ATTACHMENT I TO IPN-98-024

PROPOSED REVISIONS TO TECHNICAL SPECIFICATION CHANGES ASSOCIATED WITH PRESSURE-TEMPERATURE AND OVERPRESSURE PROTECTION SYSTEM LIMITS FOR UP TO 13.3 EFFECTIVE FULL POWER YEARS

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

LIST OF FIGURES

Amendment No. ~~101~~, ~~103~~, ~~116~~, ~~143~~, ~~173~~,

d. When the reactor coolant system T_{avg} is less than 200°F, but not in the refueling operation condition, and as permitted during special plant evolutions, at least one residual heat removal pump (connected to the Reactor Coolant System) shall be in operation. This RHR pump may be out of service for up to 1 hour provided no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and core outlet temperature is maintained at least 10°F below saturation temperature.

e. When the reactor is critical and above 2% rated power, except for natural circulation tests, at least two reactor coolant pumps shall be in operation.

f. The reactor shall not be operated at power levels above 10% rated power with less than four (4) reactor coolant loops in operation.

g. If the requirements of 3.1.A.1.e and 3.1.A.1.f above cannot be satisfied, the reactor shall be brought to the hot shutdown condition within 1 hour.

h. A reactor coolant pump (RCP) may not be started (or jogged) when the RCS cold leg temperature (T_{cold}) is at or below 319°F, unless RCS make up is not in excess of RCS losses, and one of the following requirements is met:

(1) The OPS is <u>operable</u>, steam generator pressure is not decreasing, and the temperature of each steam generator is less than or equal to the coldest T_{cold};

Or

(2) The OPS is <u>operable</u>, the temperature of the hottest steam generator exceeds the coldest T_{cold} by no more than 64°F, pressurizer level is at or below 73 percent, and T_{cold} is as per Figure 3.1.A-1;

Or

(3) With OPS <u>inoperable</u>, steam generator pressure is not decreasing, the temperature of each steam generator is less than or equal to the coldest T_{cold}, pressurizer level is at or below 73 percent, and the RCS pressure does not exceed that given by Fig. 3.1.A-2. The pressurizer level must be further restricted per Figures 3.1.A-5 and 3.1.A-6 if operation below 319°F exceeds 8 hours.

This page was intentionally left blank.

7. REACTOR VESSEL HEAD VENTS

Whenever the reactor coolant system is above 350°F, two reactor vessel head vent paths consisting of two valves in series with power available from emergency buses shall be OPERABLE.

a. If one of the above reactor vessel head vent paths is inoperable, startup and/or power operation may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of all the valves in the inoperable vent path. Restore the inoperable vent path to operable status within 90 days, or be in hot shutdown within 6 hours and be below 350°F within the following 30 hours.

b. With both reactor vessel head vent paths inoperable restore one vent path to operable status within 7 days or be in hot shutdown within 6 hours and be below 350°F within the following 30 hours.

8. OVERPRESSURE PROTECTION SYSTEM (OPS)

a. When the RCS temperature is below 319°F,

(1) the OPS shall be armed and operable. Both OPS PORVs shall have lift settings not to exceed those given in Figure 3.1.A-2, or

(2) the RCS must be vented with an equivalent opening of 2.00 square inches.

b. The requirements of 3.1.A.8.a may be modified to allow one PORV and/or its series block valve to be inoperable for a maximum of seven (7) consecutive days.

c. If the requirements of 3.1.A.8.a or 3.1.A.8.b cannot be met, then one of the following actions shall be completed within 8 hours.

(1) The RCS must be depressurized and vented with an equivalent opening of at least 2.00 square inches;

Or

(2) The RCS must be heated in accordance with the restrictions of Specification 3.1.A.1.h(3) and maintained above 411°F;

Or

(3) Restrict pressurizer level as per the curves on Figures 3.1.A-5 and 3.1.A-6.

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

The requirement that 150 kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at hot shutdown.

The power operated relief valves (PORVs) 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 ensure the ability to seal off possible RCS leakage paths.

Reactor vessel head vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one reactor vessel head vent path ensures that capability exists to perform this function.

The valve redundancy of the reactor coolant system vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve power supply or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the reactor coolant system vent systems are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November, 1980.

The OPS is designed to relieve the RCS pressure for certain unlikely incidents to prevent the peak RCS pressure from exceeding the limits established in Reg. Guide 1.99, Revision 2. The OPS is considered to be operable when the minimum number of required channels (per Table 3.5-3) are available to open the PORVs upon receipt of a high pressure signal which is based upon RCS T_{cold}, as shown in Figure 3.1.A-2. (The happy face icon contained on this and other Technical Specification figures indicates the side of the applicable curve in which operation is permissible. Conversely, the sad face icon indicates the side of the applicable curve in which operation is prohibited.) The OPS setpoint is based upon a comparative analysis of References 5 and 9, with allowances for metal/fluid temperature differences (as described below) and for the static head due to elevation differences and dynamic head effect of the operation of the reactor coolant and RHR pumps. "Arming" means that the motor operated block valve (MOV) is in the open position. This can be accomplished either automatically by the OPS when the RCS temperature is less than or equal to 319°F or manually by the control room operator.

The start of an RCP is allowed when the steam generators' temperature does not exceed the RCS and the OPS is operable. During all modes of operation, the steam generator temperature may be measured using the Control Room instrumentation or, as a backup, from a contact reading off the steam generator's shells.

Most start-ups will satisfy these requirements as provided in Specification 3.1.A.1.h (1). In order to allow start of an RCP when the steam generators are hotter than the RCS, requirements for a pressurizer bubble (gas or steam) are developed (technical specification value for pressurizer level includes an allowance for instrument uncertainty). During this Heat Input initiation event the RCS fluid temperature rise is considerably more rapid than the reactor vessel metal temperature rise. Since OPS utilizes a setpoint curve (Figure 3.1.A-2) and the temperature measured is the fluid temperature, and not the reactor vessel metal, it is necessary to shift to the right the OPS setpoint curve by 50°F to ensure the pressure does not exceed the allowable values for the vessel. For the conditions when the OPS is inoperable, additional requirements are developed for the pressurizer bubble, RCS pressure and temperature.

Due to the rate of energy transferred to the RCS, when the RCP is started, the resultant rate of temperature rise and the pressure increase are strongly dependent on the temperature difference between the RCS and the steam generators. The presence of a pressurizer bubble provides for a more moderate pressure increase. The bubble size is sufficient to prevent the RCS from going water solid for 10 minutes during which time operator action will terminate the pressure transient. Pressurizer level refers to indicated level and includes instrument uncertainty. The preventive measures for a Mass Input initiating event (i.e., up to three charging pumps and/or one SI pump) as well as the Heat Input initiating event are described in References (3), (4) and (5). (Also refer to Specifications 3.3.A.8, 3.3.A.9, and 3.3.A.10.) The OPS need not be operable when the RCS temperature is less than 319°F if the RCS is depressurized and vented with an equivalent opening of at least 2.00 square inches. One PORV, blocked fully open, also satisfies this vent area requirement. This opening is adequate to relieve the worst case analyzed. It should be noted that the analysis of record (Reference 5) is based upon a minimum vent area of 1.4 square inches, which for the sake of conservatism has been rounded up to 2.00 square inches.

The OPS enable temperature of 319°F permits the performance of an RCS hydrostatic test (see Fig. 4.3-1) without activating the OPS.

