UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

October 21, 1991

Docket No. 50-247

Mr. Stephen B. Bram
Vice President, Nuclear Power
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenue
Buchanan, New York 10511

Dear Mr. Bram:

SUBJECT: ISSUANCE OF AMENDMENT FOR INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 (TAC NO. M79910)

The Commission has issued the enclosed Amendment No. 155 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated March 7, 1991, as supplemented August 2, 1991.

The amendment revises Technical Specification 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 incorporate revised pressure-temperature limits and Overpressure Protection System (OPS) parameters in accordance with 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 revised to replace pressure-temperature requirements on the reactor coolant system when the reactor is critical, with a fixed temperature limit.

NRC FILE CENTER COPY

9111200266 911021
PDR ADOCK 05000247
P PDR

DF01
'11

CP-1

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

Original signed by:

Francis J. Williams, Jr., Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 155 to DPR-26
2. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION:
See attached sheet

OFC	PDI-1:LA	PDI-1:PM	OGC	PDI-1:D
NAME	CVogan	FWilliams:avl	MZobler	RACapra
DATE	9/18/91	9/23/91	9/27/91	10/21/91

OFFICIAL RECORD COPY
Document Name: IP2 AMDT 79910

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

Francis J. Williams, Jr., Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 155 to DPR-26
2. Safety Evaluation

cc w/enclosures:
See next page

DATED: October 21, 1991

AMENDMENT NO. 155 TO FACILITY OPERATING LICENSE NO. DPR-26-INDIAN POINT UNIT 2

Docket File
NRC & Local PDRs
PDI-1 Reading
S. Varga, 14/E/4
J. Calvo, 14/A/4
R. Capra
C. Vogan
F. Williams
C. Cowgill
OGC-WF
D. Hagan, 3302 MNBB
G. Hill (4), P-137
Wanda Jones, P-130A
C. Grimes, 11/F/23
ACRS (10)
GPA/PA
OC/LFMB
Plant File

cc: Plant Service list

Mr. Stephen B. Bram
Consolidated Edison Company
of New York, Inc.

Indian Point Nuclear Generating Station
Units 1/2

cc:

Mayor, Village of Buchanan
236 Tate Avenue
Buchanan, New York 10511

Mr. Charlie Donaldson, Esquire
Assistant Attorney General
New York Department of Law
120 Broadway
New York, New York 10271

Ms. Donna Ross
New York State Energy Office
2 Empire State Plaza
16th Floor
Albany, New York 12223

Mr. Peter Kokolakis, Director
Nuclear Licensing
Power Authority of the State
of New York
123 Main Street
White Plains, New York 10601

Mr. Charles W. Jackson
Manager of Nuclear Safety and
Licensing
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenue
Buchanan, New York 10511

Mr. Walter Stein
Secretary - NFSC
Consolidated Edison Company
of New York, Inc.
4 Irving Place - 1822
New York, New York 10003

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
Post Office Box 38
Buchanan, New York 10511

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. Brent L. Brandenburg
Assistant General Counsel
Consolidated Edison Company
of New York, Inc.
4 Irving Place - 1822
New York, New York 10003

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

DOCKET NO. 50-247

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 155
License No. DPR-26

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Consolidated Edison Company of of New York, Inc. (the licensee) dated March 7, 1991, as supplemented on August 2, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-26 is hereby amended to read as follows:

9111200268 911021
PDR ADOCK 05000247
P PDR

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 155, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Capra

Robert A. Capra, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 21, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 155

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Pages
vi	vi
viii	viii
3.1.A-2	3.1.A-2
3.1.A-3	3.1.A-3
3.1.A-6	3.1.A-6
3.1.A-7	3.1.A-7
3.1.A-9	3.1.A-9
Table 3.1.A-2 (page 1 of 2)	Table 3.1.A-2 (Page 1 of 2)
Table 3.1.A-2 (page 2 of 2)	Table 3.1.A-2 (Page 2 of 2)
Figure 3.1.A-1	Figure 3.1.A-1
Figure 3.1.A-2	Figure 3.1.A-2
Figure 3.1.A-3	Figure 3.1.A-3
3.1.B-1	3.1.B-1
3.1.B-3	3.1.B-3
3.1.B-4	3.1.B-4
3.1.B-5	3.1.B-5
3.1.B-6	3.1.B-6
3.1.B-7	3.1.B-7
3.1.B-8	3.1.B-8
Figure 3.1.B-1	Figure 3.1.B-1
Figure 3.1.B-2	Figure 3.1.B-2
Table 3.1.B-1	—
3.1.C-1	3.1.C-1
3.1.C-2	3.1.C-2
3.2.2	3.2.2
3.3.3	3.3.3
3.3.4	3.3.4
3.3.5	3.3.5
3.3.6	3.3.6
3.3.7	3.3.7
3.3.8	3.3.8
3.3.9	3.3.9
3.3.10	3.3.10
3.3.11	3.3.11
3.3.12	3.3.12
3.3.13	3.3.13
3.3.14	3.3.14
3.3.15	3.3.15
4.3.1	4.3.1
4.3.2	4.3.2
Figure 4.3-1	Figure 4.3-1

LIST OF TABLES

Title	Table No.
Frequency Notation	1-1
Reactor Coolant (RC) Pumps/Residual Heat Removal (RHR) Pump(s) Operability/Operating Requirements for Decay Heat Removal and Core Mixing	3.1.A-1
OPS Operability Requirements	3.1.A-2
Maximum Allowable Power Range Neutron Flux High Setpoint with Inoperable Steam Line Safety Valves During 4-Loop Operation	3.4-1
Engineered Safety Features Initiation Instrument Setting Limits	3.5-1
Reactor Trip Instrumentation Limiting Operating Conditions	3.5-2
Instrumentation Operating Conditions for Engineered Safety Features	3.5-3
Instrumentation Operating Conditions for Isolation Functions	3.5-4
Accident Monitoring Instrumentation	3.5-5
Non-Automatic Containment Isolation Valves Open Continuously or Intermittently for Plant Operation	3.6-1
Radioactive Liquid Effluent Monitoring Instrumentation	3.9-1
Radioactive Gaseous Effluent Monitoring Instrumentation	3.9-2
Safety-Related Shock Suppressors (Snubbers)	3.12-1
Fire Detection Instruments	3.13-1
Fire Hose Stations and Fire Hydrants	3.13-2
Meteorological Monitoring Instrumentation	3.15-1
Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels	4.1-1
Frequencies for Sampling Tests	4.1-2

LIST OF FIGURES

Reactor Core Safety Limit-Four Loops In Operation	2.1-1
PORV Opening Pressure for Operation Less Than or Equal to 305°F	3.1.A-1
Maximum Pressurizer Level with PORVs Inoperable and One Charging Pump Energized	3.1.A-2
Maximum Reactor Coolant System Pressure for Operation With PORVs Inoperable and One Safety Injection Pump and/or Three Charging Pumps Energized	3.1.A-3
Coolant System Heatup Limitations	3.1.B-1
Coolant System Cooldown Limitations	3.1.B-2
Figure Deleted (See Figure 3.8-3)	3.8-1
Spent Fuel Storage Rack Layout - IP2 Pool	3.8-2
Limiting Fuel Burnup versus Initial Enrichment	3.8-3
Required Hot Shutdown Margin versus Reactor Coolant Boron Concentration	3.10-1
Hot Channel Factor Normalized Operating Envelope	3.10-2
Rod Bank Insertion Limits (Four-Loop Operation) 100 Step Overlap	3.10-3
Figure Deleted	3.10-4
Target Band on Indicated Flux Difference as a Function of Operating Power Level	3.10-5
Permissible Operating Band on Indicated Flux Difference as a Function of Burnup	3.10-6
Vessel Leak Test Limitations	4.3-1
Map Defining Unrestricted Areas for Radioactive Gaseous and Liquid Effluents	5.1-1

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 $\leq$ 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.

