LIST OF FIGURES

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 280°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 ≤ 280°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. Injection into the RCS from the startup of:
   - Three charging pumps, or
   - One safety injection pump and 2 charging pumps

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 280°F by: (1) restricting the number of charging and safety injection pumps that are capable of injecting into the RCS 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 40°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 249°F and 280°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. The analysis for Figure 3.1.A-1 includes the time delay associated with the opening of the PORVs, the difference in elevation between the PORVs and the RCS pressure sensors, a 5°F temperature margin, a 10 psi pressure margin, and the error caused by non-uniform RCS metal and water temperatures during heat addition transients. Instrument error and bias 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 error and bias.

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).

(d) Similarly, when OPS is not available, the limitations on RCS pressure and level, and secondary-to-primary water temperature difference, include the effects of differences in elevation between the pressurizer liquid level and the RCS pressure sensors, a 5° F temperature margin, a 10 psi pressure margin, and the error caused by non-uniform RCS metal and water temperature. Allowances for other instrument errors and bias are not included in the Tech Spec values; but are included in the curves and procedures that implement the Tech Spec limits on operation.

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.

Reference

(1) UFSAR Section 14.1.12
(2) UFSAR Section 9.3.1
(3) UFSAR Section 14.1.8
(4) NET-177-01, Indian Point Unit 2 Overpressure Protection System (OPS) Thermal Hydraulic Analysis, Setpoint Development And Technical Specification Revision for 25 EFPY

Table 3.1.A-2

OPS Operability Requirements

Reactor Coolant Pumps

With OPS operable at or below 280°F, a reactor coolant pump can be started (or jogged) with no other reactor coolant pumps operating if:

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

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

- o RCS temperature is less than or equal to 249°F
- o Pressurizer level is between 30 - 85% of span.

With OPS inoperable at or below 280°F, a reactor coolant pump can be started (or jogged) with no other reactor coolant pumps operating if:

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

(2) The temperature of all steam generators is less than or equal to 40°F higher than the RCS temperature and RCS pressure operating restrictions are as specified in figure 3.1.A-5, or

(3) The temperature of all steam generators is less than or equal to 100°F higher than RCS temperature and RCS pressure operating restrictions are as specified in Figure 3.1.A-6.

Table 3.1.A-2

OPS Operability Requirements

Safety Injection and Charging Pumps

NOTE:

1. If conditions require the use of Safety Injection pumps for makeup in the event of a loss of RCS inventory, the pumps can be made capable of injecting into the RCS through manual actions.
2. With charging pumps operating for normal RCS makeup, one SI pump may be made capable of injecting into the RCS as needed to support abnormal operations such as emergency boration or response to a loss of RHR cooling.

With OPS operable at or below 280°F, no more than three charging pumps may be capable of injecting into the RCS; OR, for the reduced PORV actuation curve (See Figure 3.1.A-1), one safety injection and two charging pumps may be capable of injecting into the RCS.

OPS is not required to be operable at or below 280°F if either the conditions of Column II or the conditions of Column III below are met for the maximum number of SI and Charging pumps capable of injecting into the RCS specified in Column I:

Column I Maximum Number of SI and Charging Pumps Capable of Injecting into the RCS		Column II Operating Restrictions (pressurizer pressure, pressurizer level, and RCS temperature)	Column III Vent Area to Containment Atmosphere (square inches)
SI	Charging		
0	1	See Figure 3.1.A-2	2.00 (or 1 PORV fully open)
0	2	See Figure 3.1.A-3	2.00 (or 1 PORV fully open)
0	3	See Figure 3.1.A-4	2.00 (or 1 PORV fully open)
1	0, 1, 2 or 3	Use Column III only	2.00 (or 1 PORV fully open)
2	0, 1, 2 or 3	Use Column III only	5.00 (or 2 PORVs fully open)
3	0, 1, or 2	Use Column III only	5.00 (or 2 PORVs fully open)
3	3	Use Column III only	5.00