Upon OPS inoperability, the RCS may be heated above 411°F. This temperature is that value for which the RCS heatup and cooldown curves (Figures 3.1-1 and 3.1-2) permit pressurization to the setting of the pressurizer safety valves. Accordingly, with an inoperable OPS and an RCS temperature above 411°F, the pressurizer safety valves will preclude violation of the 10 CFR 50, Appendix G, curves. In addition, the OPS need not be operable upon satisfying the conditions of Specification 3.1.A.8.c(3) which requires the presence of a pressurizer bubble to preclude RCS overpressurization during inadvertent mass inputs. Specification 3.1.A.8.c(3) also places restrictions on the number of charging and SI pumps capable of feeding the RCS (see Specifications 3.3.A.8, 3.3.A.9, and 3.3.A.10). Any pump can be

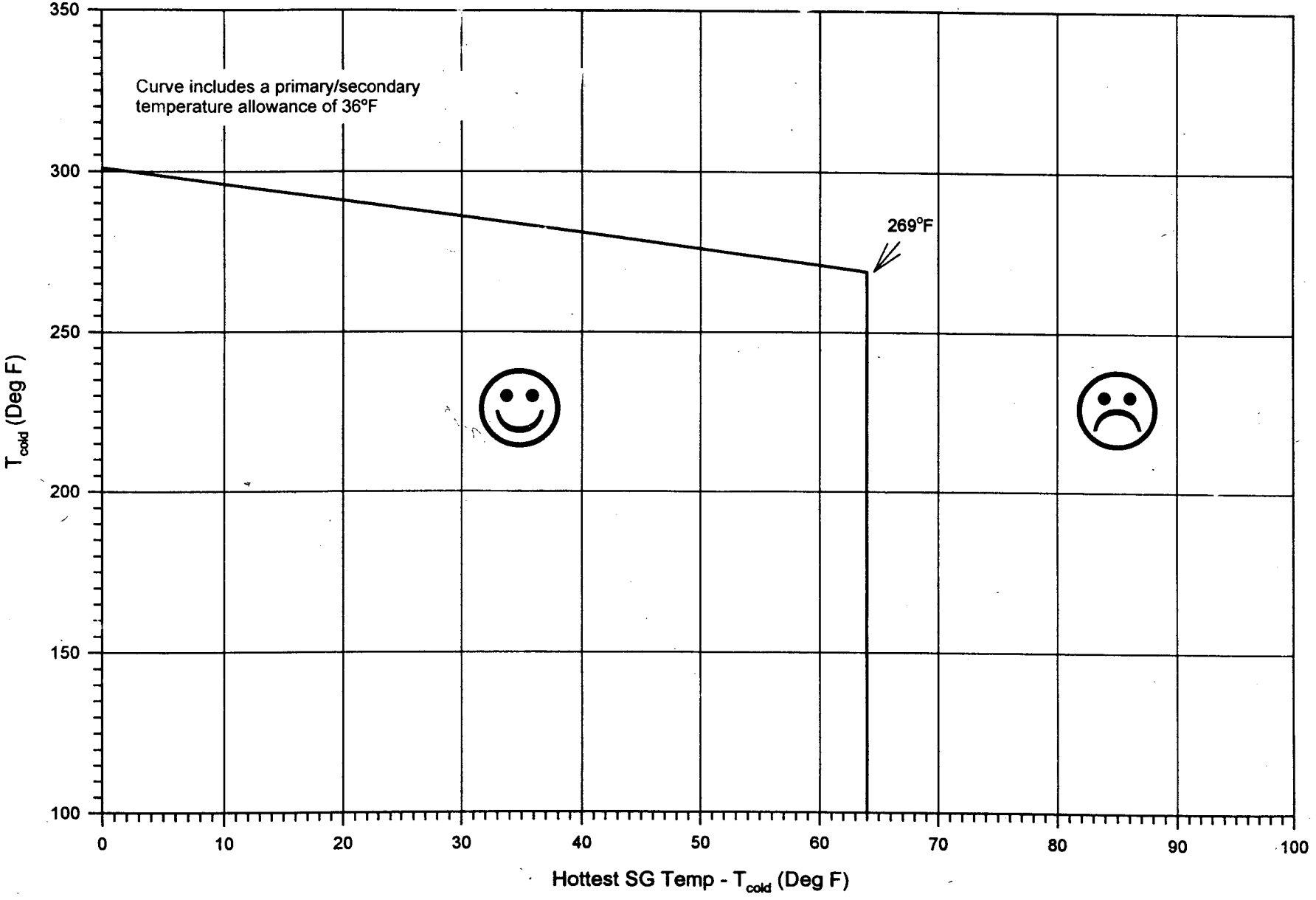
rendered incapable of feeding the RCS if, for example, its switch is in the trip pull-out position, or if at least one valve in the flow path to the RCS is closed and locked (if manual) or de-energized (if motor operated). This section has also been revised in accordance with the results of tests conducted on the capsule T, Y, and Z specimens (References 6, 7 and 8).

References

1) FSAR Section 14.1.6
2) FSAR Section 14.1.8
3) Letter dated 10/25/78 "Summary of Changes to IP-3 Plant Operating Procedures in Order to Preclude RCS Overpressurization"
4) Letter dated 2/28/76 "Conceptual Design of the Reactor Coolant Overpressure Protection System" and response to NRC questions.
5) IP-3 Low Temperature Overpressurization Protection System Analysis, NYPA Report dated 8/24/84.
6) WCAP-9491 "Analysis of Capsule T from IP-3 Reactor Vessel Radiation Surveillance Program", J.A. Davidson, S.L. Anderson, W.T. Kaiser, April 1979.
7) WCAP-10300-1, "Analysis of Capsule Y from the Power Authority of the State of New York Indian Point Unit 3 Reactor Vessel Radiation Surveillance Program," S.E. Yanichko, S.L. Anderson, March 1983.
8) WCAP-11815, "Analysis of Capsule Z from the New York Power Authority Indian Point Unit 3 Reactor Vessel Radiation Surveillance Program," S.E. Yanichko, S.L. Anderson, L. Albertin, March 1988.
9) ASME Code Case N-514, "Low Temperature Overpressure Protection," February 12, 1992.

FIGURE 3.1.A-1

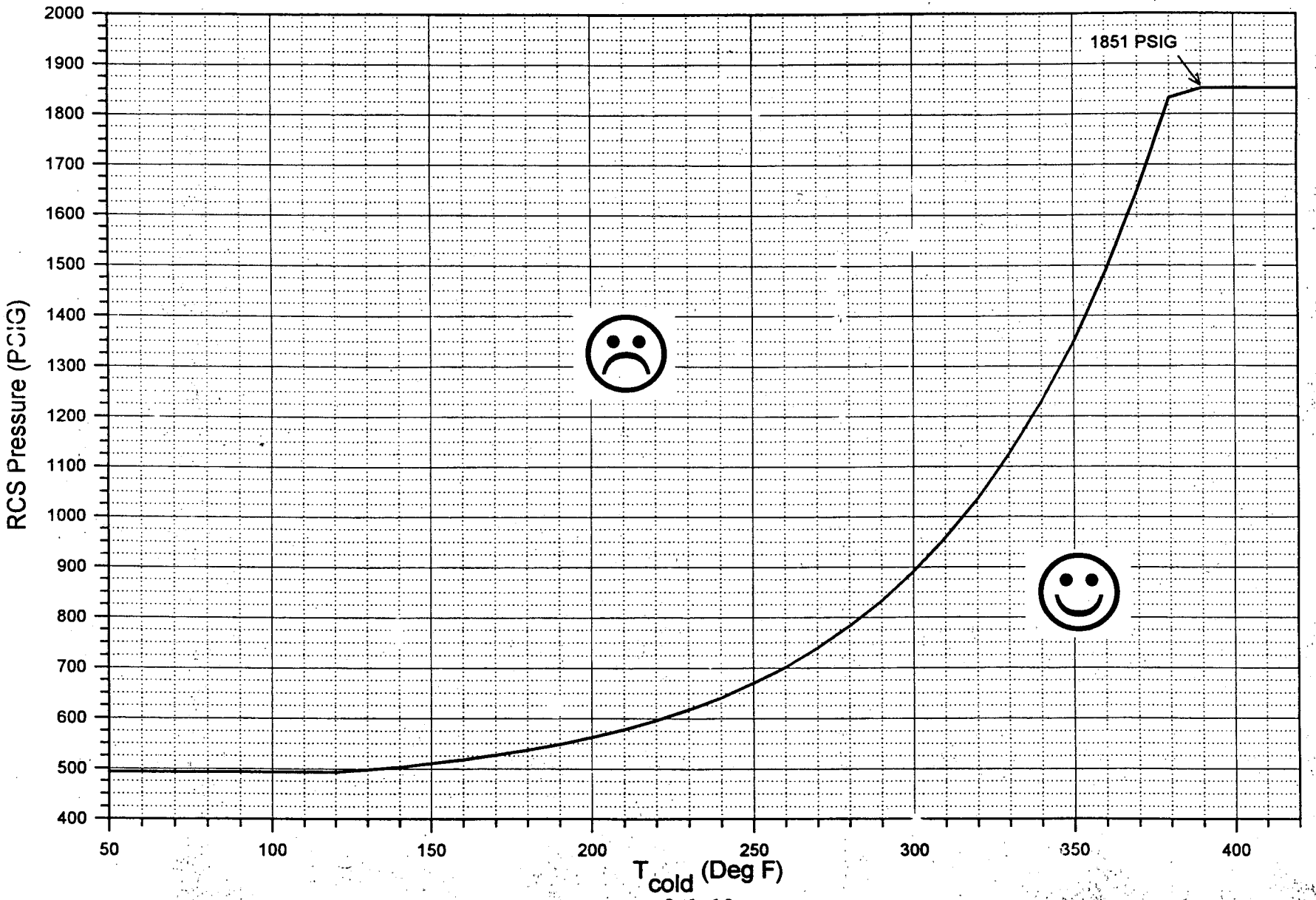
Secondary Side Limitations for RCP Start With
Secondary Side Hotter Than Primary, 13.3 EFPY



