a margin of safety is not involved.

A statistical analysis of past calibration data has not identified any time dependent error contributors. Also, past test data indicates that the channel remains within calibration tolerance over the existing operating cycle. A longer operating cycle would increase the risk of drift, however accuracy is not a prime requirement for the RPI. Therefore, it is concluded that a longer operating cycle will not result in a significant reduction in a margin of safety.

SAFETY ASSESSMENT

PLANT NOBLE GAS ACTIVITY MONITOR
(R-44)

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

DESCRIPTION OF CHANGE

Technical Specification Table 4.10-4, item 4.a, specifies that the Plant Vent Noble Gas Activity Monitor shall be calibrated during each refueling interval. This assessment refers to R-44. Currently, this calibration is performed on an 18 month (+25%) basis. It is proposed that the surveillance interval be extended to 24 months (+25%). This change is being proposed in accordance with the guidance contained in Generic Letter 91-04.

The plant vent is continuously sampled for both particulate activity and radioactive gas by radiation monitors R-43 and R-44. These series connected monitors take a continuous air sample from the plant vent, measure the activity, and return the air back to the plant vent. The particulate and iodine monitors filter the air sample through a continuously moving paper filter and charcoal canister and measure the gamma radioactivity of the filters. The gas monitor, R-44, measures the gamma radioactivity of the filtered air. High radiation detected by R-44 initiates closure of the containment purge supply and exhaust duct valves, the pressure relief line valves, and the gas release valve in the waste disposal system. This monitor also initiates an alarm in the control room when its setpoint is exceeded.

Technical Specification Table 3.9-2, item 4.a, requires that the Plant Vent Noble Gas Activity Monitor be operable during release via this pathway and also when the waste gas holdup system is in operation. The action required, in the event that the monitor becomes inoperable, is suspension of the release of radioactive effluents from the waste gas holdup system. However, with the monitor inoperable, the contents of the tank(s) may be released provided that, prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. at least two technically qualified members of the facility staff independently verify the release rate calculations and discharge valve lineup.

These detectors monitor potentially radioactive gaseous effluent through the plant vent. The alarm setpoints are set by the operators at a point as close as possible to the steady state values. This setpoint is always low enough to provide adequate warning and closure of the valves before the allowed concentration is reached. The setpoint is normally several decades below the allowed instantaneous release limit.

DESCRIPTION OF CHANGE

Technical Specification Table 4.10-4, item 4.a, specifies that the Plant Vent Noble Gas Activity Monitor shall be calibrated during each refueling interval. This assessment refers to R-44. Currently, this calibration is performed on an 18 month (+25%) basis. It is proposed that the surveillance interval be extended to 24 months (+25%). This change is being proposed in accordance with the guidance contained in Generic Letter 91-04.

The plant vent is continuously sampled for both particulate activity and radioactive gas by radiation monitors R-43 and R-44. These series connected monitors take a continuous air sample from the plant vent, measure the activity, and return the air back to the plant vent. The particulate and iodine monitors filter the air sample through a continuously moving paper filter and charcoal canister and measure the gamma radioactivity of the filters. The gas monitor, R-44, measures the gamma radioactivity of the filtered air. High radiation detected by R-44 initiates closure of the containment purge supply and exhaust duct valves, the pressure relief line valves, and the gas release valve in the waste disposal system. This monitor also initiates an alarm in the control room when its setpoint is exceeded.

Technical Specification Table 3.9-2, item 4.a, requires that the Plant Vent Noble Gas Activity Monitor be operable during release via this pathway and also when the waste gas holdup system is in operation. The action required, in the event that the monitor becomes inoperable, is suspension of the release of radioactive effluents from the waste gas holdup system. However, with the monitor inoperable, the contents of the tank(s) may be released provided that, prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. at least two technically qualified members of the facility staff independently verify the release rate calculations and discharge valve lineup.

These detectors monitor potentially radioactive gaseous effluent through the plant vent. The alarm setpoints are set by the operators at a point as close as possible to the steady state values. This setpoint is always low enough to provide adequate warning and closure of the valves before the allowed concentration is reached. The setpoint is normally several decades below the allowed instantaneous release limit.

The installed monitoring systems are not designed to determine the nature and amount of radioactivity in the system being monitored. The systems monitor gross activity and are designed to generate alarms. R-44 causes automatic closure of the containment purge supply and exhaust duct valves, the pressure relief line valves, and the gas release valve in the waste disposal system under abnormal conditions. Isotopic identification and concentrations are determined by grab sample analysis.

These monitors have no setpoints which are critical to plant operation or safety, nor are their readings used in calculations which require discrete accuracy. Their prime functions are to provide indication of changing radiation levels, to provide alarms on high radiation, and to close the containment purge supply and exhaust duct valves, the pressure relief line valves, and the gas release valve in the waste disposal system in the event of high radiation levels in the plant vent. Actual setpoints for the alarms and the valve closure functions are not critical to either plant operation or calculation of released radioactive material.

BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The proposed change does not involve a significant hazards consideration since:

1. There is no significant increase in the probability or consequences of an accident.

   It is proposed that the channel calibration frequency for the Plant Nobel Gas Activity Monitor (R-44) be changed from every 18 months (+25%) to every 24 months (+25%).

   The function of R-44 is to respond to high activity levels during normal operation.

   The setpoint for R-44 is established sufficiently above the expected radioactivity level in the discharge stream to preclude false actions but sufficiently below the allowed discharge radioactivity concentration so that discharge in excess of permissible limits does not occur. Monitor readouts are not used for quantitative purposes, but are used to respond to relative changes in radioactivity concentration.

   There is limited data to support an unqualified extension of the surveillance interval. However, the instrument is checked for operability prior to release. Should the instrument be inoperable releases may continue provided grab sample analysis is performed. Since the monitor is subject to daily channel checks, monthly source checks, and quarterly functional channel tests, abnormal instrument behavior or inoperability would be detected permitting corrective actions during the extended surveillance interval.