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

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

Reference

(1) 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

Graph Paper
PRINTED IN U.S.A.

FIGURE 3.1.A-1
PORV OPENING PRESSURE FOR OPERATION LESS THAN OR EQUAL TO 305°F

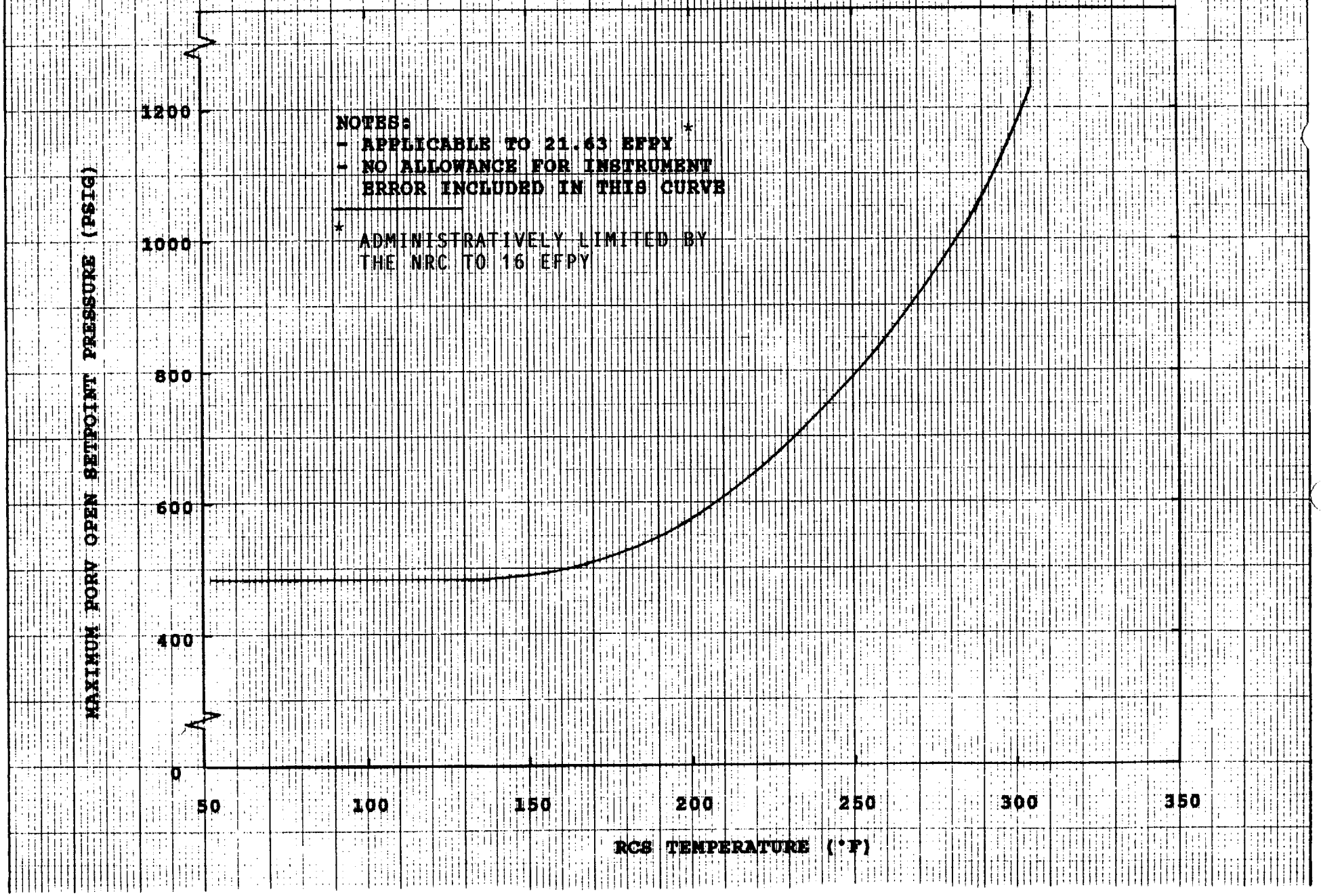
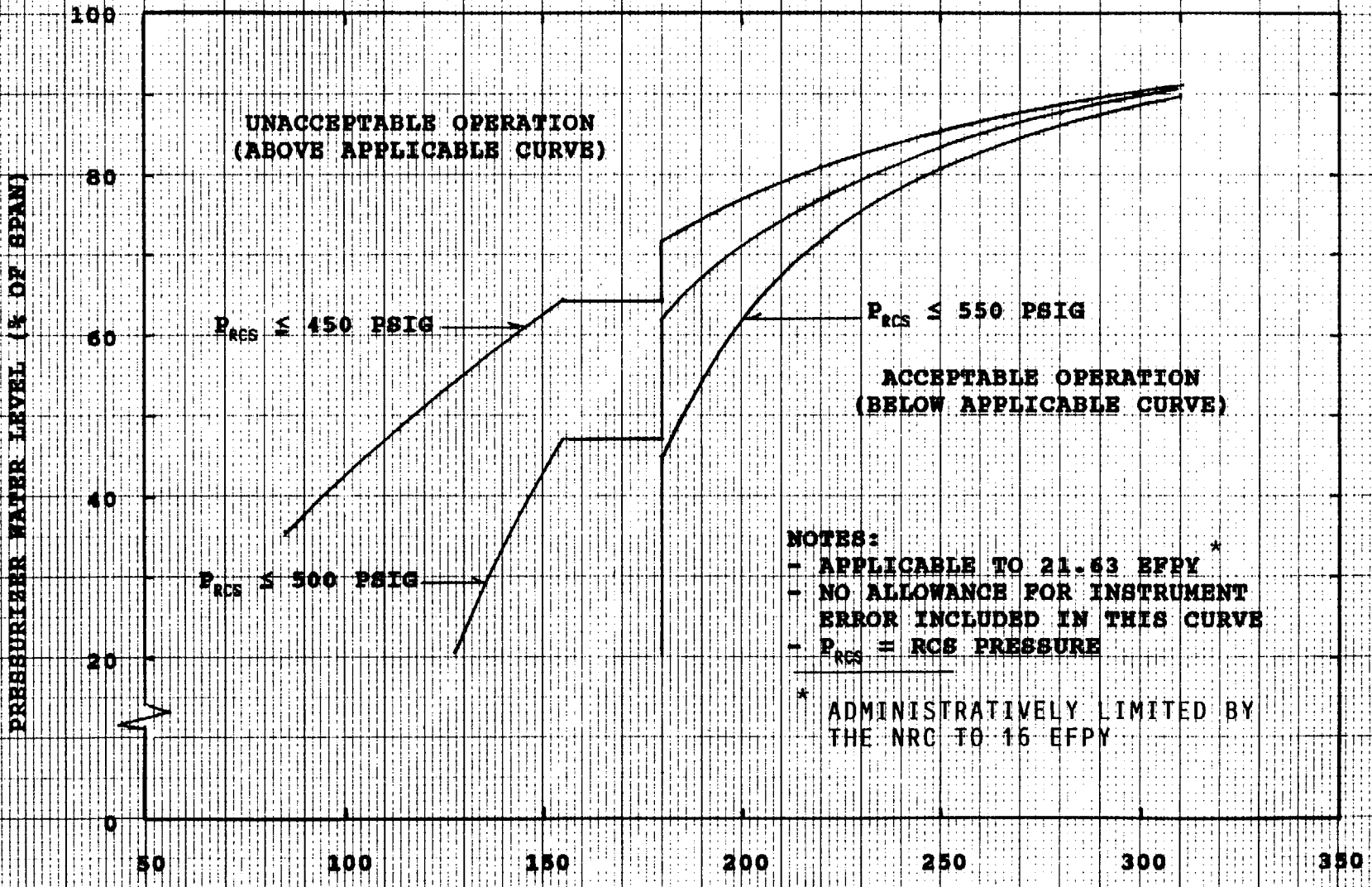
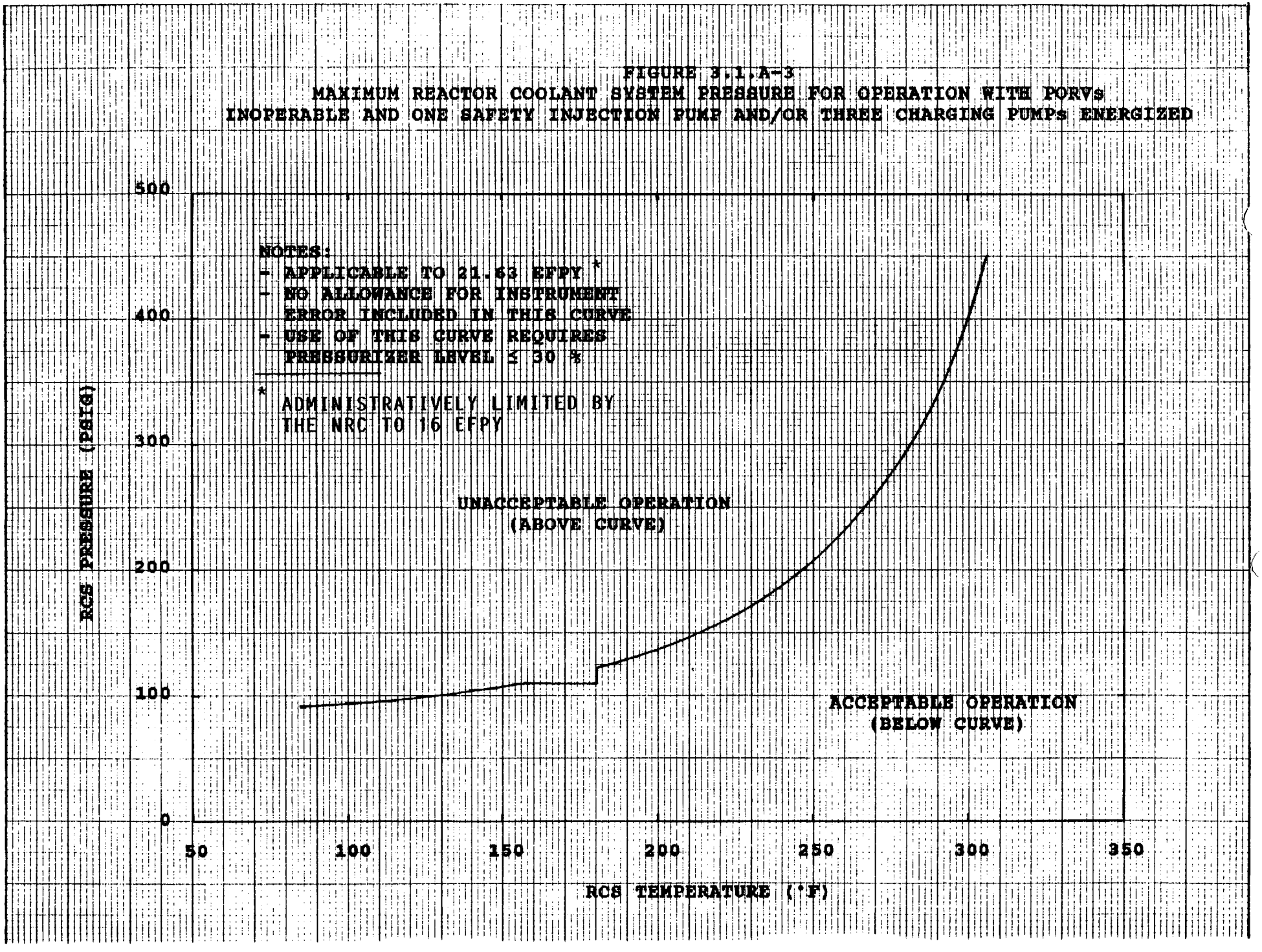