Amendment No. 67, 121,

Figu[illegible]3.1.A-2

Maximum Allowable Nominal PORV Setpoint for the
Low Temperature Overpressure Protection System (OPS), 13.3 EFPY



Analytical Curve

Amendment No. 67, 121,

FIGURE 3.1.A-3

DELETED

FIGURE 3.1.A-4

DELETED

Pressurizer Limitations for OPS Inoperable
(Up to one charging pump capable of feeding RCS), 13.3 EFPY

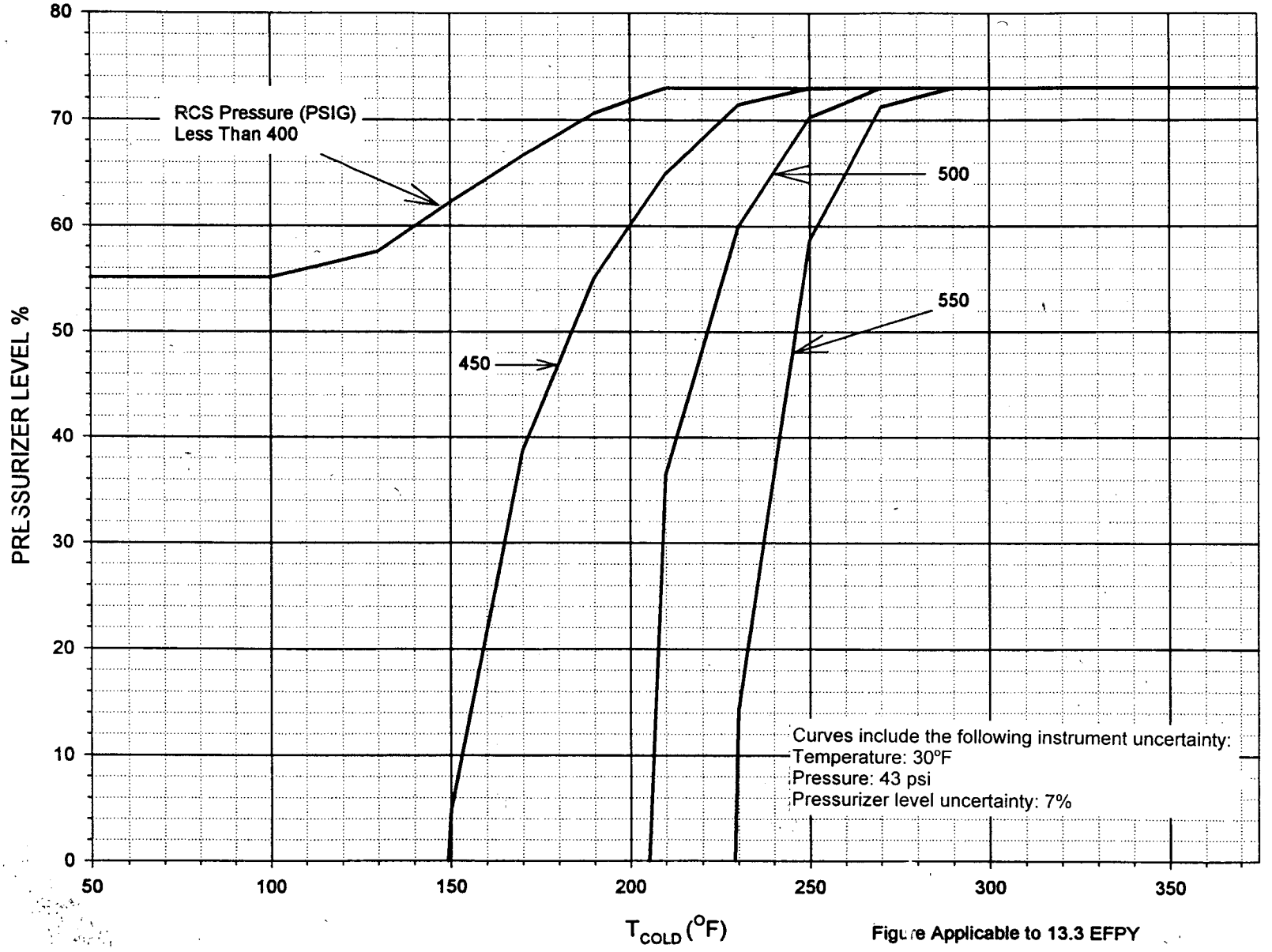


Figure Applicable to 13.3 EFPY

Curves represent maximum allowable pressurizer level for the conditions defined.

FIGURE 3.1.A-6

Pressurizer Limitations for OPS Inoperable (up to three charging pumps and/or one safety injection pump capable of feeding RCS), 13.3 EFPY