2. The possibility of a new or different kind of accident from any previously analyzed has not been created.

   Operability of the instrument is important rather than ability to maintain a specific setpoint. Operability of the instrument is verified prior to a planned discharge and this is independent of an extended surveillance cycle.

3. There has been no reduction in the margin of safety.

   As the Technical Specifications permit pre-planned release even with an inoperable instrument, the margin of safety is not impacted by an extended surveillance interval provided that instrument operability is verified prior to release. This is also required by the Technical Specifications.

SAFETY ASSESSMENT
LOW TURBINE AUTO STOP OIL PRESSURE REACTOR TRIP

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

DESCRIPTION OF CHANGE

Technical Specification Table 4.1-1, item 27, requires that the Low Turbine Auto Stop Oil Pressure system be calibrated at each refueling interval. Currently, this surveillance is performed every 18 months (+25%). It is proposed that this surveillance frequency be revised to every 24 months (+25%). This change is being made in accordance with the guidance contained in Generic Letter 91-04.

The Low Turbine Auto Stop Oil Pressure system provides a reactor trip on turbine trip above 35% power and provides protection from a load rejection in excess of the capability of the Steam Dump System. A turbine trip from a power level above the capability of the Steam Dump System will actuate a trip to minimize the pressure/temperature transient on the reactor. The safety analyses do not assume the operation of this function. A turbine trip signal energizes and opens the main turbine automatic trip solenoids, 20AST and 20ASB. When these valves open, the turbine hydraulic oil system is dumped. Pressure switches 63/AST-2, 63/AST-3, and 63/AST-4 sense the sudden loss of hydraulic oil pressure and trip the turbine, which in turn trips the Reactor. These switches are calibrated every refueling outage.

Completed test results were reviewed from the last four refueling cycles. These tests spanned a period in excess of four years. In all tests the pressure switches actuated as required.

The pressure switches associated with the Low Turbine Auto Stop Oil Pressure System are generally reliable devices. Because these devices are a Go/No-Go type operator rather than an analog sensor, and the fact that these devices are not used in the safety analyses as a primary trip for accident mitigation, extension of the surveillance interval from 18 months to 24 months for this test would not degrade the reliability of the system.

BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The proposed change does not involve a significant hazards consideration since:

1. There is no significant increase in the probability or consequences of an accident.

   It is proposed that the channel calibration frequency for the Low Turbine Auto Stop Oil Pressure system be changed from every 18 months (+25%) to every 24 months (+25%).

   No credit is taken for a reactor trip from a low turbine auto stop oil pressure signal resulting from a turbine trip. Rather, the safety analysis assumes this reactor trip does not occur during full load rejection until an overpower delta T condition causes a reactor trip. In addition, no credit is taken for this system for turbine missile protection. Therefore, extending the surveillance interval for this parameter has no impact upon the probability or consequences of an accident.

2. The possibility of a new or different kind of accident from any previously analyzed has not been created.

   As no credit is taken in the safety analysis for this trip, the possibility of a new or different kind of accident has not been created by extending the surveillance interval.

3. There has been no reduction in the margin of safety.

   Past test results have not identified any failures. Therefore, pursuant to Generic Letter 91-04, it is reasonably expected that this system will continue to function in an acceptable manner over an extended operating cycle.

SAFETY ASSESSMENT

6.9 KV UNDERVOLTAGE RELAYS

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

DESCRIPTION OF CHANGE

Technical Specification Table 4.1-1, Item 8, requires that a calibration of the 6.9 kv undervoltage channel be performed every 18 months (+25%). It is proposed that the surveillance frequency be changed to every 24 months (+25%). This change is being proposed in accordance with the guidance contained in Generic Letter 91-04.

The current Indian Point Unit 2 Technical Specification also requires that the 6.9 kv undervoltage channel trip setpoint be set at $\geq$ 70% of nominal voltage (Specification 2.3.1.B.7) and that a channel functional test be performed quarterly (Table 4.1-1 item #8).

During the 1993 Refueling Outage, the degraded voltage relays were replaced by ASEA Brown Boveri type 27 N (solid state, high accuracy devices) whose long term performance is expected to be superior to the previously installed relays. A decision to use a drift value equivalent to the calibration accuracy of the relay per year was made. Consolidated Edison will continue to trend instrument data obtained from future calibrations to verify that this assumed drift value is not being exceeded and that the current Technical Specification limit remains valid, or perform a new analysis as appropriate. Technical Specifications require a quarterly test whose data will be used for trending purposes.

Since the 6.9 kv undervoltage relays were replaced with ABB type 27N Relays during the 1993 outage, a drift value of 3.0% span per year (7.5% for 30 months) was determined as appropriate for this evaluation based on a maximum allowed drift equal to the calibration accuracy per year. This drift value was used as an input to determine the Channel Statistical Allowance (CSA) using the Westinghouse setpoint methodology. Included in the evaluation along with instrument drift is the determination of all other channel uncertainties including Sensor, Rack, M&TE, and Process Effects for normal environmental conditions.

BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The proposed change does not involve a significant hazards consideration since:

1. There is no significant increase in the probability or consequences of an accident.

   It is proposed that the calibration frequency for the 6.9 kv undervoltage channel be changed form every 18 months (+25%) to every 24 months (+25%).

   Quarterly testing of these relays is required by Technical Specifications. The data from the quarterly tests of the new relays will be used to assure that drift does not exceed projected values. The quarterly tests provide a means of maintaining calibration within specified values, viturally eliminating any impact upon safety from an extended operating cycle.

2. The possibility of a new or different kind of accident from any previously analyzed has not been created.

   Because the quarterly tests assure that relay performance remains within specified limits, there is no possibility of creating a new or different kind of accident from any previously analyzed.

3. There has been no reduction in the margin of safety.

   The requirement for a channel functional test each quarter minimizes any potential impact upon safety due to an extended operating cycle.