FIGURE 3.1.A-2
MAXIMUM PRESSURIZER LEVEL WITH PORVs
INOPERABLE AND ONE CHARGING PUMP ENERGIZED



20 DIVISIONS PER INCH BOLD WAY S1 84 LG Graph Paper DIRECT FROM CODEX BOOK CO. NORWOOD, MASS. 02062 PRINTED IN U.S.A.

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

---

* Curves have been approved but administratively limited by the NRC to 16 EFPY for up to 60°F/hr. and 12 EFPY for up to 100°F/hr. pending acceptance of the use of the Raju-Newman method in the generation of heatup and cooldown curves.

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

The 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.

---

* See first footnote on page 3.1.B-1.

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 0.98×10^{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.

---

* See first footnote on page 3.1.B-1.

The approach specifies that the allowable total stress intensity factor (K_I), at any time during heatup or cooldown, cannot be greater than that shown on the K_{IR} curve[5] for the metal temperature at that time. Furthermore, the approach applies an explicit safety factor of 2.0 on the stress intensity factor induced by pressure gradients. Thus, the governing equation for the heatup-cooldown analysis is:

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

where:

K_{Im} is the stress intensity factor caused by membrane (pressure) stress,

K_{It} is the stress intensity factor caused by the thermal gradients,

K_{IR} is provided by the code as a function of temperature relative to the RT_{NDT} of the material.

During the heatup analysis, Equation (1) is evaluated for two distinct situations.

First, allowable pressure-temperature relationships are developed for steady state (i.e., zero rate of change of temperature) conditions assuming the presence of the code reference 1/4 T deep flaw at the ID of the pressure vessel. Due to the fact that, during heatup, the thermal gradients in the vessel wall tend to produce compressive stresses at the 1/4 T location, the tensile stresses induced by internal pressure are somewhat alleviated. Thus, a pressure-temperature curve based on steady state condition (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the 1/4 T location is treated as the governing factor.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which the 3/4 T location becomes the controlling factor. Unlike the situation at the 1/4 T location, at the 3/4 T position (i.e., the tip of the 1/4 T deep O.D. flaw) the thermal gradients established during heatup produce stresses which are tensile in nature, and thus

tend to reinforce the pressure stresses present. These thermal stresses are, of course, dependent on both the rate of heatup and the time (or water temperature) along the heatup ramp. Furthermore, since the thermal stresses at 3/4 T are tensile and increase with increasing heatup rate, a lower bound curve similar to that described in the preceding paragraph cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady state and finite heatup rate situations, the final limit curves are produced in the following fashion. First, a composite curve is constructed based on a point-by-point comparison of the steady state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the two values taken from the curves under consideration. The composite curve is then adjusted to allow for possible errors in the pressure- and temperature-sensing instruments.

The use of the composite curve becomes mandatory in setting heatup limitations because it is possible for conditions to exist such that, over the course of the heatup ramp, the controlling analysis switches from the O.D. to the I.D. location, and the pressure limit must, at all times, be based on the most conservative case.