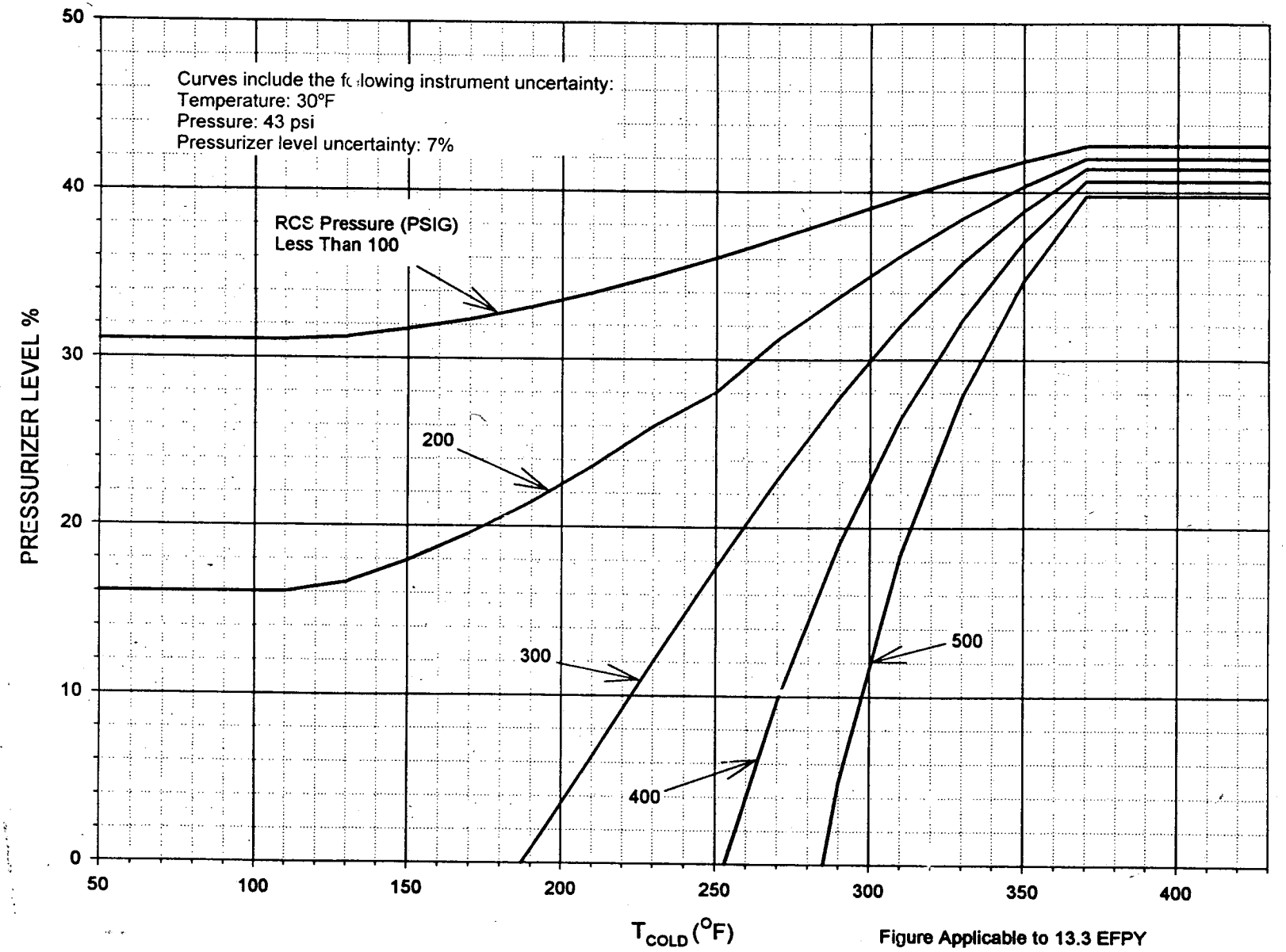


Figure Applicable to 13.3 EFPY

Curves represent maximum allowable pressurizer level for the conditions defined.

Amendment No. 67, 101, 121,

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 3.1-1 and Figure 3.1-2 for the service period up to 13.3 effective full-power years (EFPYs). The heatup and cooldown rates shall not exceed 60°F/hr and 100°F/hr respectively.

   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.

2. The limit lines shown in Figure 3.1-1 and Figure 3.1-2 shall be recalculated periodically using methods discussed in the Basis and results of surveillance specimens as covered in Specification 4.2. The order of specimen removal may be modified based on the results of testing of previously removed specimens.

3. 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.

4. 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.

5. Reactor Coolant System integrity tests shall be performed in accordance with Section 4.3.

Basis

Fracture Toughness Properties

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the Summer 1965 Section III of the ASME Boiler and Pressure Vessel Code [6] and ASTM E185 [5] and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1972 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code [1], and the calculation methods described in WCAP-7924 [2].

The first reactor vessel material surveillance capsule was removed during the 1978 refueling outage. This capsule has been tested by Westinghouse Corporation and the results have been evaluated and reported [7]. Similar reports were prepared for the surveillance capsules [10, 8] removed in 1982 and 1987. Based on the Westinghouse evaluation, heatup and cooldown curves (Figures 3.1-1 and 3.1-2) were developed for up to 13.3 EFPYs of reactor operation.

Generic Letter 88-11 requested that licensees use the methodology of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials", to predict the effect of neutron radiation on reactor vessel materials as required by paragraph V.A. of 10 CFR part 50, Appendix G. Capsule Z was analyzed [8] and new pressure-temperature curves were developed using this methodology.

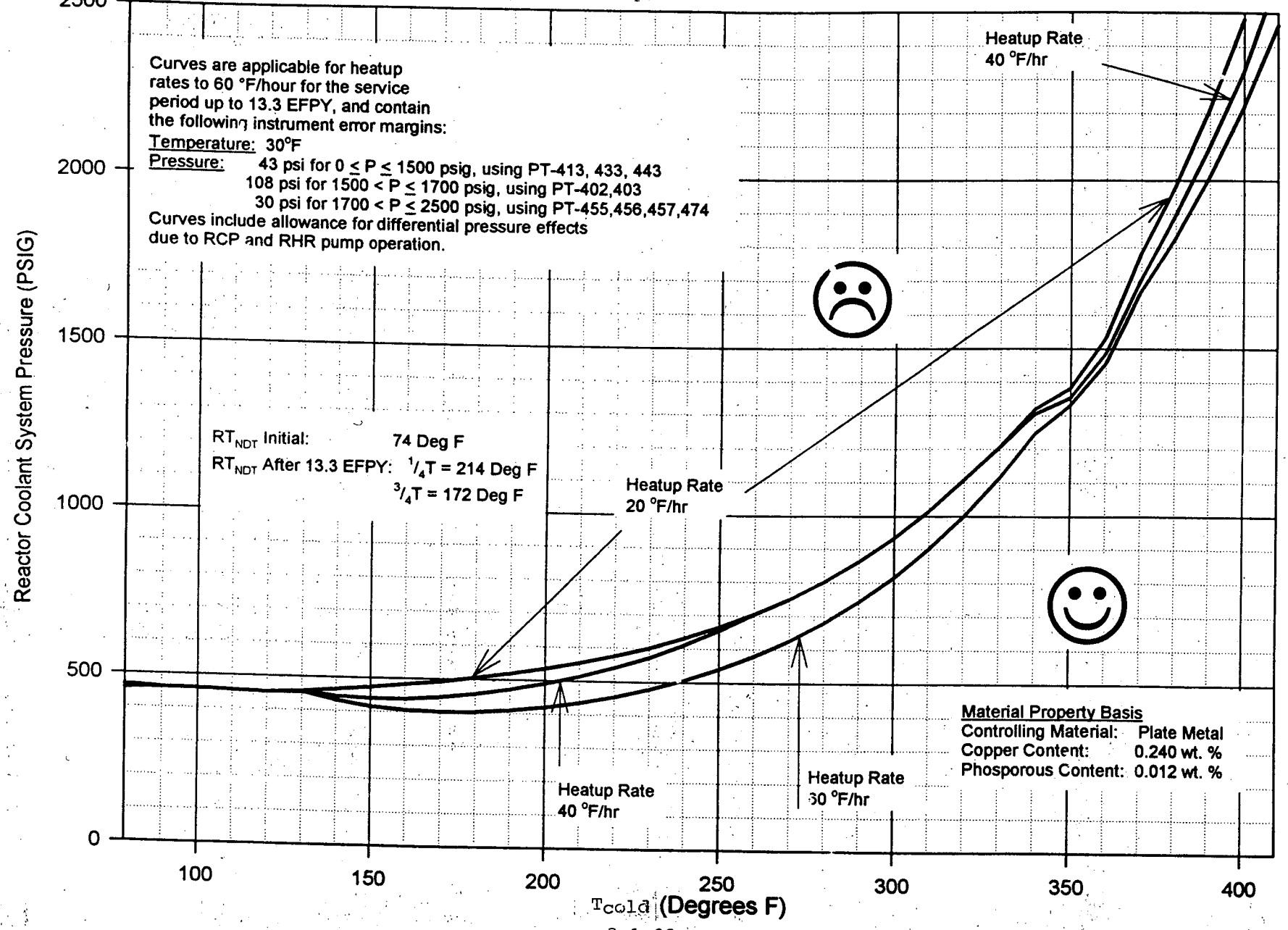
The maximum value in RT_{NDT} after 13.3 EFPYs of operation is projected to be 214°F at the 1/4 T and 172°F at the 3/4 T vessel wall locations for Plate B2803-3 the controlling plate. Plate B2803-3 was also the controlling plate for the operating period up to 11 EFPYs.

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} at the end of 13.3 years of service life. The 13.3 year service life period is chosen such that the limiting RT_{NDT} at the 1/4 T location in the core region is higher than the RT_{NDT} of the limiting unirradiated material. This service period assures that all components in the Reactor Coolant System will be operated conservatively in accordance with Code recommendations.

The highest RT_{NDT} of the core region material is determined by adding the radiation induced ΔRT_{NDT} for the applicable time period to the original RT_{NDT} shown in Table Q4.2-1 [3].