SAFETY ASSESSMENT

BORIC ACID TANK LEVEL

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

DESCRIPTION OF CHANGE

Technical Specification Table 4.1-1, item #14, requires a channel calibration be performed of the Boric Acid Tank Level at refueling intervals. Currently, this surveillance is performed at 18 month (+25%) intervals. It is proposed to change the surveillance intervals to 24 months (+25%). This change is being made in accordance with Generic Letter 91-04.

The current Indian Point Unit 2 Technical Specification governing the Boric Acid Tank requires that a minimum of 6000 gallons of 11½ to 13% boric acid at a minimum temperature of 145°F be available for boric acid injection (Specification 3.2.B.2).

The Boric Acid Tank Level Channel was reviewed using the Westinghouse methodology for evaluating channel uncertainties. Each uncertainty term was determined according to the instrument characteristics/specifications. Particular effort was made to predict a drift for the instrumentation over a 30 month period based on a statistical evaluation of plant recorded "As Left/As Found" data taken at the site since 1985. Past cycle calibration data was evaluated to determine how well the instruments had performed from one cycle to the next. This evaluation included a review of any work order data that may have been taken during a midcycle outage etc., or any modification to the channels. Also, past M&TE accuracies were reviewed to insure that the M&TE used was of an equivalent accuracy such that it would not have biased the data in a non-conservative direction. In addition to drift, process measurement accuracy terms such as density and temperature effects, along with tank tolerance, have been incorporated into the total channel uncertainty calculations.

"As Left/As Found" data from the 1985 outage to the present was evaluated for determination of the population normality and outliers. Where possible, outliers have been eliminated by use of accepted statistical tests or mechanistic causes were determined to justify elimination. Thus, the drift value determined does not exceed any assumptions of the safety analysis or affect this channels capability of performing its safety function. The drift values utilized in the uncertainty analysis have been determined with a 95% probability at a 75% confidence level.

BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The proposed change does not involve a significant hazards consideration since:

1. A significant increase in the probability or consequences of an accident previously evaluated will not occur.

   It is proposed that the channel calibration frequency for the Boric Acid Tank Level instrumentation be changed from every 18 months (+25%) to every 24 months (+25%).

   A statistical analysis of channel uncertainty for a 30 month operating cycle has been performed. Based upon this analysis it has been concluded that sufficient margin exists between the existing Technical Specification limit and the licensing basis Safety Analysis limit to accommodate the channel statistical error resulting from a 30 month operating cycle. The existing margin between the Technical Specification limit and the Safety Analysis limit provides assurance that plant protective actions will occur as required. It is therefore concluded that changing the surveillance interval from 18 months (+25%) to 24 months (+25%) will not result in a significant increase in the probability or consequences of an accident previously evaluated.

2. The possibility of a new or different kind of accident from any accident previously evaluated has not been created.

   The proposed change in operating cycle length due to an increased surveillance interval will not result in a channel statistical allowance which exceeds the current margin between the existing Technical Specification limit and the Safety Analysis limit. Plant equipment, which will be set at (or more conservatively than) Technical Specification limits, will provide protective functions to assure that Safety Analysis limits are not exceeded. This will prevent the possibility of a new or different kind of accident from any previously evaluated from occurring.

3. A significant reduction in a margin of safety is not involved.

   The above change in surveillance interval resulting from an increased operating cycle will not result in a channel statistical allowance which exceeds the margin which exists between the current Technical Specification limit and the licensing basis Safety Analysis limit. This margin, which is equivalent to the existing margin, is necessary to assure that protective safety functions will occur so that Safety Analysis limits are not exceeded.

SAFETY ASSESSMENT

VC SUMP DISCHARGE FLOW AND TEMPERATURE CHANNEL

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

DESCRIPTION OF CHANGE

The current Indian Point Unit 2 Technical Specification requires that the VC Sump Discharge Flow and Temperature Channel be calibrated every refueling interval (Table 4.1-1, item #21.d). Currently, this calibration is performed every 18 months (+25%). It is proposed that this calibration frequency be revised to every 24 months (+25%). This change is being made in accordance with the guidance contained in Generic Letter 91-04.

The VC Sump Discharge Flow and Temperature Channel was reviewed using the Westinghouse methodology for evaluating channel uncertainties. Each uncertainty term was determined according to the instrument characteristics / specifications. Past M&TE accuracies were reviewed to insure that the M&TE used was of an equivalent accuracy such that it would not have biased the data in a non-conservative direction. In addition to drift, sensor, rack and M&TE terms have been incorporated into the total channel uncertainty calculations.

Using the above mentioned criteria and the Westinghouse methodology for evaluating channel uncertainties, a total channel uncertainty has been calculated to reflect a 30 month surveillance period.

It is concluded that the surveillance interval can be extended to 30 months with no other change to the Technical Specification or Safety Analysis.

Basis for No Significant Hazards Considerations

The proposed change does not involve a significant hazards consideration since:

1. A significant increase in the probability or consequences of an accident previously evaluated will not occur.

   It is proposed that the calibration frequency for the VC sump discharge flow and temperature channel be changed from every 18 months (+25%) to every 24 months (+25%).

   A statistical analysis of channel uncertainty for a 30 month operating cycle has been performed. Based upon this analysis it has been concluded that sufficient margin exists between the existing Technical Specification and the licensing basis Safety Analysis to accommodate the channel statistical error resulting from a 30 month operating cycle. The existing margin between the Technical Specification and the Safety Analysis provides assurance that plant protective actions will occur as required. It is therefore concluded that changing the surveillance interval from 18 months (+25%) to 24 months (+25%) will not result in a significant increase in the probability or consequences of an accident previously evaluated.

2. The possibility of a new or different kind of accident from any accident previously evaluated has not been created.

   The proposed change in operating cycle length due to an increased surveillance interval will not result in a channel statistical allowance which exceeds the current margin between the existing Technical Specification and the Safety Analysis. Plant equipment, which will be set at (or more conservatively than) Technical Specification limits, will provide protective functions to assure that Safety Analysis limits are not exceeded. This will prevent the possibility of a new or different kind of accident from any previously evaluated from occurring.