The cooldown analysis proceeds in the same fashion as that for heatup, with the exception that the controlling location is always at 1/4 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.

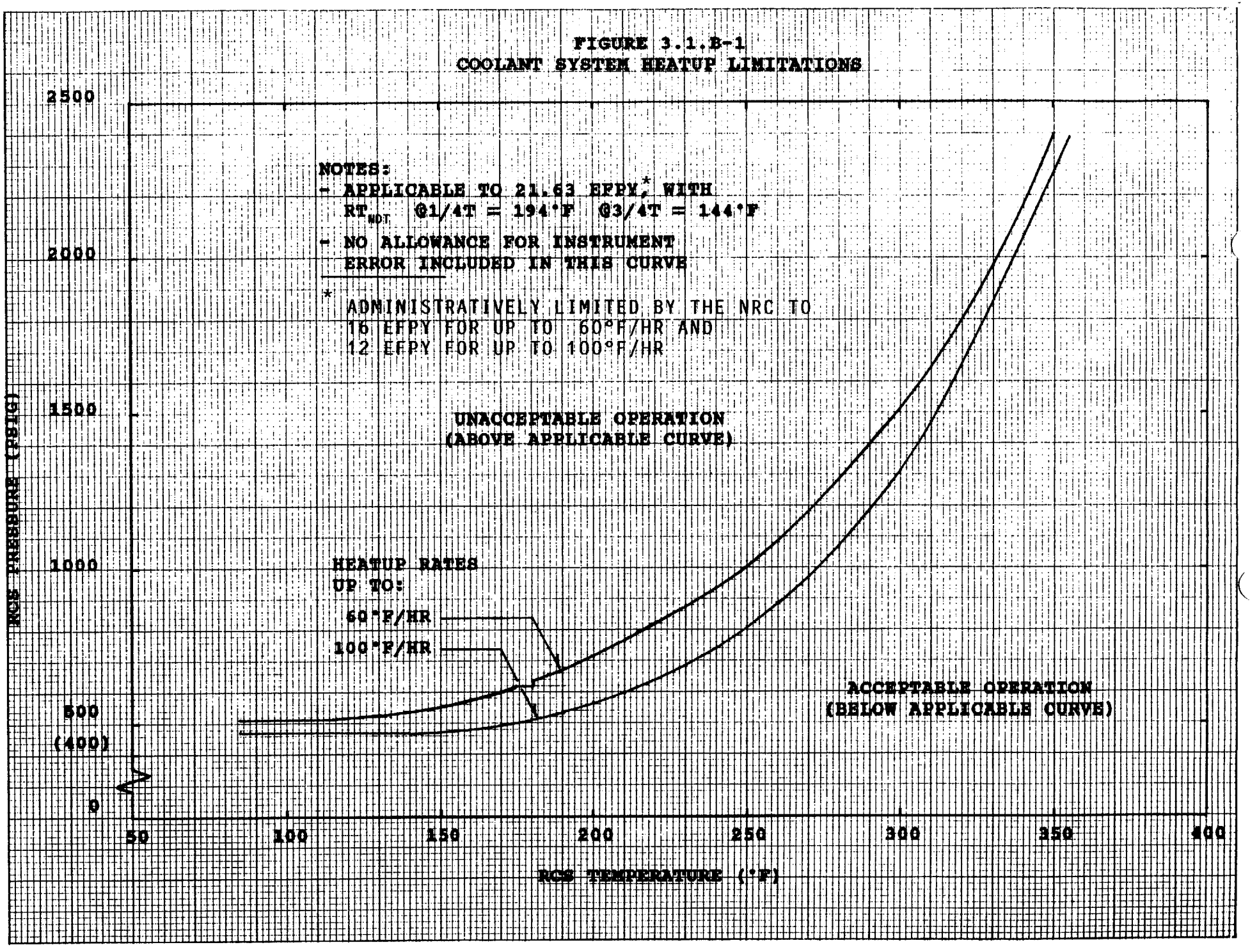
(10) Final Report - SWRI Project No. 02-5212 - "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule Y," E.B. Norris, November 1980.

(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.

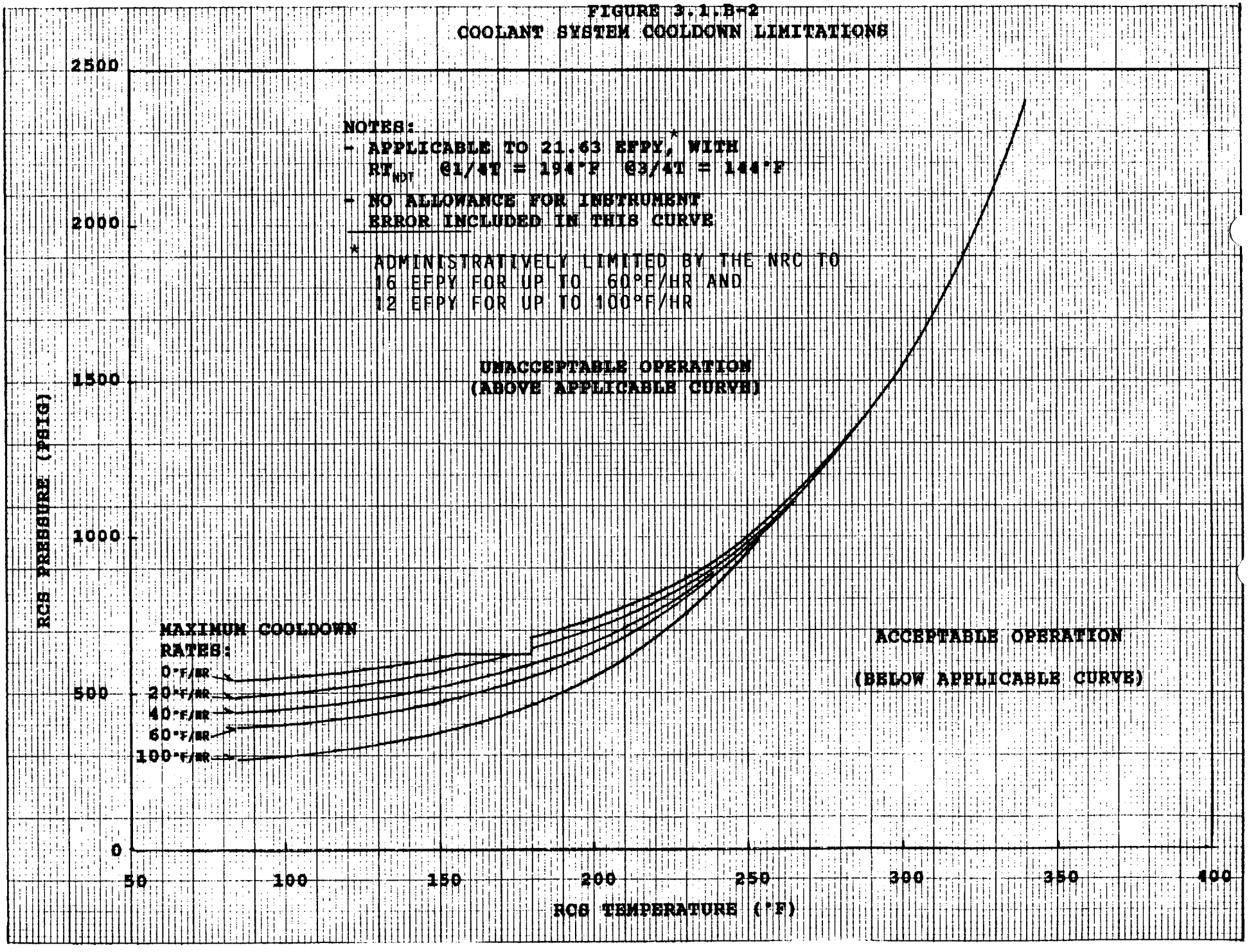
(13) WCAP-12796, "Heatup and Cooldown Limit Curves for the Consolidated Edison Company Indian Point Unit 2 Reactor Vessel," N.K. Ray, Westinghouse Electric Corporation.