FIGURE 3.1-1

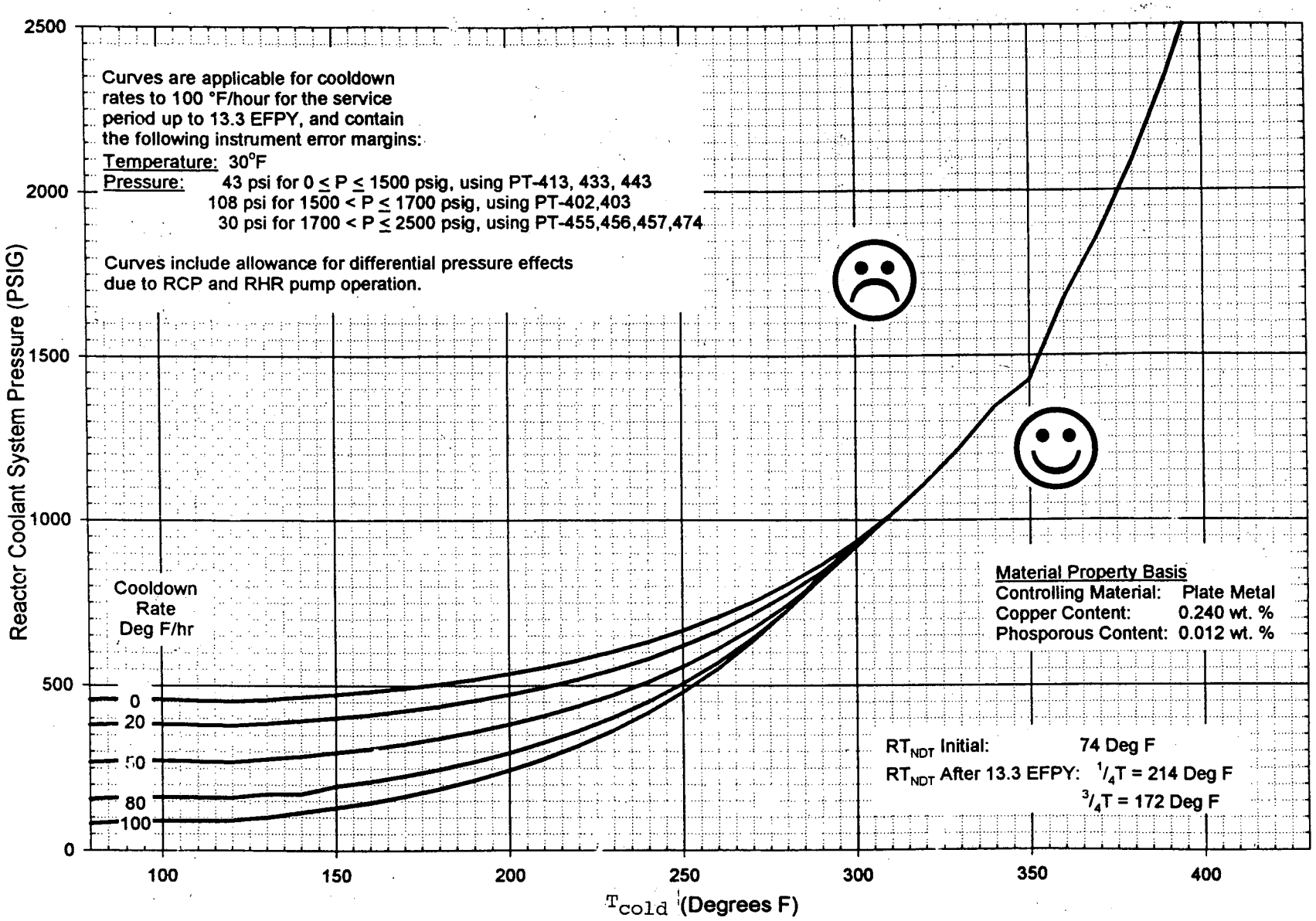
Reactor Coolant System Heatup Limitations - Indian Point 3



Amendment No. 28, 100, 121,

FIGURE 3.1-2

Reactor Coolant System Cooldown Limitations - Indian Point 3



Amendment No. 28, 109, 121,

d. One pressure and one level transmitter shall be operating per accumulator.

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

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

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

h. Valves 856B and 856G in the Safety Injection discharge headers shall be closed and their power supplies de-energized.

i. Valve 1810 in the suction line of the high-level SI pumps and valves 882 and 744 in the suction and discharge lines, respectively, of the residual heat removal pumps shall be open and their power supplies de-energized.

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

k. The refueling water storage tank low level alarms are operable and set to alarm between 10.5 feet and 12.5 feet of water in the tank.

l. Valve 883 in the RHR return line to the RWST is de-energized in the closed position.

m. Valves 1870 and 743 in the miniflow line for the Residual Heat Removal Pumps shall be open and their power supplies de-energized.

n. The RHR system is in the ESF alignment with the normal RHR suction line isolated from the RCS.

4. The requirements of 3.3.A.3 may be modified to allow any one of the following components to be inoperable at any one time:

2) RCS temperature and the source range detectors are monitored hourly;

and

3) no operations are permitted which would reduce the boron concentration of the reactor coolant system.

8. When the RCS average cold leg temperature (T_{cold}) is below 319°F, or when RHR is in service (i.e., not isolated from the RCS), no safety injection pumps shall be energized and aligned to feed the RCS.

9. The requirements of 3.3.A.8 may be relaxed to allow one safety injection pump energized and aligned to feed the RCS under the following circumstances:

   a. emergency boration; OR

   b. for pump testing, for a period not to exceed 8 hours; OR

   c. loss of RHR cooling.

10. The requirements of 3.3.A.8 may be further relaxed when the RCS is < 200°F, such that two safety injection pumps may be energized and aligned to feed the RCS under the following circumstances:

   a. the RCS is vented with an opening greater than or equal to the size of one code pressurizer safety valve flange, OR

   b. indicated pressurizer level is at 0% and the plant is vented in accordance with Technical Specification 3.1.A.8.c.1. (Alternate methods and instrumentation may be used to confirm actual RCS elevation.)

B. <u>Containment Cooling and Iodine Removal Systems</u>

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

   a. The spray additive tank contains a minimum of 4000 gallons of solution with a sodium hydroxide concentration ≥35% and ≤38% by weight.

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

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

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 their performance is to sufficiently limit any increase in clad temperature below a value where emergency core cooling objectives are met.[13]

During operating modes in the temperature range between 200°F and 350°F, a sufficient decay heat removal capability is provided by a reactor coolant pump with a steam generator heat sink or a residual heat removal loop. This redundancy ensures that a single failure will not result in a complete loss of decay heat removal. Above 350°F, the normal RHR suction line is isolated from the RCS to protect RHR piping from overpressurization due to inadvertent SI pump actuation.

During operating modes when the reactor coolant T_{avg} is less than 200°F, but not in the refueling operation condition, a sufficient decay heat removal capability is provided by a residual heat removal loop.

The containment cooling and iodine removal functions are provided by two independent systems: (a) fan-coolers plus charcoal filters and (b) 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).[4] In the event of a Design Basis Accident, any one of the following configurations will provide sufficient cooling to reduce containment pressure at a rate consistent with limiting off-site 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, any one of three configurations of fan-cooler units (with charcoal filters) and/or containment spray pumps (with sodium hydroxide addition) will reduce airborne organic and molecular iodine activities sufficiently to limit off-site doses to acceptable values.[5] Any one of these three configurations constitutes the minimum safeguards for iodine removal.

The combination of three fan-coolers and one containment spray pump is capable of being operated on emergency power with one diesel generator failing to start. Adequate power for operation of the redundant containment heat removal systems (i.e., five fan-cooler units or two containment spray pumps) is assured by the availability of off-site power or operation of all emergency diesel generators.

These toxic gas monitoring systems are designed to alarm in the control room upon detection of the short term exposure limit (STEL) value. The operability of the toxic gas monitoring systems provides assurance that the control room operators will have adequate time to take protective action in the event of an accidental toxic gas release. Selection of the gases to be monitored are based on the results described in the Indian Point Unit 3 Habitability Study for the Control Room, dated July, 1981. The alarm setpoints will be in accordance with industrial ventilation standards as defined by the American Conference of Governmental Industrial Hygienists.[16]

The RHR suction line is required to be isolated from the RCS when temperature is above 350°F. This protects the RHR system from overpressurization when the SI system is required to be in service. The requirement to prevent safety injection pumps from being able to feed the RCS under specific conditions prevents overpressurization of the RHR system or the RCS beyond the capacity of the OPS to mitigate. These conditions include when OPS is required to be in service and when RHR is in service. Special allowances are made for pump testing, loss of RHR cooling (during which time an SI pump may be required to recirculate coolant to the core), or emergency boration. Two SI pumps may be energized and aligned to feed the RCS when situations prevail that could not result in overpressurization. This is satisfied when the RCS is vented with an opening greater than or equal to the size of one code pressurizer safety flange or when the pressurizer level is low enough (indicating 0%) and the plant is vented in accordance with Technical Specification 3.1.A.8.c.1 to ensure at least a ten minute operator response time on inadvertent SI actuation without the pressurizer completely filling. Alternate methods and instrumentation may be used to confirm actual RCS elevation. Methods to ensure that an SI pump is unable to feed the RCS include placing the SI pump switches in the trip pull-out position, or by closing and locking (if manual) or de-energizing (if motor operated) at least one valve in the flow path from these pumps to the RCS.