3. A significant reduction in a margin of safety is not involved.

   The above change in surveillance interval resulting from an increased operating cycle will not result in a channel statistical allowance which exceeds the margin which exists between the current Technical Specification and the licensing basis Safety Analysis. This margin, which is equivalent to the existing margin, is necessary to assure that protective safety functions will occur so that Safety Analysis limits are not exceeded.

SAFETY ASSESSMENT

LOSS OF POWER
UNDERVOLTAGE AND DEGRADED VOLTAGE RELAYS

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

DESCRIPTION OF CHANGE

Technical Specification table 4.1-1, items 29a and 29b, specify that the Loss of Power (undervoltage and degraded voltage) relays be calibrated and tested at a refueling interval, and item 29c specifies that the undervoltage alarm be calibrated at every refueling interval. Item 30c, in Table 4.1-1, specifies that the undervoltage (station blackout) input to Auxiliary Feedwater be calibrated at every refueling interval. Currently, these surveillances are performed every 18 months (+25%). The proposed change in test frequency is to every 24 months (+25%). The proposed change is being made in accordance with the guidance in Generic Letter 91-04.

The 480 volt electrical system is divided into four buses. The 480 Volt buses are supplied through transformers from the 6.9 kv buses as follows: 2A from 2, 3A from 3, 5A from 5, and 6A from 6 (buses 2A and 3A are within the same power train). Tie breakers are provided between 480 volt buses 2A and 3A, 2A and 5A, and 3A and 6A. The required safeguards equipment circuits are dispersed among the 480 Volt buses. The normal source of power for buses 5A and 6A is the 138 kv system (by a station auxiliary transformer and 6.9 kv buses 5 and 6). The normal source of power for buses 2A and 3A is the main generator (by the Unit Auxiliary transformer and 6.9 kv buses 2 and 3). Buses 2 and 3 are tied to buses 5 and 6 via the "dead-fast" transfer scheme in the event of a unit trip.

One emergency diesel-generator set is connected to bus 5A, one to 6A, and the other to buses 2A and 3A. Each set will automatically start on a safety injection signal or upon undervoltage on any 480 V bus.

Along with other trips, the 480 Volt bus normal supply breakers are tripped by the following:

1. Safety Injection and Blackout (approximately 45% on bus 5A or 6A), or No Safety Injection with Unit trip and Blackout.

2. Degraded voltage on each respective bus.

3. Degraded voltage with a safety injection signal for 10 seconds.

The "short time" undervoltage relays (item 1 above) provide logic inputs to the sequencing logic and diesel start circuitry. Their setpoints (appr. 45%) are designed to give fast trip response for complete loss of power (dead bus) conditions.

The transfer from normal supply to EDG supply of 480 Volt safeguards buses upon sustained undervoltage is actuated by two undervoltage relays (one set at appr. 85% on each bus). Two out of two logic will activate an Agastat (one set at 180 $\pm$30 seconds and a second one set at 10$\pm$2 seconds which is in series with safety injection contacts) which in turn trips its respective 480-V normal supply breaker. This trip provides additional protection of the safeguards loads against degraded voltage conditions and provides an alternate power supply to establish a correct voltage.

The station blackout relays provide a loss of offsite power input to the automatic start feature on the steam driven auxiliary feedwater pump. Separate relays provide an alarm in the central control room to alert the operator when a 480 V bus falls to approximately 90 percent.

Completed test data for the 1984, 1987, 1989 and 1991 refueling intervals were reviewed. LER's based on the test results of the 1984 and 1986 refueling intervals were also reviewed. The undervoltage and station blackout relays were always within specifications. This review identified eight occurrences where degraded voltage relays (27S1, 27S2) were judged to be out of tolerance, but no instances of a failure to operate. These eight occurrences involved four of the eight relays installed. Except for some difficulty in calibration and one out of tolerance during one test, the alarm relays were found within specification.

All of the undervoltage and station blackout relays and most of the degraded voltage and alarm relays were found to be within specification at each of the refueling outage calibration periods. These calibration periods covered a total time period of about seven years. There were no instances of a failure of a relay to operate.

During the 1993 Refueling Outage, the degraded voltage relays were replaced by ASEA Brown Boveri type 27 N (solid state, high accuracy devices) whose long term performance is expected to be superior to the currently installed relays based upon equipment specifications.

BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The proposed change does not involve a significant hazards consideration since:

1. There is no significant increase in the probability or consequences of an accident.

   The Technical Specifications specify that the Loss of Power (undervoltage and degraded voltage) relays be calibrated and tested at a refueling interval; that the undervoltage alarm be calibrated at a refueling interval, and that the undervoltage (station blackout) input to Auxiliary Feedwater be calibrated at refueling intervals. It is proposed that the surveillance frequency be revised from 18 months (+25%) to 24 months (+25%).

   All of the undervoltage and station blackout relays were found to be within specification at each of the refueling outage calibration periods.

   Since the old relays have been replaced with relays from a different manufacturer whose drift characteristics are expected to be superior, extending the surveillance interval by several months will not significantly increase the probability or consequences of an accident.

2. The possibility of a new or different kind of accident from any previously analyzed has not been created.

   Past test results provide reasonable assurance that the relays will perform in an acceptable manner for an extended operating cycle. With the installation of the new relays, whose performance will surpass the old relays, it is concluded that the plant will perform within its design basis for an extended operating cycle. Therefore, the possibility of a new or different kind of accident from any previously analyzed has not been created.

3. There has been no significant reduction in the margin of safety.

   Since the new relays will surpass the performance of the old relays, there is reasonable assurance that a significant reduction in the margin of safety has not resulted from an extended operating cycle.

SAFETY ASSESSMENT

REFUELING WATER STORAGE TANK LEVEL

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

DESCRIPTION OF CHANGE

The current Indian Point Unit 2 Technical Specification requires that a channel calibration for the Refueling Water Storage Tank (RWST) be performed every refueling outage (Table 4.1-1, item 15.) As a result of a statistical analysis of channel accuracy based upon a review of past test data, it is proposed that this calibration frequency be increased to once per quarter. This will formalize current and past surveillance practices at the plant.