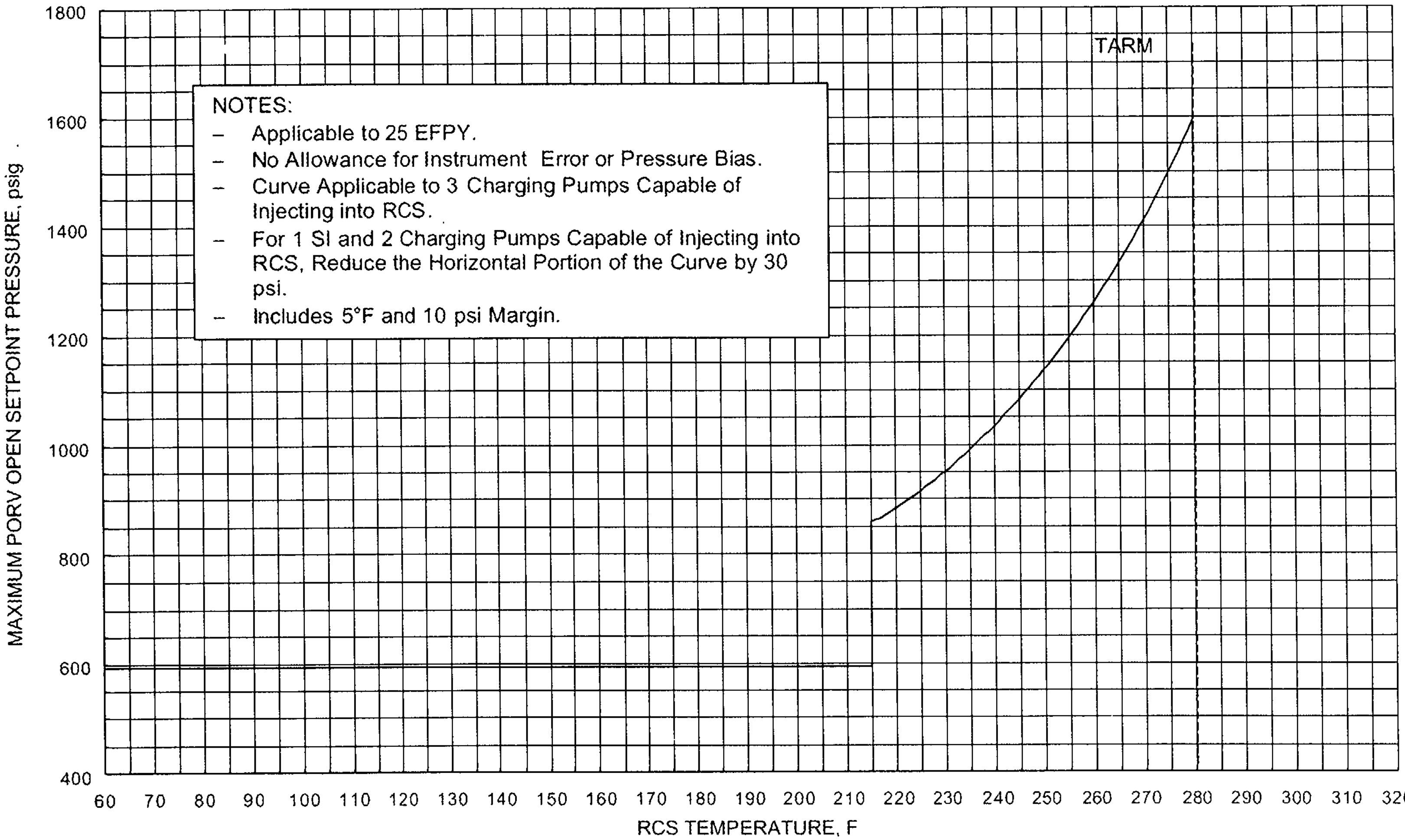

Figure 3.1.A-1 PORV Open Pressure

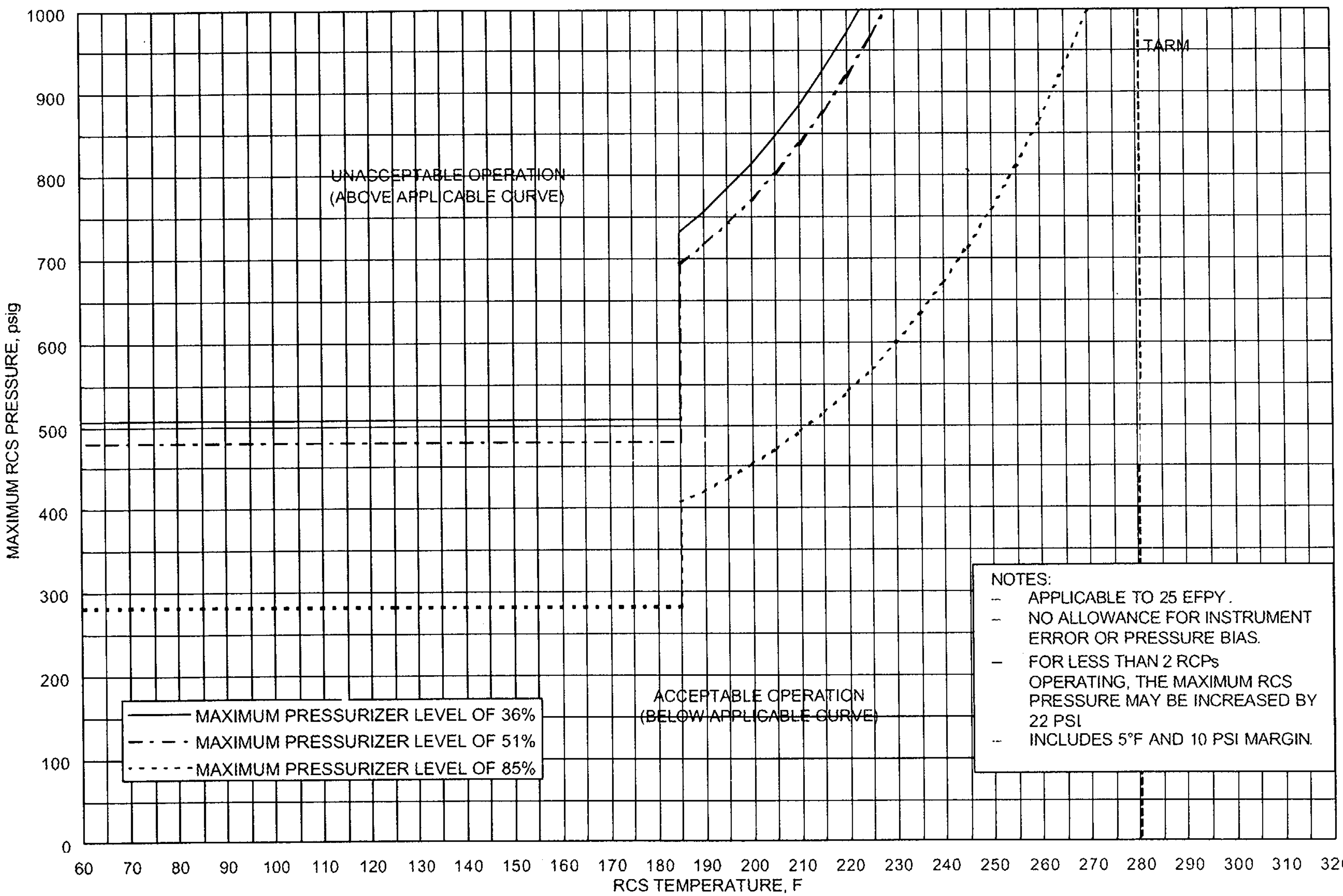

Figure 3.1.A-2 Maximum RCS Pressure: OPS Inoperable and 1 Charging Pump Capable of Injecting into the RCS