LITHO PAPER ®
PRINTED IN U.S.A.



Graph Paper
DIRECT FROM CODEX BOOK CO. NORWOOD, MASS. 02062
PRINTED IN U.S.A.

FIGURE 3.1.B-2
COOLANT SYSTEM COOLDOWN LIMITATIONS



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

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.

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.

B. CONTAINMENT COOLING AND IODINE REMOVAL SYSTEMS

1. The reactor shall not be made critical unless the following conditions are met:

   a. The spray additive tank contains not less than 4000 gallons of solution with a sodium hydroxide concentration of not less than 33% by weight.

   b. The five fan cooler-charcoal filter units and the two spray pumps, with their associated valves and piping, are operable.

2. During power operation, the requirements of 3.3.B.1 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 3.3.B.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.B.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 fan cooler unit may be inoperable during normal reactor operation for a period not to exceed 7 days provided both containment spray pumps are operable.

   b. One containment spray pump may be inoperable during normal reactor operation, for a period not to exceed 72 hours, provided the five fan cooler units and the remaining containment spray pump are operable.

   c. Any valve required for the functioning of the system during and following accident conditions may be inoperable provided it is restored to operable status within 7 days or 72 hours for the fan cooler or containment spray systems respectively, and all valves in the system that provide the duplicate function are operable.

d. The spray additive tank and its associated piping, valves and eductors may be inoperable during normal reactor operation for a period not to exceed 72 hours provided both containment spray pumps and the five fan cooler units are operable.

C. ISOLATION VALVE SEAL WATER SYSTEM (IVSWS)

1. The reactor shall not be brought above cold shutdown unless the following requirements are met:

   a. The IVSWS shall be operable.

   b. The IVSW tank shall be maintained at a minimum pressure of 52 psig and contain a minimum of 144 gallons of water.

2. The requirements of 3.3.C.1 may be modified to allow any one of the following components to be inoperable at any one time:

   a. Any one header of the IVSWS may be inoperable for a period not to exceed seven consecutive days.

   b. Any valve required for the functioning of the system during and following accident conditions may be inoperable provided it is restored to an operable status within seven days and all valves in the system that provide a duplicate function have been demonstrated to be operable.

3. If the IVSWS System is not restored to an operable status within the time period specified, then:

   a. If the reactor is critical, it shall be brought to the hot shutdown condition utilizing normal operating procedures. The shutdown shall start not later than at the end of the specified time period.

b. If the reactor is subcritical, the reactor coolant system temperature and pressure shall not be increased more than 25°F and 100 psi, respectively, over existing values.

c. In either case, if the IVSW System is not restored to an operable status within an additional 48 hours, the reactor shall be brought to the cold shutdown condition utilizing normal operating procedures. The shutdown shall start no later than the end of the 48-hour period.

D. WELD CHANNEL AND PENETRATION PRESSURIZATION SYSTEM (WC & PPS)

1. The reactor shall not be brought above cold shutdown unless:

a. All required portions of the four WC & PPS zones are pressurized at or above 47 psig.

b. The uncorrected air consumption for the WC & PPS is less than or equal to 0.2% of the containment volume per day.

2. The requirements of 3.3.D.1 may be modified as follows:

a. Any one zone of the WC & PPS may be inoperable for a period not to exceed seven consecutive days.

b. The uncorrected air consumption for the WC & PPS may be in excess of 0.2% of the containment volume per day for a period not to exceed seven consecutive days.

3. If the WC & PP System is not restored to an operable status within the time period specified, then:

a. If the reactor is critical, it shall be brought to the hot shutdown condition utilizing normal operating procedures. The shutdown shall start no later than at the end of the specified time period.

b. If the reactor is subcritical, the reactor coolant system temperature and pressure shall not be increased more than 25°F and 100 psi, respectively, over existing values.

c. In either case, if the WC & PP System is not restored to an operable status within an additional 48 hours, the reactor shall be brought to the cold shutdown condition utilizing normal operating procedures. The shutdown shall start no later than the end of the 48-hour period.

E. COMPONENT COOLING SYSTEM

1. The reactor shall not be made critical unless the following conditions are met:

   a. Three component cooling pumps together with their associated piping and valves are operable.

   b. Two auxiliary component cooling pumps together with their associated piping and valves are operable.

   c. Two component cooling heat exchangers together with their associated piping and valves are operable.

2. During power operation, the requirements of 3.3.E.1 may be modified to allow one of the following components to be inoperable at any one time. If the system is not restored to meet the conditions of 3.3.E.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.E.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 of the three operable component cooling pumps may be out of service provided the pump is restored to operable status within 14 days.

b. An additional component cooling pump may be out of service provided a second pump is restored to operable status within 24 hours.

c. One auxiliary component cooling pump may be out of service provided the pump is restored to operable status within 24 hours and the other pump is demonstrated to be operable.

d. One component cooling heat exchanger or other passive component may be out of service for a period not to exceed 48 hours provided the system may still operate at design accident capability.

F. SERVICE WATER SYSTEM

1. DESIGNATED ESSENTIAL HEADER

a. The reactor shall not be above 350°F unless three service water pumps with their associated piping and valves are operable on the designated essential header.

b. When the reactor is above 350°F and one of the three service water pumps or any of its associated piping or valves is found inoperable, and an essential service water header that meets the requirements of 3.3.F.1.a. cannot be restored within 12 hours, the reactor shall be placed in the hot shutdown condition within the next 6 hours and subsequently cooled below 350°F using normal operating procedures.

2. DESIGNATED NON-ESSENTIAL HEADER

a. The reactor shall not be above 350°F unless two service water pumps with their associated piping and valves are operable on the designated non-essential header.

b. When the reactor is above 350°F and one of the two service water pumps or any of its associated piping or valves is found inoperable, and a non-essential service water header that meets the requirements of 3.3.F.2.a cannot be restored within 24 hours, the reactor shall be placed in the hot shutdown condition within the next 6 hours and subsequently cooled below 350°F using normal operating procedures.

3. INTERCONNECTION OF HEADERS

Isolation shall be maintained between the essential and non-essential headers at all times when the reactor is above 350°F except for a period of up to 8 hours when the header may be connected to facilitate safety-related activities.

4. SERVICE WATER INLET TEMPERATURE

a. The reactor shall not be above 350°F unless the service water inlet temperature is less than or equal to 95°F, or

b. When the reactor is above 350°F and the service water inlet temperature exceeds 95°F, the reactor shall be placed in the hot shutdown condition within the next 7 hours and subsequently cooled below 350°F using normal operating procedures.

c. The provisions of Specification 3.0.1 do not apply.