The purpose of the RWST low level alarm is to alert the operator to check the RWST level and start to terminate the injection phase and initiate the recirculation phase of safety injection during a large break LOCA by initiating the 8 switch sequence. The alarm must be set high enough to allow sufficient time for the operator to switch over without depleting the tank to a point where the SI pumps could be damaged and to allow sufficient volume for NaOH spray into containment.

Pursuant to Technical Specification Section 3.3.A.3, 246,000 gallons is required for the injection phase, 60,000 gallons for the recirculation phase, with the rest of the tank inventory being made up of unavailable volume, margin and instrument uncertainties.

All completed test data for the last six calibrations were reviewed. The "As Left/ As Found" data from the completed test procedures was statistically evaluated to determine a projected 30 month drift value with a 95% probability at a 75% confidence level. This data has been evaluated for determination of population normality and outliers. Where possible, outliers have been eliminated by use of accepted statistical tests or where mechanistic causes were determined to justify elimination. Also, past M&TE accuracies were reviewed to insure that the M&TE used was of an equivalent accuracy such that it would not have biased the data in a non-conservative direction.

The resulting 30 month projected drift value in combination with the other channel uncertainties resulted in a Channel Statistical Allowance that was too large to support the present licensing basis over a 30 month surveillance interval. Therefore, this evaluation was completed on the basis that the present practice of instrument calibration every 3 months would continue. The current licensing basis limits channel uncertainties to 12,400 gallons which is equivalent to the most adverse drift that could occur in a period somewhat greater than 3 months. The evaluated 3 month drift was used as an input to determine the Channel Statistical Allowance (CSA) using the Westinghouse setpoint methodology. Included in the evaluation along with instrument drift was the determination of all other channel uncertainties, including Sensor, Rack, M&TE, and Process Effects for normal environmental conditions.

Basis for No Significant Hazards Consideration Determination

The proposed change does not involve a significant hazards consideration since:

1. A significant increase in the probability or consequences of an accident previously evaluated will not occur.

   It is proposed that the channel calibration frequency for the RWST instrumentation be changed from every 18 months (+25%) to quarterly (once every 3 months).

   A statistical analysis of channel uncertainty for a 3 month surveillance has been performed. Based upon this analysis it has been concluded that sufficient margin exists between the existing Technical Specification limit and the licensing basis Safety Analysis limit to accommodate the channel statistical error resulting from a 3 month quarterly surveillance. The existing margin between the Technical Specification limit and the Safety Analysis limit provides assurance that plant protective actions will occur as required. It is therefore concluded that changing the surveillance interval from 18 months (+25%) to quarterly will not result in a significant increase in the probability or consequences of an accident previously evaluated.

2. The possibility of a new or different kind of accident from any accident previously evaluated has not been created.

   The proposed change in surveillance interval will result in a channel statistical allowance which provides the necessary margin between the existing Technical Specification limit and the Safety Analysis limit. Plant equipment, which will be set at (or more conservatively than) Technical Specification limits, will provide protective functions to assure that Safety Analysis limits are not exceeded. This will prevent the possibility of a new or different kind of accident from any previously evaluated from occurring.

3. A significant reduction in a margin of safety is not involved.

   The above change in surveillance interval will result in a channel statistical allowance which is necessary between the current Technical Specification limit and the licensing basis Safety Analysis limit. This margin is necessary to assure that protective safety functions will occur so that Safety Analysis limits are not exceeded.

SAFETY ASSESSMENT

OPS & LOPAR TRIP

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

DESCRIPTION OF CHANGE

The current Indian Point Unit 2 Technical Specifications require that a calibration of the LOPAR (Low Parasitic) trip system be performed every refueling interval (Table 4.1-1, item 28). Currently, this calibration is performed every 18 months (+25%). It is proposed that this calibration frequency be revised to every 24 months (+25%). This change is being made in accordance with the guidance contained in Generic Letter 91-04.

In this assessment, past data from several procedures was evaluated. One procedure calibrates the RCS cold leg wide range temperature channels which provides input to the OPS and LOPAR trip, amongst others. Two other procedures, one which calibrates portions of the instrumentation that provides input to OPS and a second which calibrates the bistable for the LOPAR trip, were also considered.

As part of this evaluation, completed test procedures from the February 1986 outage to the present were reviewed, including midcycle outage calibrations that may have resulted due to channel failures or modifications, and the impact of measurement and Test Equipment (M&TE) used to record the data. The "As Left/As Found" data from the completed test procedures was statistically evaluated to determine a projected 30 month drift value with a 95% probability at a 95% confidence level. To reduce the total uncertainties for these channels, a commitment exists to calibrate the complete instrument string from the R/I down, per the Technical Specification requirements, 31 days prior to entering a condition when OPS will be required and every 31 days while in a plant condition requiring OPS. Based on this commitment, drift for the OPS function racks is based on a monthly interval instead of 30 months. Drift allowance for the input sensors is based on a 30 month surveillance interval. These drift values were used as an input to determine the Channel Statistical Allowance (CSA) using the Westinghouse setpoint methodology. Included in the evaluation along with instrument drift was the determination of all other channel uncertainties, including Sensor, Rack, M&TE, and Process Effects for normal environmental conditions.

The results of the channel statistical uncertainty evaluation have been evaluated and factored into the OPS system and the LOPAR trip setpoint. The latter Technical Specification value has been revised from 350°F to 381°F to accommodate a potential 30 month operating cycle.

Basis for No Significant Hazards Determination

The proposed change does not involve a significant hazards consideration since:

1. A significant increase in the probability or consequences of an accident previously evaluated will not occur.

   It is proposed that the channel calibration frequency for the Over-pressurization protection system and the LOPAR trip system be changed from every 18 months (+25%) to every 24 months (+25%). This necessitates a change in the LOPAR Technical Specification trip setpoint from 350°F to 381°F.