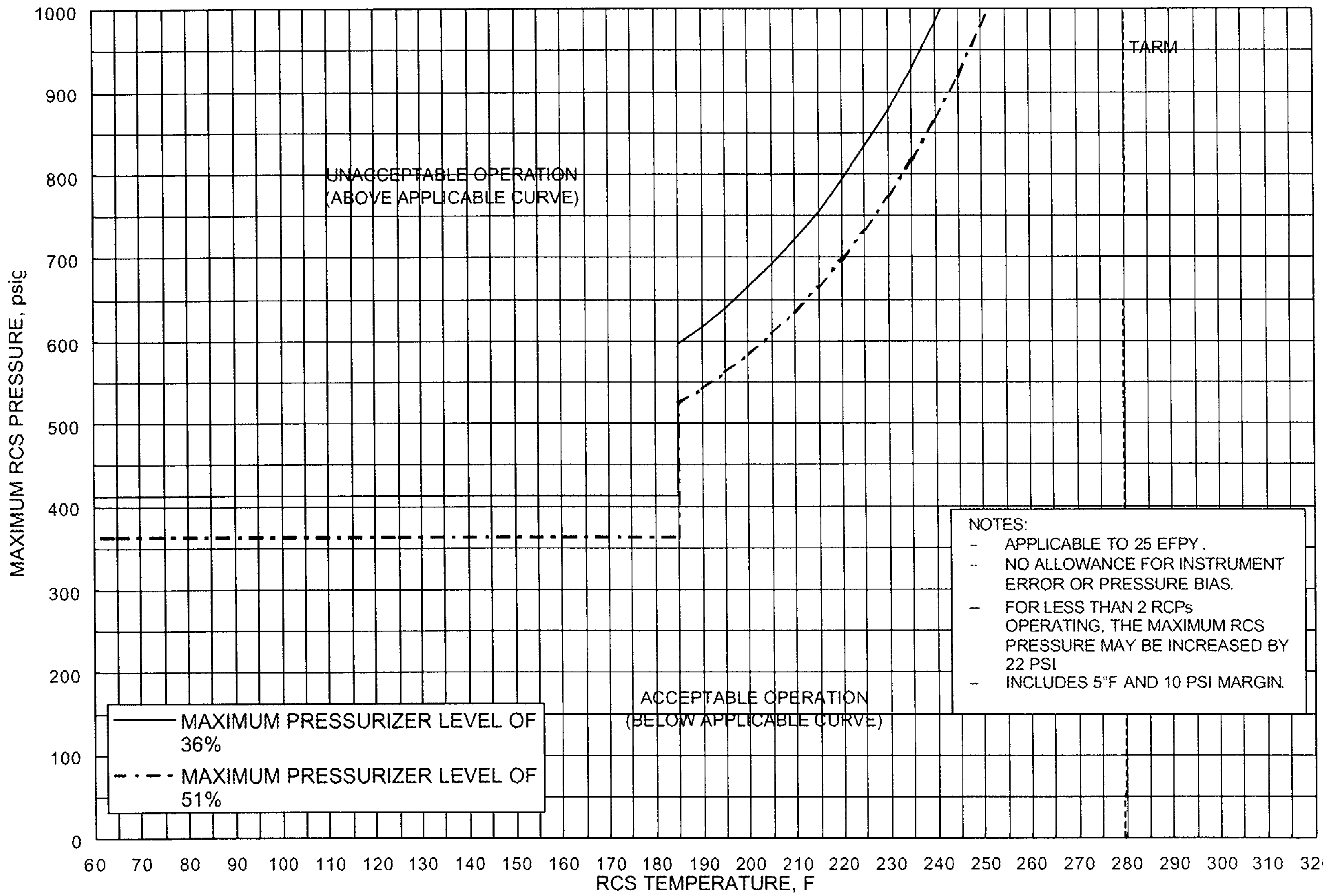

Figure 3.1.A-3 Maximum RCS Pressure: OPS Inoperable and 2 Charging Pumps Capable of Injecting into the RCS