References

1) FSAR Section 9
2) FSAR Section 6.2
3) FSAR Section 6.2
4) FSAR Section 6.3
5) FSAR Section 14.3.5
6) FSAR Section 1.2
7) FSAR Section 8.2
8) FSAR Section 9.6.1
9) FSAR Section 14.3
10) FSAR Section 6.8
11) FSAR Section 6.5
12) Response to Question 14.6, FSAR Volume 7
13) FSAR Appendix 14C
14) Response to Question 9.35, FSAR Volume 7
15) WCAP-12313, "Safety Evaluation for an Ultimate Heat Sink Temperature Increased to 95° at IP-3"
16) American Conference of Governmental Industrial Hygienists 1982 Industrial Ventilation, 19th Edition
17) NYPA calculation IP3-CALC-SI-00725, Rev. 0, "Instrument Loop Accuracy/Setpoint Calc./RWST Level."
18) Nuclear Safety Evaluation 93-3-162-SI, Rev. 0, Adequate Post-LOCA Coolant Inventory.

4.3 REACTOR COOLANT SYSTEM (RCS) TESTING

A. 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.

Specification

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 ASME Section XI.

b) Testing of repairs, replacements or modifications for the Reactor Coolant System shall meet the requirements of ASME Section XI.

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 13.3 EFPYs of operations. Figure 4.3-1 will be recalculated periodically. Allowable pressures during cooldown from the leak test temperature shall be in accordance with Figure 3.1-2.

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 test temperatures are shown on Figure 4.3-1. The temperatures are calculated in accordance with ASME Code Section III, 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.

Amendment No. 28, 101, 100, 121, 170, 171,

For the first 13.3 effective full power years, it is predicted that the highest RT_{NDT} in the core region taken at the 1/4 thickness will be 214°F. The temperature determined by methods of ASME Code Section III for 1989 psig is 134°F above this RT_{NDT} and for 2485 psig (maximum) is 153°F above this RT_{NDT}. The minimum inservice leak test temperature requirements for periods up to 13.3 effective full power years are shown on Figure 4.3-1[2].

The heatup limits specified on the heatup curve, Figure 4.3-1, must not be exceeded while the reactor coolant system is being heated to the inservice leak test temperature. For cooldown from the leak test temperature, the limitations of Figure 3.1-2 must not be exceeded. Figures 4.3-1 and 3.1-2 are recalculated periodically, using methods discussed in the Basis for Specification 3.1.B and results of surveillance specimens, as covered in Specification 4.2.

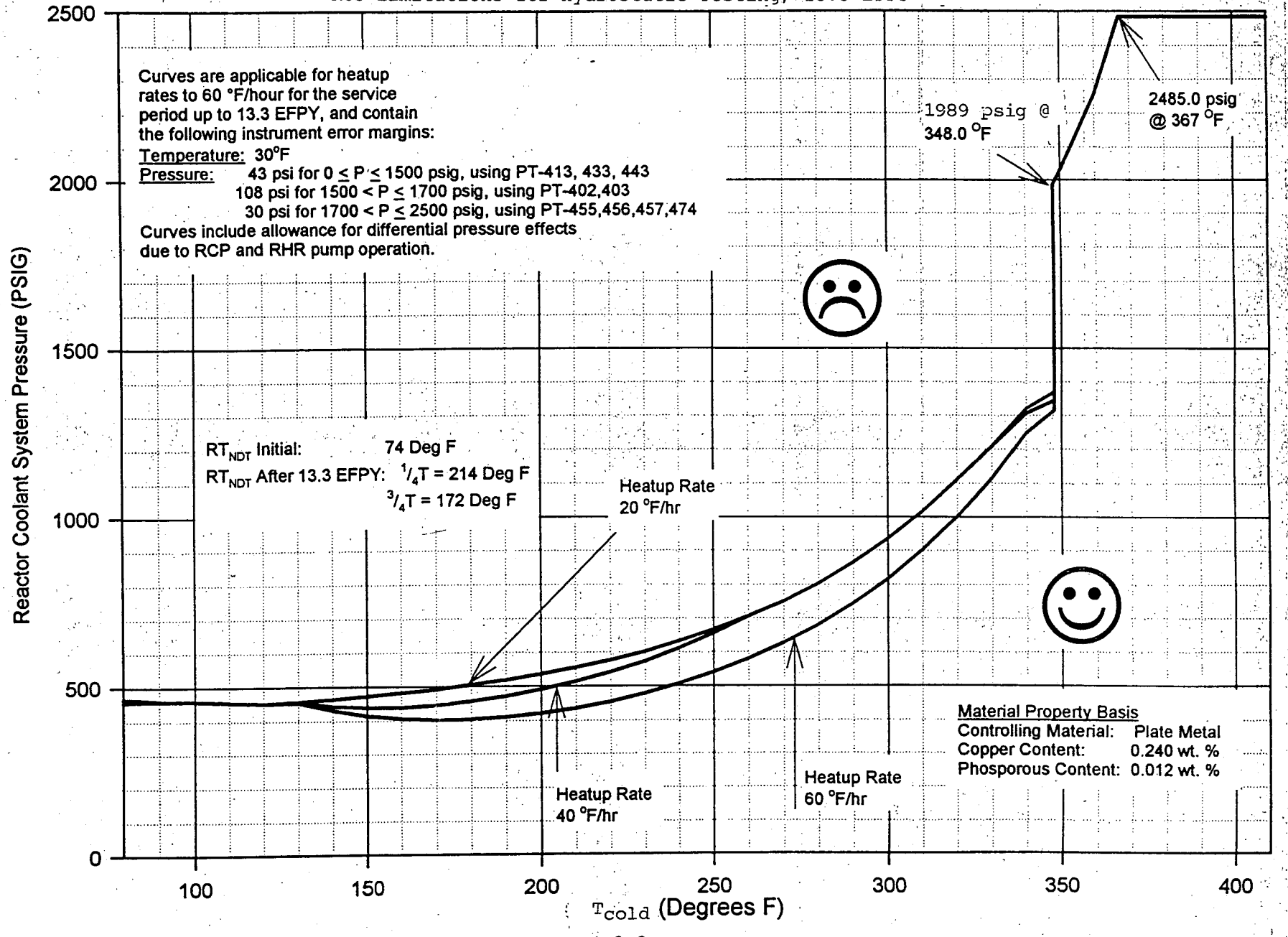
Reference

1. FSAR, Section 4.

2. "Indian Point Unit 3 Final Report on Appendix G Reactor Vessel Pressure-Temperature Limits" ABB-Combustion Engineering, July 24, 1990

FIGURE 4.3-1

RCS Limitations for Hydrostatic Testing, 13.3 EFPY



Amendment No. 28, 109, 121,

ATTACHMENT II TO IPN-98-024

SAFETY EVALUATION OF REVISED PROPOSED TECHNICAL SPECIFICATION CHANGES ASSOCIATED WITH PRESSURE-TEMPERATURE AND OVERPRESSURE PROTECTION SYSTEM LIMITS FOR UP TO 13.3 EFFECTIVE FULL POWER YEARS

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

SAFETY EVALUATION RELATED TO REVISED PROPOSED TECHNICAL SPECIFICATION CHANGES ASSOCIATED WITH PRESSURE-TEMPERATURE AND OVERPRESSURE PROTECTION SYSTEM LIMITS FOR UP TO 13.3 EFFECTIVE FULL POWER YEARS

Section I - Description of Changes

This revised amendment request seeks to modify Sections 3.1, 3.3, and 4.3 of Appendix A of the Indian Point 3 Technical Specifications. These revisions extend the Heatup-Cooldown limits from 11 to 13.3 effective full power years (EFPYs), provide the corresponding Overpressure Protection System (OPS) limits and include some minor revisions which ensure specification clarity and conservatism.