   A statistical analysis of channel uncertainty for a 30 month operating cycle has been performed based upon historical test data. Based on this analysis, a change to the Technical Specifications is required. Sufficient margin exists between the Safety Analysis limit and the proposed Technical Specification limit to accommodate projected channel uncertainty over a 30 month operating cycle. A statistical basis exists to assure that protective action will occur to prevent Safety Analysis limits from being exceeded. Thus, there will not be a significant increase in the probability or consequences of an accident previously evaluated.

2. The possibility of a new or different kind of accident previously evaluated has not been created.

   Based upon a statistical analysis of past historical test data it has been demonstrated that reasonable assurance exists to conclude that Safety Analysis limits will not be exceeded over a 30 month operating cycle. The proposed Technical Specification limits provide margin with respect to the Safety Analysis limits and confidence that appropriate plant protective response will be provided to prevent the possibility of a new or different kind of accident from that previously evaluated from being created.

3. A significant reduction in a margin of safety is not involved.

   The proposed changes to the Technical Specification limits are being made to assure that the previously established margin remains the same between plant protective function set points and Safety Analysis limits. This margin is based upon an evaluation of past historical test data and analytical methods for projecting instrument channel uncertainty over a 30 month operating cycle. It is therefore concluded that the existing margin of safety has been preserved.

SAFETY ASSESSMENT

CONDENSER EVACUATION SYSTEM
(R-45)

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

DESCRIPTION OF CHANGE

Technical Specification Table 4.10-4, Item 3, specifies that the condenser evacuation system noble gas activity monitor (R-45) be calibrated each refueling interval. Currently, this calibration is performed every 18 months (+25%). It is proposed that the surveillance interval be extended to 24 months (+25%). This change is being proposed in accordance with the guidance contained in Generic Letter 91-04.

The discharge from the air ejector exhaust header of the condensers is monitored for gaseous radiation by the Condenser Air Ejector Gas Monitor, R-45. Gaseous radiation is indicative of a primary to secondary system leak. The gas discharge is normally routed to the turbine building exhaust vent. On high radiation level alarm, the condenser exhaust gases are diverted from the vent stack to the containment through a blower.

A beta scintillator is used to monitor the gaseous radiation level. The detector is inserted into an off-line fixed volume container which includes adequate shielding to reduce the background radiation.

Technical Specification Table 3.9-2, item 3, requires that the Condenser Evacuation System Noble Gas Activity Monitor be operable. The action required in the event that the monitor becomes inoperable is to take grab samples every 12 hours.

Due to the Radiation Monitoring Betterment Program, the Condenser Air Ejector Gas Monitor, R-15, was recently replaced by R-45. Because of this, there was only one completed procedure, PC-EM28. This was reviewed and found to be satisfactory.

This detector monitors activity in a potentially radioactive gas discharge pathway. The setpoint of the alarm/trip associated with the detector is set conservatively low at a level somewhat higher than the expected concentration. The setpoint is normally decades below the allowed release limits.

The installed monitoring system is not designed to determine the nature and amount of radioactivity in the system being monitored. The system monitors gross activity and is designed to generate an alarm and automatic diversion of the exhaust gases under abnormal conditions. Isotopic identification and concentrations are determined by grab sample analysis.

The Technical Specification Limiting Conditions for Operation associated with this detector places no restraints on plant operation. Also, this monitor has no setpoints which are critical to plant operation or safety, nor are its readings used in calculations which require discrete accuracy. Its primary functions are to provide indication of changing radiation levels, to provide an alarm on high radiation, and to divert the exhaust gases in the event of high radiation levels in the exhaust gases. The ODCM manual does not take credit for diversion of the gas discharges into the containment. It assumes releases are to the environment. Actual setpoints for the alarm and the diversion function are not critical to either plant operation or calculation of released radioactive material.

BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The proposed change does not involve a significant hazards consideration since:

1. There is no significant increase in the probability or consequences of an accident.

   It is proposed that the channel calibration frequency for the Condenser Evacuation System Noble Gas Activity Monitor (R-45) be changed from every 18 months (+25%) to every 24 months (+25%).

   Since this radiation monitor is relatively new a degree of uncertainty is introduced by extending the surveillance interval by several months. However, the setpoint for automatic diversion is set some what conservatively. It is established sufficiently high to avoid spurious actuations and yet sufficiently low so that diversion and alarm can occur should a step increase in radioactivity level occur. Under these circumstances considerable departure from the setpoint can be accommodated and the monitor will still perform its intended safety function. Continued monitor operability is important and malfunction would be detected by monthly checks during the extended operating cycle. Thus, despite the introduction of a new monitor, the capability of R-45 to tolerate drift in addition to monthly operator checks, leads to the conclusion that an extended operating cycle will not result in a significant increase in the probability or consequences of an accident.

2. The possibility of a new or different kind of accident from any previously analyzed has not been created.

   Monthly checks would identify abnormal operating characteristics, should the instrument fail to perform its intended function. In the event of tube rupture with a reactor coolant system radioactivity concentration corresponding to 1% defective fuel, the resultant site boundary dose would be within 10 CFR 20 limits should the monitor fail to perform its function (as discussed in FSAR). In addition, alternate means of alarms to indicate a tube rupture event are available. Thus, the possibility of a new or different kind of accident has not been created.

3. There has been no reduction in the margin of safety.

   Although this monitor is not necessary to mitigate releases below regulatory limits, it does provide the earliest alarm of a steam generator tube leak. In this regard, continued instrument operation is important. Continued instrument operability would be verified by the monthly checks in an extended operating cycle.

SAFETY ASSESSMENT

SERVICE WATER INLET TEMPERATURE
MONITORING INSTRUMENTATION

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

DESCRIPTION OF CHANGE

The current Indian Point Unit 2 Technical Specification requires that the Service Water Inlet Temperature monitoring instrumentation be calibrated every refueling interval (Table 4.1-1, item #45). Currently, this calibration is performed every 18 months (+25%). It is proposed that the calibration interval be extended to 24 months (+25%). This change is being made in accordance with the guidance contained in Generic Letter 91-04.