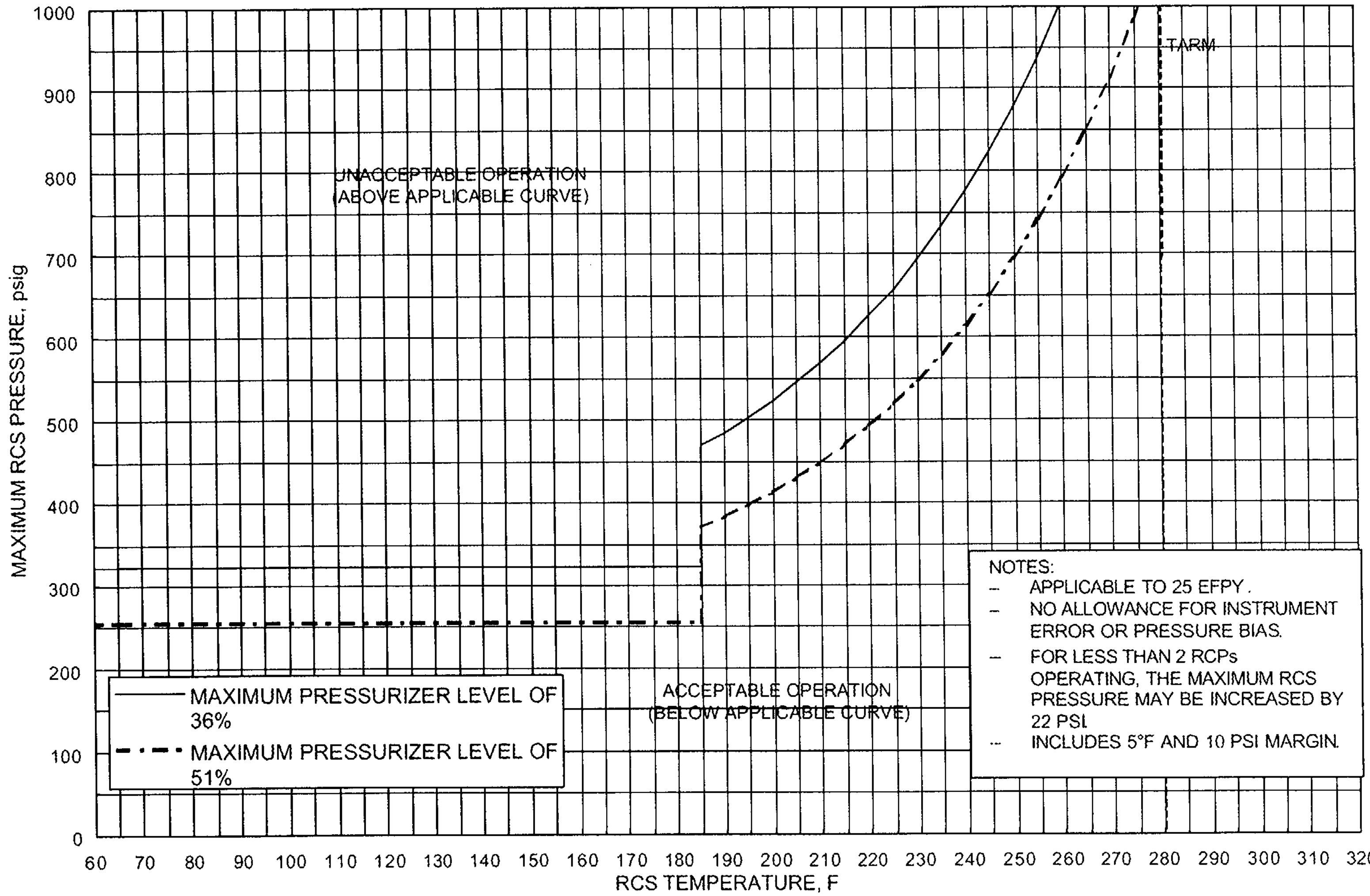

Figure 3.1.A-4 Maximum RCS Pressure: OPS Inoperable and 3 Charging Pumps Capable of Injecting into the RCS