5. SERVICE WATER INLET TEMPERATURE MONITORING INSTRUMENTATION

a. The service water inlet temperature monitoring instrumentation shall measure the Hudson River water temperature at the Indian Point Unit No. 2 intake structure,

b. The service water inlet temperature monitoring instrumentation shall be operable when intake water temperature, averaged over a 24 hour period, reaches 80°F, and when the reactor is above 350°F,

c. When the requirements of Specification 3.3.F.5.b apply, temperature measurements shall be taken every 4 hours up to and including a service water inlet temperature of 90°F; when the service water inlet temperature exceeds 90°F, temperature measurements shall be taken once an hour,

d. If the service water inlet temperature monitoring instrumentation is declared inoperable, it shall be either restored to operable status or alternative measurements shall be taken with a calibrated portable instrument within the applicable measurement time frame requirements of Specification 3.3.F.5.c, and

e. If the requirements of Specification 3.3.F.5.d cannot be met, the reactor shall be placed in the hot shutdown condition within the next 7 hours and subsequently cooled below 350°F using normal operating procedures.

G. HYDROGEN RECOMBINER SYSTEM AND POST-ACCIDENT CONTAINMENT VENTING SYSTEM

1. The reactor shall not be made critical unless the following conditions are met:

a. Both hydrogen recombiner units together with their associated piping, valves, oxygen supply system and control system are operable, with the exception of one recombiner unit's equipment located outside the containment which may be inoperable, provided it is under repair and can be made operable if needed.

b. The post-accident containment venting system is operable.

c. Hydrogen and oxygen supplies shall not be connected to the hydrogen recombiner units except under conditions of an accident or those specified in Specification 4.5.C.1.

2. During power operation, the requirements of 3.3.G.1 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 3.3.G.1 within the time specified, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures.

   a. One hydrogen recombiner unit or its associated flow path, or oxygen supply system or control system may be inoperable for a period not to exceed thirty days, provided the other recombiner unit and the post-accident containment venting system are operable.

   b. The post-accident containment venting system may be inoperable for a period not to exceed thirty days provided that both hydrogen recombiners are operable.

H. CONTROL ROOM AIR FILTRATION SYSTEM

1. The control room air filtration system shall be operable at all times when containment integrity is required.

2. From the date that the control room air filtration system becomes and remains inoperable for any reason, operations requiring containment integrity are permissible only during the succeeding 3.5 days. At the end of this 3.5 days period, if the conditions for the control room air filtration system cannot be met, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the conditions are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

3. Two independent toxic gas detection systems, each capable of detecting chlorine, anhydrous ammonia, and hydrogen cyanide, shall be operable at all times except as specified in 3.a, 3.b, or 3.c below. The alarm/trip setpoint for each toxic gas system shall be adjusted to actuate at a toxic gas concentration of less than or equal to 3.5 ppm.

   a. With one toxic gas detection system inoperable, restore the inoperable detection system to operable status within 7 days.

   b. If 3.a above cannot be satisfied within the specified time, then, within the next 6 hours, initiate and maintain operation of the control room ventilation system in the recirculation mode of operation.

   c. With both toxic gas detection systems inoperable for any one toxic gas, within one hour initiate and maintain operation of the control room ventilation in the recirculation mode of operation.

I. CABLE TUNNEL VENTILATION FANS

1. The reactor shall not be made critical unless the two cable tunnel ventilation fans are operable.

2. During power operation, the requirement of 3.3.I.1 may be modified to allow one cable tunnel ventilation fan to be inoperable for seven days, provided the other fan is operable.

Basis

The normal procedure for starting the reactor is, first, to heat the reactor coolant to near operating temperature by running the reactor coolant pumps. The reactor is then made critical by withdrawing control rods and/or diluting boron in the coolant[1]. With this mode of start-up, the energy stored in the reactor coolant during the approach to criticality is substantially equal to that during

power operation, and therefore the minimum required engineered safeguards and auxiliary cooling systems are required to be operable. During low-temperature physics tests there is a negligible amount of stored energy in the reactor coolant; therefore, an accident comparable in severity to the Design Basis Accident is not possible, and the engineered safeguards systems are not required.

When the reactor is critical, the probability of sustaining both a major accident and a simultaneous failure of a safeguards component to operate as designed is necessarily very small. Thus operation with the reactor critical with minimum safeguards operable for a limited period does not significantly increase the probability of an accident having consequences which are more severe than the Design Basis Accident.

The operable status of the various systems and components is to be demonstrated by periodic tests, defined by Specification 4.5. A large fraction of these tests will be performed while the reactor is operating in the power range. If a component is found to be inoperable, it will be possible in most cases to effect repairs and restore the system to full operability within a relatively short time. Inoperability of a single component does not negate the ability of the system to perform its function[2], but it reduces the redundancy provided in the reactor design and thereby limits the ability to tolerate additional equipment failures. To provide maximum assurance that the redundant component(s) will operate if required to do so, the redundant component(s) are to be tested prior to initiating repair of the inoperable component. If it develops that (1) the inoperable component is not repaired within the specified allowable time period, or (2) a second component in the same or related system is found to be inoperable, the reactor will initially be put in the hot shutdown condition to provide for reduction of the decay heat from the fuel and consequent reduction of cooling requirements after a postulated loss-of-coolant accident. This will also permit improved access for repairs in some cases. After a limited time in hot shutdown, if the malfunction(s) are not corrected, the reactor will be placed in the cold shutdown condition, utilizing normal shutdown and cooldown procedures. In the cold shutdown condition there is no possibility of an accident that would release fission products or damage the fuel elements.

The plant operating procedures require immediate action to effect repairs of an inoperable component, and therefore in most cases repairs will be completed in less than the specified allowable repair times. The specified repair times do not apply to regularly scheduled maintenance of the engineered safeguards systems, which is normally to be performed during refueling shutdowns. The limiting times to repair are based on two considerations:

1. assurance with high reliability that the safeguard system will function properly if required to do so, and

2. allowance of sufficient time to effect repairs using safe and proper procedures.

Assuming the reactor has been operating at full-rated power for at least 100 days, the magnitude of the decay heat decreases after initiating hot shutdown. Thus the requirement for core cooling in case of a postulated loss-of-coolant accident while in the hot shutdown condition is significantly reduced below the requirements for a postulated loss-of-coolant accident during power operation. Putting the reactor in the hot shutdown condition significantly reduces the potential consequences of a loss-of-coolant accident, and also allows more free access to some of the engineered safeguards components in order to effect repairs.

Failure to complete repairs within 48 hours of going to the hot shutdown condition is considered indicative of a requirement for major maintenance, and therefore in such a case the reactor is to be put into the cold shutdown condition.

Valves 1810, 744 and 882 are kept in the open position during plant operation to assure that flow passage from the refueling water storage tank will be available during the injection phase of a loss-of-coolant accident. As an additional assurance of flow passage availability, the valve motor operators are de-energized to prevent an extremely unlikely spurious closure of these valves to take place. This additional precaution is acceptable since failure to manually re-establish power to close valves 1810 and 882, following the injection phase, is tolerable as

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.