The service water system is designed to supply cooling water from the Hudson River to various heat loads in both the primary and secondary portions of the plant. Provision is made to ensure a continuous flow of cooling water to those systems and components necessary for plant safety either during normal operation or under abnormal and accident conditions. Since maximum service water temperature is an input variable to the Loss-of-Coolant-Accident, a Technical Specification limit of 95°F has been specified. This temperature is monitored by the river water temperature monitor at the Unit 2 Intake Structure. Instrumentation also monitors the temperature of the river water at the Intake Structures for Units 1 and 3. These instruments can be used to cross check the accuracy of the Unit 2 Instruments. The Unit 2 Intake Temperature Monitor is calibrated along with the Units 1 and 3 Intake Temperature Monitors and the discharge temperature monitoring instruments each refueling.

The original Bendix plant instrumentation used to monitor river water temperature was recently replaced by an Esterline Angus system, resulting in only two test procedures being completed. This includes any midcycle outage calibrations that may have resulted due to channel failures or modifications, and the impact of Measurement and Test Equipment (M&TE) used to record the data. The "As Left/As Found" data from the completed test procedures was reviewed with no deficiencies noted for the test criteria. The projected 30 month drift for this channel is based on conservative engineering judgement based on the plant data reviewed. Included in the evaluation was the determination of all channel uncertainties, including Sensor, Rack, M&TE, and Process Effects for normal environmental conditions.

The evaluation of instrument uncertainties was based on the currently installed hardware. Vendor specifications were used as appropriate, as well as verified Engineering Calculations.

"As Left/As Found" data applicable to the newly installed Esterline Angus System was reviewed for this evaluation. The largest Δ between the As Left/As Found data was 0.8°F for 16 months. Based on the review of this data and engineering judgement, this value was then extrapolated out to 30 months and used as an input into the overall CSA calculation for the alarm functions. Conservative engineering judgement was used to predict drift for both the sensor and the indicators.

The calculated uncertainty permits extension of the surveillance interval to 30 months with no other changes in the Technical Specification or Safety Analysis.

Basis for No Significant Hazards Considerations Determination

The proposed change does not involve a significant hazards consideration since:

1. A significant increase in the probability or consequences of an accident previously evaluated will not occur.

   It is proposed that the channel calibration frequency for the Service Water Inlet Temperature Monitoring Instrumentation be changed from every 18 months (+25%) to every 24 months (+25%).

   A statistical analysis of channel uncertainty for a 30 month operating cycle has been performed. Based upon this analysis it has been concluded that sufficient margin exists between the existing Technical Specification and the licensing basis Safety Analysis to accommodate the channel statistical error resulting from a 30 month operating cycle. The existing margin between the Technical Specification and the Safety Analysis provides assurance that plant protective actions will occur as required. It is therefore concluded that changing the surveillance interval from 18 months (+25%) to 24 months (+25%) will not result in a significant increase in the probability or consequences of an accident previously evaluated.

2. The possibility of a new or different kind of accident from any accident previously evaluated has not been created.

   The proposed change in operating cycle length due to an increased surveillance interval will not result in a channel statistical allowance which exceeds the current margin between the existing Technical Specification and the Safety Analysis. Plant equipment, which will be set at (or more conservatively than) Technical Specification limits, will provide protective functions to assure that Safety Analysis are not exceeded. This will prevent the possibility of a new or different kind of accident from any previously evaluated from occurring.

3. A significant reduction in a margin of safety is not involved.

   The above change in surveillance interval resulting from an increased operating cycle will not result in a channel statistical allowance which exceeds the existing margin between the current Technical Specification and the licensing basis Safety Analysis. This margin, which is equivalent to the existing margin, is necessary to assure that the protective safety functions occur and that the Safety Analysis limits are not exceeded.

SAFETY ASSESSMENT
SAMPLER FLOW RATE MONITORS

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

DESCRIPTION OF CHANGE

Technical Specification Table 4.10-4, Items 4.E and 5.E require that the Sample Flow Rate Monitors be calibrated at a refueling interval. Currently, this is performed on an 18 month (+25%) basis. It is proposed that the calibration interval be extended to 24 months (+25%). This change is being proposed in accordance with the guidance contained in Generic Letter 91-04.

Both the Plant Vent and the Stack Vent (Unit 1) effluents are monitored for Iodine and particulate activity. A continuous sample is drawn through filters for future counting. The totalized air flow through the filters is measured by the Sample flow Rate Monitors. These monitors are standard gas meters calibrated for use with air. These monitors are surveilled each refueling in accordance with the Chemistry Maintenance Program. The surveillance consists of removing the monitors from the system and replacing them with calibrated monitors from stock.

Table 3.9-2, items 4.e and 5.e require one Sampler Flow Rate Monitor to be operable during release via the respective pathway. The required action for an inoperable monitor is to estimate the flow rate every four hours.

These monitors are sent to a Con Edison facility for recalibration after each operating cycle. The removed monitors are replaced with calibrated monitors drawn from stock. These monitors do not have setpoints which are critical to plant operation or safety. Technical Specifications permit the use of estimated flow values in the event that a monitor is inoperable.

BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The proposed change does not involve a significant hazards consideration since:

1. There is no significant increase in the probability or consequences of an accident.

   It is proposed that the channel calibration frequency for the Sample Flow Rate Monitors be changed from every 18 months (+25%) to every 24 months (+25%).

   The flow rate monitors are used to estimate the total volume of air passed through filters. There is no setpoint or safety function served by these monitors. A high level of radioactivity in the discharge stream is detected by R-43 and/or R-44.

   Insofar as discharge via the unit vent is permissible with the monitors inoperable, extension of the surveillance interval will have no impact upon safety.

2. The possibility of a new or different kind of accident from any previously analyzed has not been created.

   As the nuclear safety function is provided by other monitors in the event of high radioactivity levels in the discharge stream, extension of the surveillance interval will have no impact upon the creation of a new or different kind of accident.