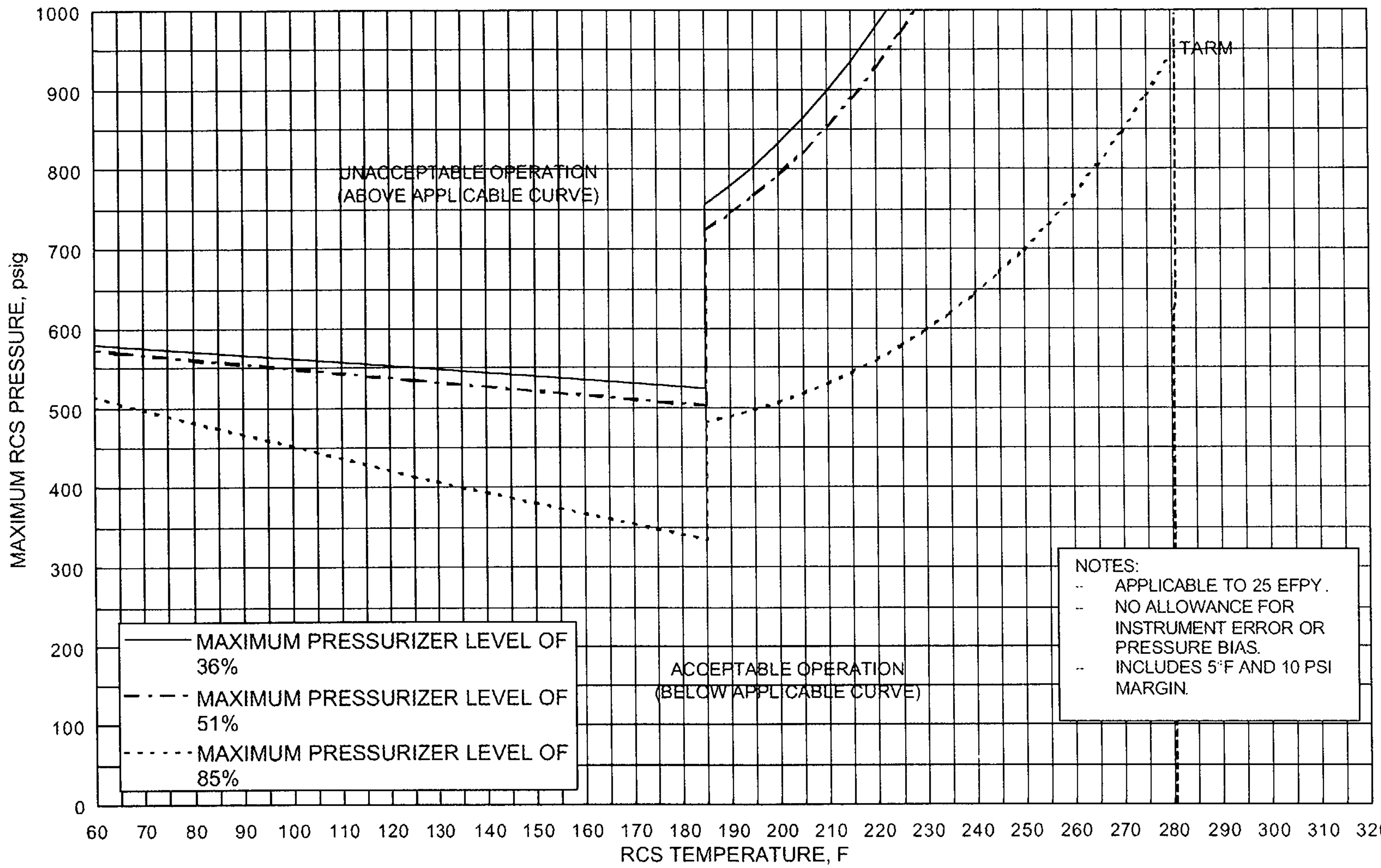

Figure 3.1.A-5 Maximum RCS Pressure: OPS Inoperable and Start of 1 RCP with SGs 40°F Hotter than RCS

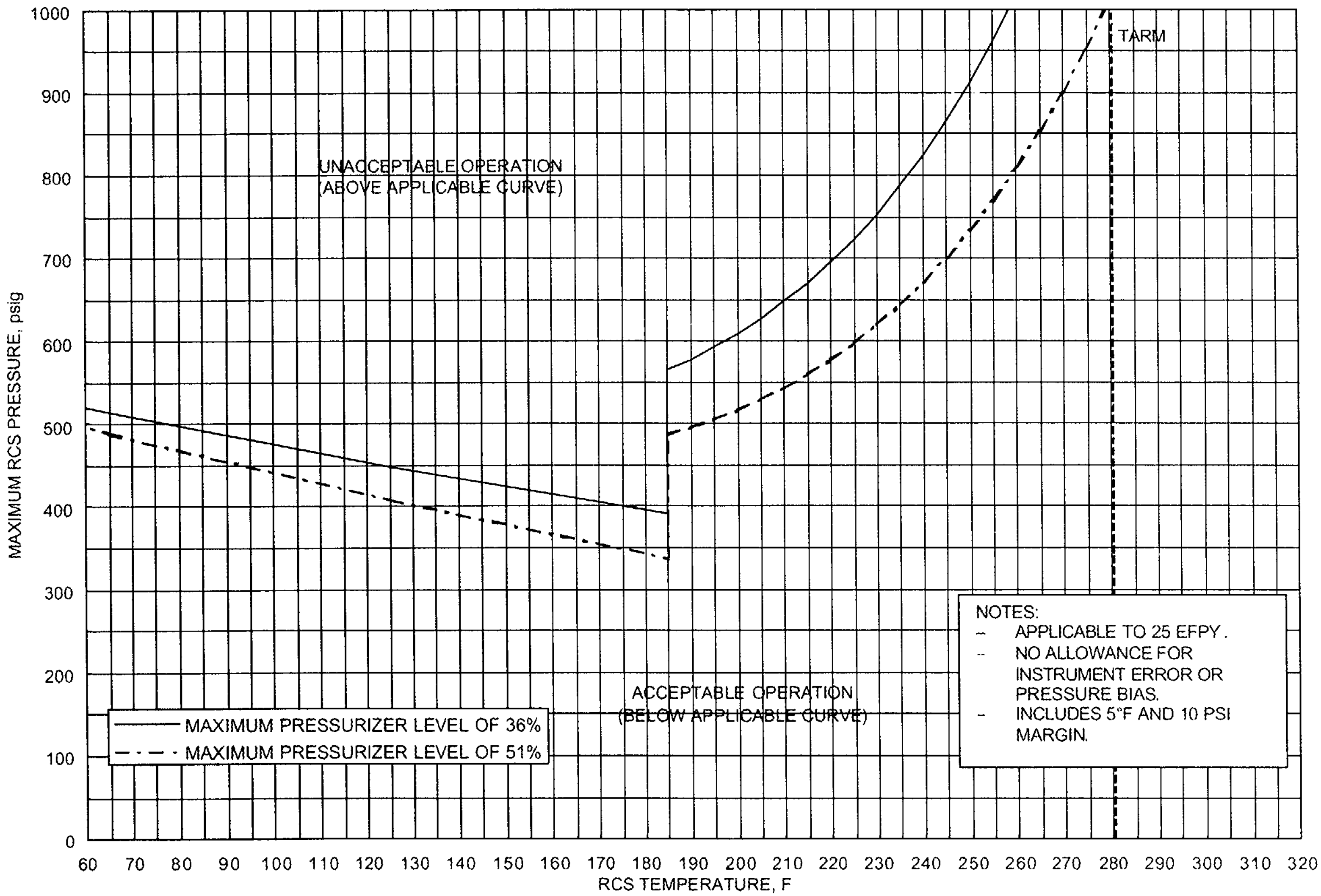

Figure 3.1.A-6 Maximum RCS Pressure: OPS Inoperable and Start of 1 RCP with SGs 100°F Hotter than RCS

B. HEATUP AND COOLDOWN

Specifications

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) averaged over one hour shall be limited in accordance with Figure 3.1.B-1 and Figure 3.1.B-2 for the service period up to 25 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 presented 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 NRC approved methods and results of surveillance specimen testing as covered in WCAP-7323[7] and as specified in Specification 3.1.B.3 below. The order of specimen removal may be modified based on the results of testing of previously removed specimens. The NRC will be notified in writing as to any deviations from the recommended removal schedule no later than six months prior to scheduled specimen removal.