Section II - Evaluation of Changes

Pressure-Temperature Limits

The pressure-temperature limit curves define an acceptable region for normal plant operation. They limit the pressure and temperature changes during RCS heatup and cooldown to within the design assumptions and the stress limits for cyclic operation. The pressure-temperature limits are periodically modified as the reactor vessel material toughness decreases due to neutron embrittlement caused by neutron irradiation. Generic Letter 88-11 (Reference 1) requested licensees to use the methods described in Revision 2 to Regulatory Guide (RG) 1.99 to predict the effect of neutron radiation on reactor vessel material. A report containing a series of heatup and cooldown curves for several representative points in the reactor life was prepared in accordance with RG 1.99, Revision 2 for Indian Point 3. This report was previously submitted to the NRC by Reference 2. Additional information pertaining to the methodology and calculations used to generate the pressure temperature curves was submitted to the NRC by References 3 and 4. The new heatup and cooldown pressure-temperature curves contained in Attachment I are effective up to 13.3 EFPYs.

OPS Limits

The low temperature overpressurization protection (LTOP) system controls reactor coolant system (RCS) pressure at low temperatures so the integrity of the reactor coolant pressure boundary is not compromised by violating the pressure and temperature limits of 10 CFR 50, Appendix G. The LTOP system limits contained in this submittal are based upon the information submitted to the NRC by References 2 and 5 and the use of ASME Code Case N-514. (Reference 9 contains an exemption request to use this Code Case in lieu of the requirements of 10 CFR 50.60.) ASME Code Case N-514 limits the OPS curve to no greater than 110% of the pressure determined to satisfy Appendix G, paragraph G-2215 of ASME Code Section XI, Division 1, further reduced to allow for static head due to elevation differences

and dynamic head effect due to the operation of the reactor coolant and RHR pumps. The OPS enable temperature is based on the coolant temperature corresponding to a reactor vessel metal temperature derived from the Code Case formula of RT_{NDT} + 50°F for the limiting beltline position (1/4 of the vessel section thickness from the inside surface). The use of Code Case N-514 to determine LTOP parameters enables Indian Point 3 to maintain sufficient operating margin to reduce the potential of unnecessary LTOP system actuation and to ensure that the reactor vessel is protected during low pressure transients. Reference 5 submitted a NYPA calculation which further describes the generation of the OPS curves and setpoints.

For conditions in which the OPS is inoperable, a pressurizer bubble is established to serve as a pressure cushion which mitigates the effects of the postulated heat input or mass input events. The size of this bubble is determined using methodology previously provided to the NRC in Reference 2 and is revised only to include the effects of increased vessel lifetime burnup of 13.3 EFPYs.

Additional Changes

Clarification of PORV Opening Requirements

Specification 3.1.A.8 and its associated basis has been revised to clarify the requirements associated with the operation of the overpressure protection system. Specifically, sections 3.1.A.8.a, b, and c have been rewritten to clearly state that if the RCS temperature is below the OPS enable temperature, either the OPS shall be armed and operable or the RCS shall be vented with an equivalent opening of 2.0 square inches. Section 3.1.A.8.c then specifies the actions required if the requirements of 3.1.A.8.a and b cannot be met. The eight hour completion time associated with Specification 3.1.A.8.c is consistent with the Westinghouse Standard Technical Specifications (STS) (Reference 6). The STS basis states that the completion time considers the time required to place the plant in this condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

The basis for Section 3.1.A.8 has also been revised to clarify that one PORV, blocked fully open, satisfies the vent area requirement of 2.0 square inches. (The minimum vent area associated with one fully open PORV is 2.25 square inches.) This requirement is conservative in comparison to the analysis of record (Reference 7) which is based upon a minimum vent area of 1.4 square inches.

RHR Overpressurization

The proposed technical specification changes in Sections 3.1.A.8, 3.3.A.3.n, 3.3.A.8, 3.3.A.9, and 3.3.A.10 limit the operation of the safety injection (SI) pumps. Specifically, when the RCS average cold leg temperature is below the OPS enable temperature or the residual heat removal (RHR) system is in service, operation of the SI pumps is precluded. These requirements do not apply during pump testing, loss of RHR cooling, or during emergency boration. These changes serve to protect the RHR system from an overpressurization event. The relief capacity of the RHR system (Reference 8) is not sufficient to mitigate the results of

an overpressure event if one safety injection pump is actuated when RHR is aligned to cool the core, as currently allowed by Technical Specification 3.3.A.8. Current plant procedures require that all three SI pumps be placed in the trip pullout position whenever RHR is in service. For cases involving pump testing, reduced inventory operation, or response to loss of RHR cooling, procedures require that one SI pump be returned to service. These procedures are therefore more restrictive than the current Technical Specifications. These proposed changes will make the Technical Specifications as conservative as the plant procedures by placing conservative restrictions on SI actuation when RHR is in service, thus precluding overpressurization of the RHR system.

As stated in Section 3.3.A.10, two SI pumps may be energized and aligned to feed the RCS when situations prevail that cannot result in overpressurization. This occurs when the RCS is vented with an opening greater than or equal to the size of one code pressurizer safety valve flange or when the pressurizer level indicates 0% and the plant is vented as per Technical Specification 3.1.A.8.c.1. A pressurizer level of 0% ensures at least a ten minute operator response time on inadvertent SI actuation without the pressurizer completely filling.

Changes to Section 3.1.A.1

Section 3.1.A.1.h, I, and j have been rewritten to clarify the requirements associated with reactor coolant pump (RCP) starts. Specifically, Specification 3.1.A.1.j has been removed and incorporated into 3.1.A.1.h. Specification 3.1.A.1.h.3 has been revised to add the requirement that the pressurizer level must be restricted as per Technical Specification Figures 3.1.A-5 and 3.1.A-6 if operation below the OPS enable temperature exceeds eight hours. This requirement is consistent with Specification 3.1.A.8.c.

In addition, Specification 3.1.A.1.h.4 has been deleted. This specification allowed an RCP to be started with the OPS inoperable and the temperature of the hottest steam generator greater than the coldest T_{cold}. Current plant procedures do not allow operators to start an RCP under these conditions, but require more conservative actions, such as returning the OPS to service. Therefore, deletion of the section makes the technical specifications more conservative as it prohibits the start of an RCP under these plant conditions. Similarly, Technical Specification 3.1.A.1.i has been eliminated. This specification allowed operation at a pressure greater than the nominal PORV setpoint when OPS was inoperable and a pressurizer bubble was established. Elimination of this specification results in more conservative operation as it requires operation below the PORV setpoint at all times.

Finally, the maximum pressurizer level has been revised to 73% (in lieu of 75%). The pressurizer level revision is the result of the switch over to 24 month operating cycles. The current specifications (75%) are based on 18 month cycles with an uncertainty of ±5%, while the proposed specification (73%) is based on 24 month cycles and an uncertainty of ±7%. As stated in Reference 5, the 18 month surveillance interval for pressurizer level is not being revised by this amendment request. Therefore the increased uncertainty adds additional conservatism to the Technical Specification requirements.

Section III - No Significant Hazards Evaluation

Consistent with the criteria of 10 CFR 50.92, the enclosed application is judged to involve no significant hazards based on the following information.

(1) Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously analyzed?

Response:

The proposed license amendment does not involve a significant increase in the probability or consequences of a previously analyzed accident. The pressure-temperature limit changes proposed by this amendment are based on supporting data and evaluation methodologies previously submitted to the NRC in References 2, 3 and 4. These limits are based upon the irradiation damage prediction methods of Regulatory Guide 1.99, Revision 2. The LTOPS changes contained in this submittal have been conservatively adjusted in accordance with the new pressure-temperature limits, in accordance with the information contained in References 2 and 5 and ASME Code Case N-514.

The revised version of Section 3.1.A.8 clarifies existing requirements related to the OPS system and adds an eight hour completion time for compensating actions, consistent with the STS. The changes to Section 3.1.A.1.h, I, and j revise the requirements associated with the start of an RCP. These changes improve specification clarity and do not increase the probability or consequences of an accident.

The Technical Specification changes associated with the restriction on SI pumps provides added conservatism to the Technical Specifications and limits the likelihood of an RHR overpressurization event. Current plant procedures prohibit actuation of any SI pumps when RHR is in service, except during testing, loss of RHR cooling, or reduced inventory operations. Therefore, the change to the Technical Specifications will not alter current plant operation.