3. There has been no reduction in the margin of safety.

These flow monitors are utilized to determine the total air flow through filters for computational purposes. As adequate measures (other monitors) exist to prevent the possibility of discharging radioactivity in excess of applicable limits, there is virtually no impact upon safety incurred by extending the surveillance interval.

SAFETY ASSESSMENT

BORIC ACID MAKEUP FLOW SYSTEM

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

DESCRIPTION OF CHANGE

The current Indian Point Unit 2 Technical Specification requires a channel calibration for the Boric Acid Makeup Flow System be performed at each refueling interval (Table 4.1-1, item #20). Currently, this calibration is performed every 18 months (+25%). It is proposed that this calibration frequency be revised to every 24 months (+25%). This change is being made in accordance with the guidance contained in Generic Letter 91-04.

The Boric Acid Makeup Flow System was reviewed using the Westinghouse methodology for evaluating channel uncertainties. Each uncertainty term was determined according to the instrument characteristics/specifications. Past cycle calibration data was evaluated to determine how well the instruments had performed from one cycle to the next. This evaluation included a review of any work order data that may have been taken during a midcycle outage etc., or any modifications to the channels. Also, past M&TE accuracies were reviewed to insure that the M&TE used was of an equivalent accuracy such that it would not have biased the data in a non-conservative direction. In addition to drift, sensor, rack and M&TE terms have been incorporated into the total channel uncertainty calculations.

The calculated uncertainty permits extension of the surveillance interval to 30 months with no other changes in the Technical Specifications or Safety Analysis.

Basis for No Significant Hazards Consideration Determination

The proposed change does not involve a significant hazards consideration since:

1. A significant increase in the probability or consequences of an accident previously evaluated will not occur.

   It is proposed that the channel calibration frequency for the Boric Acid Makeup flow System be revised from every 18 months (+25%) to every 24 months (+25%).

   A statistical analysis of channel uncertainty for a 30 month operating cycle has been performed. Based upon this analysis it has been concluded that sufficient margin exists between the existing Technical Specification limit and the licensing basis Safety Analysis limit to accommodate the channel statistical error resulting from a 30 month operating cycle. The existing margin between the Technical Specification limit and the Safety Analysis limit provides assurance that plant protective actions will occur as required. It is therefore concluded that changing the surveillance interval from 18 months (+25%) to 24 months (+25%) will not result in a significant increase in the probability or consequences of an accident previously evaluated.

2. The possibility of a new or different kind of accident from any accident previously evaluated has not been created.

   The proposed change in operating cycle length due to an increased surveillance interval will not result in a channel statistical allowance which exceeds the current margin between the existing Technical Specification limit and the Safety Analysis limit. Plant equipment, which will be set at (or more conservatively than) Technical Specification limits, will provide protective functions to assure that Safety Analysis limits are not exceeded. This will prevent the possibility of a new or different kind of accident from any previously evaluated from occurring.

3. A significant reduction in a margin of safety is not involved.

   The above change in surveillance interval resulting from an increased operating cycle will not result in a channel statistical allowance which exceeds the margin which exists between the current Technical Specification limit and the licensing basis Safety Analysis limit. This margin, which is equivalent to the existing margin, is necessary to assure that protective safety functions will occur so that Safety Analysis limits are not exceeded.

SAFETY ASSESSMENT

PLANT VENT NOBLE GAS EFFLUENT MONITOR
(R-27)

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

DESCRIPTION OF CHANGE

Technical specification Table 4.1-1, item 38, specifies that the Plant Vent Noble Gas Effluent Monitor (R-27) be calibrated at every refueling interval. Currently, this calibration is performed on an 18 month (+25%) basis. It is proposed that the surveillance interval be extended to 24 months (+25%). This change is being proposed in accordance with the guidance contained in Generic Letter 91-04.

The Plant Vent Noble Gas Effluent Monitor (R-27) is intended to provide information about the magnitude of releases of radioactive materials in the event of an accident. It is installed in the Boric Acid Evaporator building. The monitor is skid mounted and fixed in place by anchor bolts. Connections have been provided for data processors, displays, electric power, and nitrogen purge.

The calibration procedure is a relatively new procedure and only one completed procedure was available for review. This procedure found that R-27 met the operability criteria and no equipment discrepancies were noted.

The installed monitoring system is not designed to determine the nature and amount of radioactivity in the system being monitored. The system monitors gross activity and is designed to record the reading. Isotopic identification and concentrations are determined by grab sample analysis.

The Technical Specification Limiting Conditions for Operation associated with this unit place no restraints on plant operation. Also, this monitor has no setpoints which are critical to plant operation or safety. Its primary function is to provide indication of high radiation levels. Actual readings are not critical to either plant operation or calculation of released radioactivity material.

BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The proposed change does not involve a significant hazards consideration since:

1. There is no significant increase in the probability or consequences of an accident.

   It is proposed that the channel calibration frequency for the Plant Vent Noble Gas Effluent Monitor (R-27) be changed from every 18 months (+25%) to every 24 months (+25%).

   R-27 is a high range noble gas monitor intended for use after an accident to provide information about the magnitude of radioactive releases. It serves no purpose during normal operation. It provides no function to prevent or mitigate an accident but does provide a role in assessing the consequences of an accident. As the monitor is a high range monitor, an estimate of the magnitude of release rather than accuracy is important. Accordingly, continued operability of the instrument during an extended operating cycle is more important than the device exhibiting minimal drift characteristics. Malfunction of the instrument would be detected by the shift checks and functional tests performed during the extended operating cycle.

2. The possibility of a new or different kind of accident from any previously analyzed has not been created.

   Since the monitor provides no preventive or mitigating action in the event of an accident, no new or different type of accident has been created by extending the operating cycle. In terms of post accident assessment capability, alternate means exist to assess offsite releases in the event of failure of this instrument.

3. There has been no reduction in the margin of safety.

   Since the instrument provides no safety function and alternate means exist for post accident assessment purposes, there will be no impact on safety due to an extended period between calibrations.