3. The reactor vessel surveillance program* includes six specimen capsules to evaluate radiation damage based on pre-irradiation and post-irradiation tensile and Charpy V notch (wedge open loading) testing of specimens.

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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.

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

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].

Heatup and cooldown limit curves define acceptable regions for normal operation. The heatup and cooldown limit curves establish operating limits that provide a margin to non-ductile failure of the reactor coolant pressure boundary. The reactor vessel is the most limiting component most subject to non-ductile failure. Therefore, the reactor vessel limits control the heatup and cooldown limits provided in Figures 3.1.B-1 and 3.1.B-2. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

Reactor Vessel Fracture Toughness Properties

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.

10CFR50 Appendix G requires the establishment of P/T limits for specific material fracture toughness requirements of the Reactor Coolant Pressure Boundary materials. 10CFR50 Appendix G requires an adequate margin to non-ductile failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section XI, Appendix G. IP2 has obtained exemptions from the requirements of 10CFR50 Appendix G to allow the use of ASME code Cases N-588 and N-640 in conjunction with ASME Code Section XI, Appendix G. The neutron embrittlement effect on the material toughness is reflected by increasing the nil ductility reference temperature (RTNDT) as exposure to neutron fluence increases. The actual shift in the RTNDT of the vessel beltline material is established periodically in accordance with the Regulatory Guide 1.99 Revision 2 requirements. The operating P/T limit curves are periodically adjusted based on the evaluation findings and the recommendations of Regulatory Guide 1.99.

The actual shift in RT_{NDT} is established periodically during plant operation by testing vessel material samples, which are irradiated cumulatively by securing them on the outside surface of the thermal shield within the active core region. These samples are removed using the specified schedule and evaluated according to the requirements of 10CFR50 Appendix H.

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].

The 25 EFPY heatup and cooldown curves are based upon a maximum fluence of 1.02×10^{19} n/cm^2 at the inner reactor vessel surface (45° angle, vessel belt line). This fluence is based upon plant operation for a nominal period of 25 EFPYs (Operation up to Cycle 9 for 9.63 EFPYs at 2758 MWt power level, operation during Cycle 10 for 1.13 EFPY at 2758 or 2948 MWt, operation from Cycle 11 until Cycle 15 at 3071.4 MWt and operation beyond Cycle 15 at 3216 MWt). The curves are based on operation beyond cycle 15 with T average of 579°F. Any changes in the operating conditions could result in a change to the allowable EFPYs, since the fluence (or ΔRT_{NDT} due to irradiation) is the controlling factor in the generation of these curves.

The ART at a fluence of 1.02×10^{19} n/cm^2, (nominal 25 EFPYs of operation) is to be 200°F at the 1/4 T vessel wall location for the Intermediate to Lower shell girth weld and 145°F at the 3/4 T vessel wall locations, for Plate B2002-3.

Heatup and Cooldown Curves

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods discussed in detail in WCAP-15629[13] and as modified by ASME Code Cases N-588 and N-640. Also, the 1995 Edition with the 1996 Addenda of the ASME Section XI, Appendix G was used for the operating period of up to 25 EFPY.

The heatup and cooldown curves for operation up to 25 EFPY have been computed on the basis of the RT_{NDT} for both the Intermediate to Lower shell girth weld and Plate B2002-3. It was determined that heatup and cooldown curves based on Plate B2002-3 are more limiting than those calculated for the girth weld. Hence, the Heatup and Cooldown Limitation curves of Figures 3.1.B-1 and 3.1.B-2 are based on the Plate B2002-3 being the limiting vessel beltline material.

The Heatup and Cooldown curves of Figures 3.1.B-1 and 3.1.B-2 are not adjusted to account for pressure and temperature instrument error. Those adjustments are made in procedures implementing the Heatup and Cooldown curves of Figures 3.1.B-1 and 3.1.B-2.

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 XI, 1996 Edition, Appendix G

(6) DELETED

(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-15629, Revision 1, Indian Point Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation

(14) WCAP-14040-NP-A, Rev. 2, Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves.