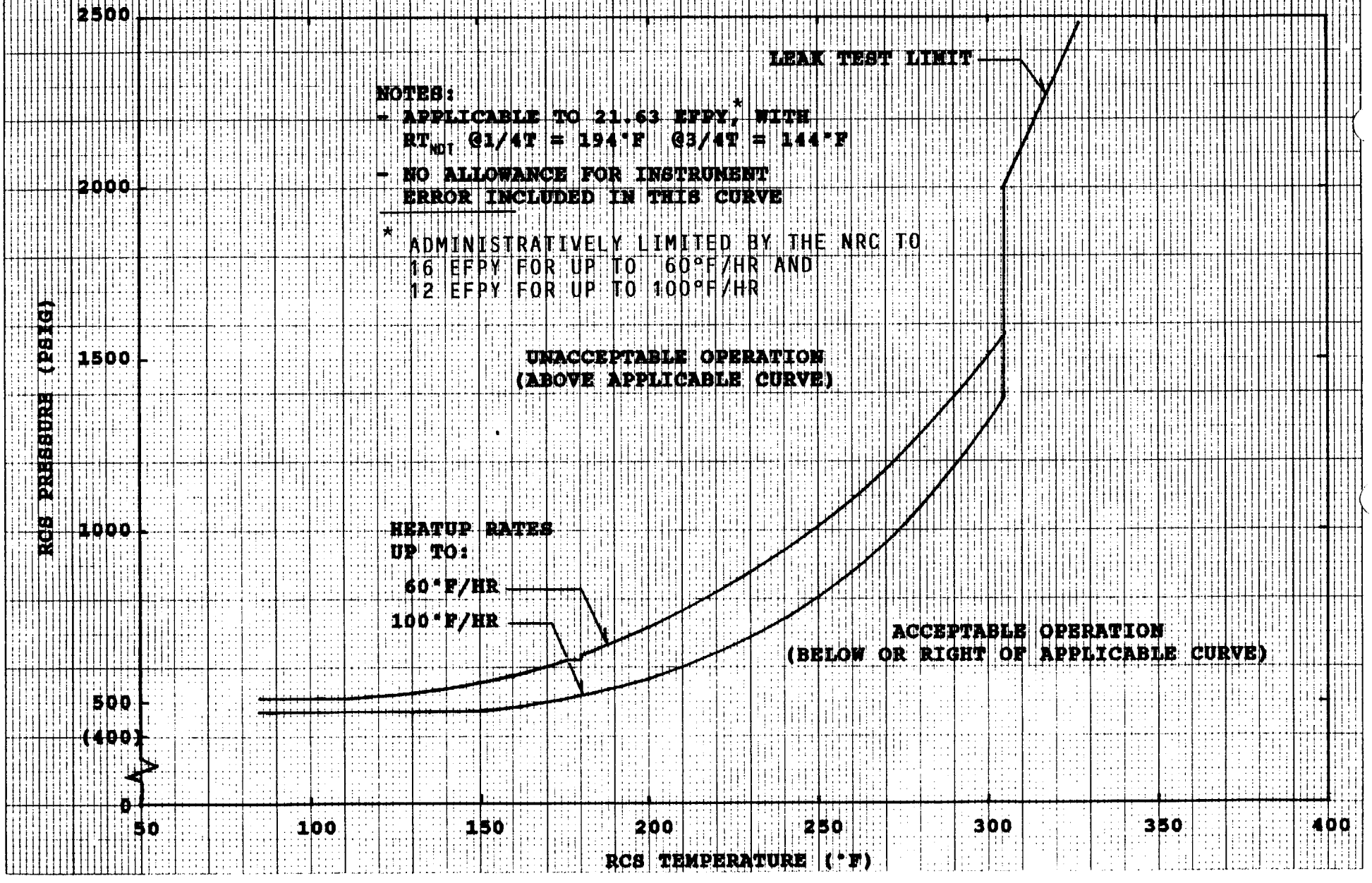
Reference

UFSAR Section 4

---

* Curves have been approved but administratively limited by the NRC to 16 EFPY for up to 60°F/hr. and 12 EFPY for up to 100°F/hr. pending acceptance of the use of the Raju-Newman method in the generation of heatup and cooldown curves.

FIGURE 4.3-1
VESSEL LEAK TEST LIMITATIONS



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 155 TO FACILITY OPERATING LICENSE NO. DPR-26

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

1.0 INTRODUCTION

By letter dated March 7, 1991, the Consolidated Edison Company of New York, Inc. (the licensee) requested permission to revise the pressure/temperature (P/T) limits and the Overpressure Protection System (OPS) parameters in the Indian Point 2 Technical Specifications, Section 3.1. The proposed P/T limits were requested for 21.63 effective full power years (EFPY). The request was a response to Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Effect on Plant Operations," which recommends Regulatory Guide (RG) 1.99, Rev. 2, be used in calculating P/T limits. The P/T limits in the current Technical Specifications are valid for 15 EFPY. The plant has operated to about 10.7 EFPY as of June 3, 1991. On August 2, 1991, the licensee submitted clarifying supplemental information which put administrative limits on the P/T curves. For heatup and cooldown rates of up to 60°F/hr. the limit is 16 EFPY and for heatup and cooldown rates of up to 100°F/hr. the limit is 12 EFPY. An administrative limit of 16 EFPY was placed on the curves for the Overpressure Protection System (OPS). This information did not change the initial proposed no significant hazards consideration determination.

To evaluate the P/T limits, the staff uses the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; the ASTM Standards and the ASME Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); RG 1.99, Revision 2; Standard Review Plan (SRP) Section 5.3.2; and Generic Letter 88-11.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide Technical Specifications for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the technical specifications. The P/T limits are among the limiting conditions of operation in the technical specifications for all commercial nuclear plants in the U.S. Appendices G and H of 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits. An acceptable method for constructing the P/T limits is described in SRP Section 5.3.2.

9111200275 911021
PDR ADOCK 05000247
P PDR

Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE). Generic Letter 88-11 requested that licensees and permittees use the methods in RG 1.99, Revision 2, to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H of 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline.

By letter dated March 30, 1991, the licensee submitted the final report for the reactor vessel material surveillance analysis of capsule V. The experimental portion of the capsule analysis was performed by the Southwest Research Institute (SRI) and the theoretical neutron transport analysis was performed by Westinghouse and reported by SRI. The NRC staff evaluates the fast neutron fluence estimates by reviewing the procedures and methodology for the measured and calculated values.

2.0 EVALUATION

2.1.1 Neutron Dosimetry

The handling, the weighing and counting of the neutron counters was performed according to SRI procedures. Calibration sources from the National Institute of Standards and Technology were used for the counting. The saturation activities were determined from the measured activities accounting for the irradiation intervals and the decay time. We find that acceptable methods were used in neutron dosimetry, thus, we find the dosimetry acceptable.

2.1.2 Neutron Transport Calculations

These calculations use the measured counter activities to estimate the fast neutron flux (E greater than 1.0 MeV) at the surveillance capsule and the azimuthal flux at the inner surface of the pressure vessel. Westinghouse performed a two-dimensional S_n transport analysis, using the Department of Transportation (DOT) two-dimensional discrete ordinates transport code. The

geometrical transport code for computer (SAIL) cross-section library was used which is based on ENDF/F-IV. The calculation was carried out with P_3 scattering approximation and an S_8 angular quadrature approximation.[3] The calculation assumed a thermal power level of 2758 MWth. The core power distribution for each cycle was taken from the Indian Point 2 reload reports. Both a forward and adjoint calculations were performed.

The above calculations conform to staff requirements and to accepted practice. In addition there is excellent agreement between the measured and calculated fast neutron flux values, therefore, they are acceptable.

2.2 P/T Limits

The licensee used the Raju-Newman (R-N) method to calculate the proposed P/T limits which are less conservative than the P/T limits calculated by the Appendix G method in Section III of the ASME Code as recommended in SRP 5.3.2. The R-N method yields a lower stress intensity factor due to thermal stresses (KIthermal) than the Appendix G method. The R-N method is currently under review for possible inclusion in Section XI of the ASME Code. The NRC staff, therefore, considers that it is premature presently to allow the use of the R-N method in the P/T limits calculation.

Nevertheless, the staff determined that the proposed limits may be applicable for limited EFPY. Hence, the staff used SRP 5.3.2 to estimate an equivalent EFPY for the proposed P/T limits.