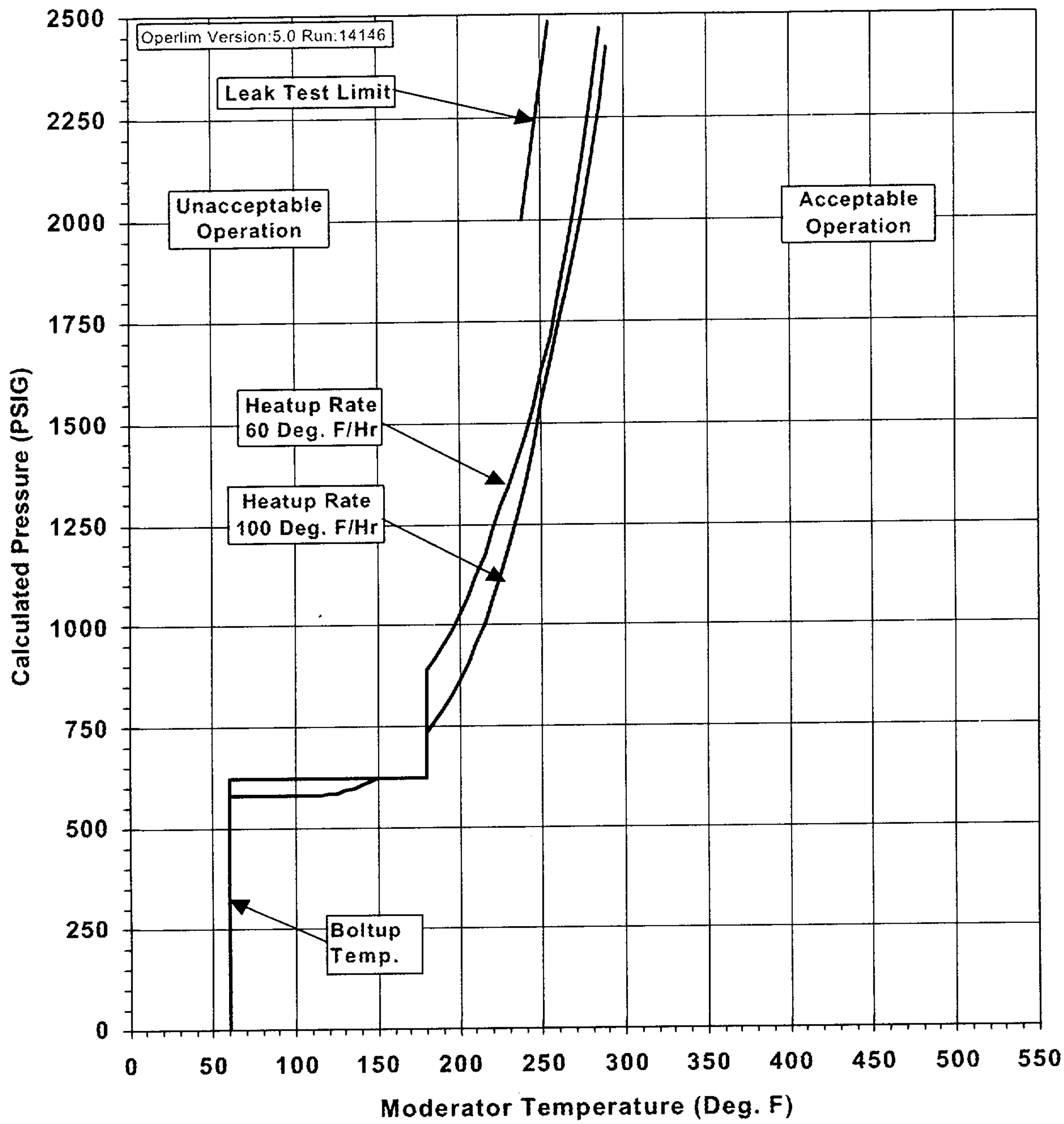

Figure 3.1.B-1 Reactor Coolant System Heatup and Leak Test Limitations Applicable for the First 25 EFPY (Without Margins for Instrumentation Errors)

Acceptable operation is to the right of or below the applicable curve. Unacceptable operation is to the left of or above the applicable curve.

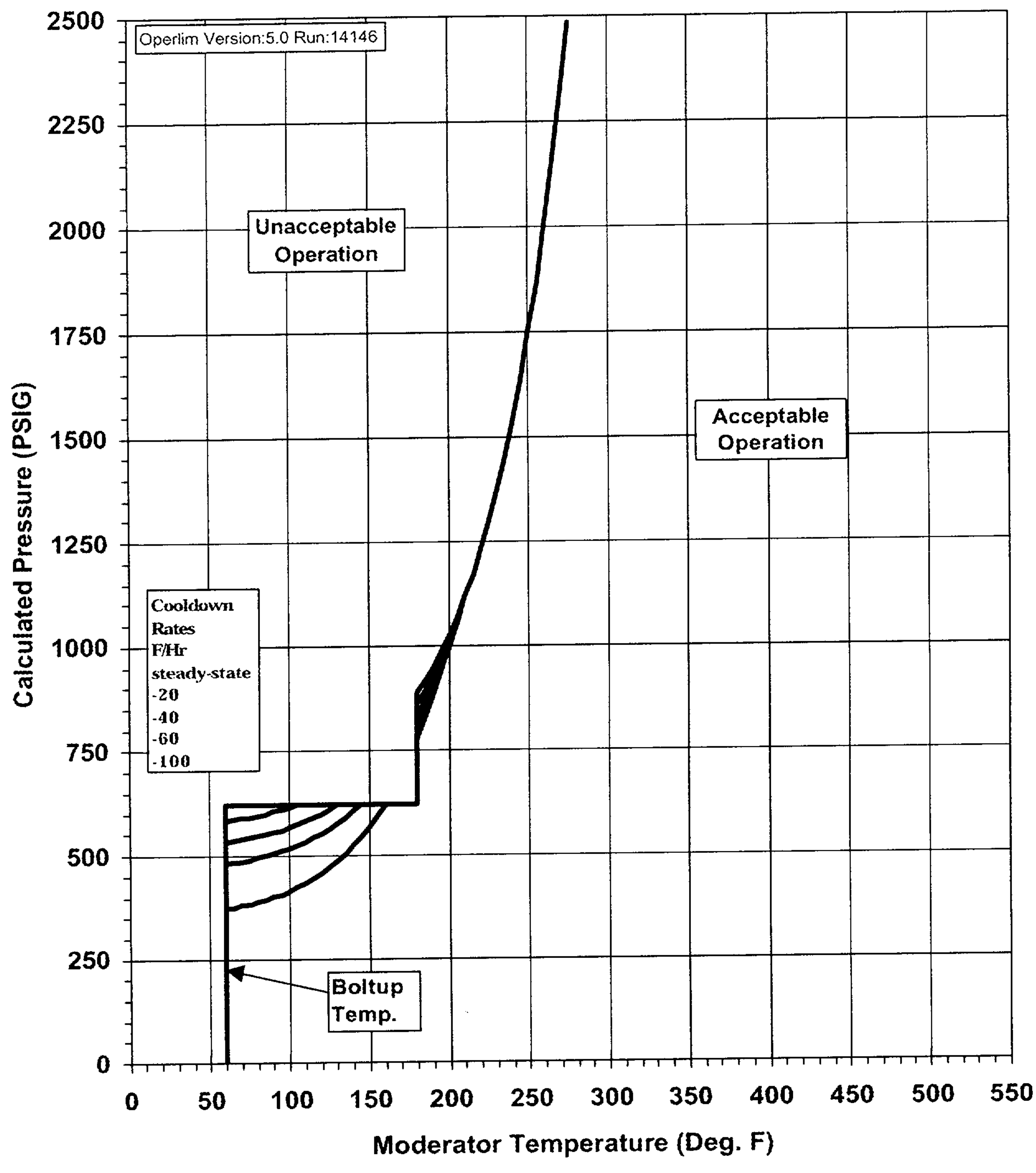

Figure 3.1.B-2 Reactor Coolant System Cooldown Limitations Applicable for the First 25 EFPY (Without Margins for Instrumentation Errors)

Acceptable operation is to the right of or below the applicable curve.
Unacceptable operation is to the left of or above the applicable curve.

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. 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 operable 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 280°F, the requirements of Table 3.1.A-2 regarding the number of charging pumps allowed to be capable of injecting into the RCS shall be adhered to.

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 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 280°F, the requirements of Table 3.1.A-2 regarding the number of safety injection (SI) pumps allowed to be capable of injecting into the RCS 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 recirculation fluid pH control system shall be operable with ≥ 8000 lbs. (148 cu. ft.) of trisodium phosphate (w/12 hydrates), or equivalent, available in storage baskets in the containment.

b. The five fan cooler 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.

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. 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 capable of injecting into the RCS when RCS temperature is less than or equal to 280°F is discussed under Specification 3.1.A.

Table 4.1-1

<u>Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels</u>

Footnotes:

*1 By means of the movable incore detector system.

*2 Prior to each reactor startup if not done previous week.

*3 Monthly visual inspection of condensate weirs only.

*4 Within 31 days prior to entering a condition in which the Control Rod Protection System is required to be operable unless the reactor trip breakers are manually opened during RCS cooldown prior to T_{cold} decreasing below 381°F and the breakers are maintained opened during RCS cooldown when T_{cold} is less than 381°F.

*5 Except when block valve operator is deenergized.

*6 At monthly intervals when OPS is required to be operable. Not required to be performed until 12 hours after entering a condition in which OPS is required to be operable if performed within the prior 24 months.

*7 Acceptable criteria for calibration are provided in Table II.F-13 of NUREG-0737.

*8 Calibration will be performed using calibration span gas.

*9 Each train shall be tested at least every 62 days on a staggered test basis (i.e., one train per month).

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 refueling, repair, replacement or modification.

Specifications

a. The Reactor Coolant System shall be tested for leakage at normal operating pressure prior to plant startup following each refueling outage, in accordance with the requirements of the applicable edition and addenda of the ASME Section XI Code.

b. Testing of repairs, replacements or modifications for the Reactor Coolant System shall meet the requirements of the applicable edition and addenda of the ASME Section XI Code.

c. The Reactor Coolant System leak test temperature-pressure relationship shall be in accordance with the limits of Figure 3.1.B-1 for heatup for the first 25 effective full-power years of operation. Figure 3.1.B-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

Leak test of the Reactor Coolant System is required by the ASME Boiler and Pressure Vessel Code, Section XI, to ensure leak tightness of the system during operation. The test frequency and conditions are specified in the Code.

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 3.1.B-1. The temperatures are calculated in accordance with the methods described in the Basis of Technical Specification 3.1.B.

The minimum inservice leak test temperature requirements for periods up to 25 effective full-power years are shown on Figure 3.1.B-1.

The heatup limits specified on the heatup curve, Figure 3.1.B-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 3.1.B-1 and 3.1.B-2 are recalculated periodically, using methods discussed in the Basis of Technical Specification 3.1.B.

Reference

UFSAR Section 4

4.18 OVERPRESSURE PROTECTION SYSTEM

Applicability

This specification applies to the surveillance requirements for the OPS provided for prevention of RCS overpressurization.

Objective

To verify the operability of OPS.

Specifications

A. When the OPS PORVs are being used for overpressure protection as required by Specification 3.1.A.4, their associated series MOVs shall be verified to be open at least twice weekly with a maximum time between checks of 5 days.

B. When RCS venting is being used for overpressure protection as permitted by Specification 3.1.A.4, the vent(s) shall be verified to be open at least daily. When the venting pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then only these valves need be verified to be open at monthly intervals.

C. When pressurizer pressure and level control is being used for overpressure protection, as permitted by Specification 3.1.A.4, then these parameters shall be verified to be within their limits at least once per shift.

D. When safety injection pumps and/or charging pumps are required to be not capable of injecting into the RCS per Specification 3.1.A.4, the pumps shall be demonstrated to be inoperable at monthly intervals by verifying lockout of the pump circuit breakers at the 480 volt switchgear, or once per shift if other means of making the pumps not capable of injecting into the RCS are used.

E. The PORV backup nitrogen system shall be demonstrated to be operable at Refueling Intervals (#).