The staff estimated that the proposed P/T curve that uses 60°F/hour heatup/cooldown rate will be valid up to 16 EFPY and the 100°F/hour curve will be valid up to 12 EFPY. These effective EEPYs are applicable to the heatup limits, cooldown limits, and leak test limits in proposed Figures 3.1.B-1, 3.1.B-2, and 4.3-1, respectively.

The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff has determined that the material with the highest (limiting) ART at both 12 EFPY and 16 EFPY was intermediate shell plate B2002-3 with 0.25% copper (Cu), 0.60% nickel (Ni), and an initial RT_{ndt} of 21°F.

For the limiting beltline material, plate B2002-3, the staff calculated the ART to be 174.2°F at the 1/4T location (T = reactor vessel beltline thickness) and 127.9°F at the 3/4T location at 16 EFPY. The staff used a neutron fluence of 4.1E18 n/cm at 1/4T and 1.5E18 n/cm at 3/4T.

At 12 EFPY, the staff calculated the ART to be 160.4°F at 1/4T and 116.7°F at 3/4T. The staff used a neutron fluence of 3.1E18 n/cm at 1/4T and 1.1E18 n/cm at 3/4T. The ARTs were determined by the least squares extrapolation method in RG 1.99 using the Indian Point 2 surveillance data.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Based on the flange reference temperature of 60°F, the NRC staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

Section IV.B of Appendix G requires that the predicted Charpy USE at end of life be above 50 ft-ib. The beltline material with the lowest initial USE is intermediate shell plate B2003-1 with 71 ft-lb. Using the method in RG 1.99, Rev. 2, the predicted Charpy USE of the plate metal at the end of life will be 51 ft-lb. This satisfies the 50 ft-lb requirement.

The licensee has removed four surveillance capsules from Indian Point 2. The results from capsules T, Y, Z, and V were published in Southwest Research Institute Reports SwRI 02-4531, SwRI 02-5212, SwRI 06-7379, and SwRI 17-2108, respectively. Surveillance capsules Y and V contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal. Surveillance capsules T and Z contained Charpy impact specimens and tensile specimens made from base metal and HAZ metal. The reactor vessel material surveillance program satisfies Appendix H to 10 CFR Part 50.

2.3 OPS Parameters

Overpressure protection at low temperatures is provided by the Power Operated Relief Valves (PORVs) on the pressurizer. These PORVs are set at pressures low enough to prevent violation of the Appendix G heatup and cooldown curves should a reactor coolant system (RCS) pressure transient occur during low temperature operations. The licensee, in its March 17, 1991, submittal, confirmed that the results of previous analyses of the most limiting overpressure transients in determining the PORV setpoints for the OPS remain valid. The PORV setpoint limits have been set by two design criteria. These are the limiting transients for mass addition and heat addition.

The licensee has referred to the Westinghouse generic thermal-hydraulic analyses for the limiting transients as the methodology employed in the development of the OPS setpoints described in the current technical specifications. The design basis mass addition transients are an inadvertent actuation of one or more charging pumps with loss of letdown and the startup of a high pressure safety injection (SI) pump. The proposed Technical Specification 3.1.A.1 (Reactor Coolant Pump) allows only one SI and three charging pumps to be energized when the RCS temperature is less than or equal to 305°F, which is the maximum RCS temperature for which OPS is operable. The analysis was performed to determine the pressure overshoot past the OPS setpoint such that the Appendix G curves are not exceeded during the transient.

The limiting heat addition transient is the startup of one reactor coolant pump (RCP) with the steam generator secondary fluid temperature hotter than the RCS fluid temperature. The heat addition transient with OPS operable was previously analyzed assuming a 40°F temperature difference between the steam generator and the RCS. A reactor coolant pump startup in one loop was assumed to maximize the heat transfer effect. As was the case for the mass addition transient, the pressure overshoot was calculated such that the Appendix G P/T curves will not be exceeded.

Considering the above factors, the NRC staff concludes that the assumptions applied to the licensee's analyses are reasonably conservative and acceptable.

Revised Technical Specification Figures 3.1.A-1, 3.1.A-2, and 3.1.A-3 are provided to identify the operating restrictions for pressurizer pressure, pressurizer level and RCS temperature resulting from the new P/T limits. The changes are identification of the new OPS enable temperature of 305°F, a 30°F temperature difference between the steam generator secondary side and the RCS, and a RCS temperature less than or equal to 275°F in Table 3.1.A-2. These changes correspond to or are bounded by the assumptions in the supporting safety analyses and are acceptable.

The licensee has proposed to replace the current pressure-temperature limits for criticality in Section 3.1.C (Minimum Conditions for Criticality) with a fixed temperature limit. This limit is imposed in accordance with requirements of 10 CFR Part 50, Appendix G, as amended February 2, 1976. The selected temperature of (450°F) bounds the allowed pressure-temperature range for criticality and is acceptable.

Other technical specification sections are changed to identify the OPS enable temperature of 305°F (up from 295°F).

The licensee proposed OPS enable temperature change in Technical Specification 3.1.A.1, et al., and the associated Bases section reflect the above discussed OPS alignment temperatures and the heatup and cooldown rates identified by the updated Figures 3.1.B-1 and 3.1.B-2 in Technical Specification 3.1.B. The NRC staff finds that they are reasonably conservative and acceptable.

3.0 CONCLUSION

The NRC staff considers that it is premature to allow the use of the Raju-Newman method in the P/T limits calculation because the method is currently under review for possible inclusion in Section XI of the ASME Code. The staff will not approve the proposed P/T limits for 21.63 EFPY as requested; however, the NRC staff approves the P/T curves as resubmitted by the licensee with administrative limits of 16 EFPY for heatup and cooldown rates up to 60°F/hr. and of 12 EFPY for heatup and cooldown rates up to 100°F/hr. The NRC staff also approves the parameter changes related to the OPS.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (56 FR 13661). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. March 30, 1990, Letter from S. B. Bram (ConEd) to USNRC Document Control Desk, Subject: NRC Generic Letter No. 88-11.

2. Regulatory Guide 1.99, Radiation Embrittlement of Reactor Vessel Material, Revision 2, May 1988.

3. NUREG-0800, Standard Review Plan, Section 5.3.2 Pressure-Temperature Limits.

4. March 7, 1991, Letter from S. B. Bram (ConED) to UNNRC Document Control Desk, Subject: Indian Point Unit No. 2, Docket No. 50-247.

5. E. B. Norris, "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2: Analysis of Capsule T," SwRI 02-4531, June 30, 1977.

6. E. B. Norris, "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2: Analysis of Capsule Y," SwRI 02-5212, November 1980.

7. E. B. Norris, "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2: Analysis of Capsule Z," SwRI 06-7379, April 1984.

8. F. A. Iddings, et al, "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2: Analysis of Capsule V," SwRI 17-2108, March 1990.

9. NRC Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Material and Its Impact on Plant Operations," July 12, 1988.

10. "Pressure Mitigating Systems Transient Analysis Results," Westinghouse Owners' Group on RCS Overpressurization, July 1977.

11. WCAP-12796, "Heatup and Cooldown Limit Curves for the Consolidated Edison Company Indian Point Unit 2 Reactor Vessel," N. K. Ray, Westinghouse Electric Corporation.

12. WCAP-7924A, "Basis for Heatup and Cooldown Limit Curves," W. S. Hazelton, et al., April 1975.

Principal Contributor:
J. Tsao
M. McCoy
L. Lois

Date: October 21, 1991