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

Response:

The proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously analyzed. The pressure-temperature limits are updating the existing limits by taking into account the effects of radiation embrittlement, utilizing criteria defined in Regulatory Guide 1.99, Revision 2, and extending the effective period to 13.3 EFPYs. The updated OPS limits have been adjusted to account for the effect of irradiation on the limiting reactor vessel material. These changes do not affect the way the pressure-temperature or OPS limits provide plant protection and no physical plant alterations are necessary.

The revisions to Section 3.1.A.8 concerning the OPS system improve on the clarity of existing specifications and add a completion time for compensating actions that is consistent with the STS. These changes do not involve any hardware modifications and do not affect the function of the OPS system.

The revisions concerning the operation of SI pumps bring the Technical Specifications into line with current operating procedures. The changes to Specification 3.1.A.1.h, I, and j provide specification clarity and are more conservative than existing Technical Specifications. Therefore, the changes cannot create the possibility of a new or different kind of accident.

(3) Does the proposed amendment involve a significant reduction in a margin of safety?

Response:

The proposed amendment does not involve a significant reduction in a margin of safety. The margins of safety against fracture provided by the pressure-temperature limits are those limits specified in 10 CFR Part 50, Appendix G, ASME Boiler and Pressure Vessel Code Section XI, Appendix G, and Reference 4. The guidance in these documents has been utilized to develop the pressure-temperature limits with the requisite margins of safety for the heatup and cooldown conditions. The new LTOP limits are based upon References 2 and 5 and ASME Code Case N-514.

The revisions to Section 3.1.A.8 clarify the requirements associated with the OPS system. The revisions associated with the operation of SI pumps with RHR in service (Sections 3.3.A.3, 8, 9 and 10) and the changes regarding RCP starts (Section 3.1.A.1.h, I, and j) are more conservative than the current Technical Specifications, and are consistent with plant operating procedures. Therefore, they do not reduce a margin of safety.

Section IV - Impact of Changes

These changes will not adversely affect the following:

ALARA Program
Security and Fire Protection Programs
Emergency Plan
FSAR or SER Conclusions
Overall Plant Operations and the Environment

Section V - Conclusions

The incorporation of these changes: a) will not increase the probability nor the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the

Safety Analysis Report; b) will not increase the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report; c) will not reduce the margin of safety as defined in the bases for any technical specification; and d) involves no significant hazards considerations as defined in 10 CFR 50.92.

Section VI - References

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

2. NYPA letter to the NRC (IPN-90-046), "Proposed Changes to Technical Specifications Regarding Pressure-Temperature Limits," dated August 31, 1990.

3. NYPA letter to NRC (IPN-98-015), "Response to Request for Additional Information on Proposed Technical Specification Changes Regarding Pressure-Temperature Limits," dated February 6, 1998.

4. NYPA letter to the NRC (IPN-98-013), "Proposed Exemption From Requirements of 10 CFR 50.60 to Utilize Alternate Methodology to Determine K_{IT}," dated January 28, 1998.

5. NYPA letter to NRC (IPN-98-023), "Response to Request for Additional Information on Proposed Technical Specification Changes Regarding Overpressure Protection System Limits," dated February 27, 1998.

6. NUREG-1431, Revision 1, "Standard Technical Specifications - Westinghouse Plants," April 1995.

7. IP3 Low Temperature Overpressurization Protection System Analysis, NYPA Report dated August 24, 1984.

8. Westinghouse Report, "IP3 RHRS Relief Valve Evaluation Report," dated May 1994 (SE/SS-INT-7901).

9. NYPA letter to the NRC (IPN-97-149), "Proposed Exemption From Requirements of 10 CFR 50.60 and Proposed Technical Specification Changes Associated With Pressure-Temperature and Overpressure Protection System Limits for Up to 13 Effective Full Power Years," dated November 3, 1997.

ATTACHMENT III TO IPN-98-024

MARK-UP OF REVISED TECHNICAL SPECIFICATION PAGES ASSOCIATED WITH AMENDMENT REQUEST FOR PROPOSED TECHNICAL SPECIFICATION CHANGES ASSOCIATED WITH PRESSURE-TEMPERATURE AND OVERPRESSURE PROTECTION SYSTEM LIMITS FOR UP TO 13.3 EFFECTIVE FULL POWER YEARS

(FOR INFORMATION ONLY)

NOTE 1: Deletions are shown in ~~strikeout~~, and additions are shown in **bold**.

NOTE 2: Previous amendment revision bars are not shown.

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

LIST OF FIGURES

Amendment No. ~~101~~, ~~103~~, ~~116~~, ~~143~~, ~~173~~,

d. When the reactor coolant system T_{avg} is less than 200°F, but not in the refueling operation condition, and as permitted during special plant evolutions, at least one residual heat removal pump (connected to the Reactor Coolant System) shall be in operation. This RHR pump may be out of service for up to 1 hour provided no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and core outlet temperature is maintained at least 10°F below saturation temperature.

e. When the reactor is critical and above 2% rated power, except for natural circulation tests, at least two reactor coolant pumps shall be in operation.

f. The reactor shall not be operated at power levels above 10% rated power with less than four (4) reactor coolant loops in operation.

g. If the requirements of 3.1.A.1.e and 3.1.A.1.f above cannot be satisfied, the reactor shall be brought to the hot shutdown condition within 1 hour.

h. A reactor coolant pump (RCP) may not be started (or jogged) when the RCS cold leg temperature (T_{cold}) is at or below ~~332°F~~ **319°F**, ~~with no other RCP's operating~~, unless RCS make up is not in excess of RCS losses, and one of the following requirements is met:

(1) The OPS is <u>operable</u>, steam generator pressure is not decreasing, and the temperature of each steam generator is less than or equal to the coldest T_{cold};

Or

(2) The OPS is <u>operable</u>, the temperature of the hottest steam generator exceeds the coldest T_{cold} by no more than 64°F, pressurizer level is at or below ~~75~~ **73** percent, and T_{cold} is as per Figure 3.1.A-1;

Or

(3) ~~The~~ **With** OPS ~~is~~ <u>inoperable</u>, steam generator pressure is not decreasing, the temperature of each steam generator is less than or equal to the coldest T_{cold}, pressurizer level is at or below ~~75~~ **73** percent, and the RCS pressure does not exceed that given by ~~Curve I on~~ Fig. 3.1.A-2~~,~~. **The pressurizer level must be further restricted per Figures 3.1.A-5 and 3.1.A-6 if operation below 319°F exceeds 8 hours.**

~~Or~~

~~(4) The OPS is inoperable, the temperature of the hottest steam generator exceeds the coldest T_{cold} by no more than 64°F, and pressurizer level and RCS pressure do not exceed the boundaries given on Fig. 3.1.A-4.~~

Amendment 48, 53, 67, 84, 93, 121,

~~i. Additional pumps may not be started (or jogged) unless the OPS is operable and the pressurizer level is not increasing.~~

~~(1) Specification 3.1.A.1.i above may be modified to allow the OPS inoperable, providing the temperature of each steam generator has remained less than or equal to the coldest T_{cold} since the first RCP start, pressurizer level is at or below 75 percent and the RCS pressure does not exceed that given by Curve I on Fig. 3.1.A-2.~~

~~(2) Specification 3.1.A.1.i above may be further modified to allow the OPS inoperable and the temperature of the hottest steam generator to be no greater than 64°F higher than the coldest T_{cold}, provided that pressurizer level is at or below 75 percent and RCS pressure does not exceed that given by Curve II on Fig. 3.1.A-2.~~

~~j. Following the start of one or more RCP's and prior to reaching 332°F, the RCS pressure shall not exceed that given by Curves I and II on Fig. 3.1.A-3 as appropriate.~~

This page was intentionally left blank.

7. REACTOR VESSEL HEAD VENTS

Whenever the reactor coolant system is above 350°F, two reactor vessel head vent paths consisting of two valves in series with power available from emergency buses shall be OPERABLE.

a. If one of the above reactor vessel head vent paths is inoperable, startup and/or power operation may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of all the valves in the inoperable vent path. Restore the inoperable vent path to operable status within 90 days, or be in hot shutdown within 6 hours and be below 350°F within the following 30 hours.

b. With both reactor vessel head vent paths inoperable restore one vent path to operable status within 7 days or be in hot shutdown within 6 hours and be below 350°F within the following 30 hours.

8. OVERPRESSURE PROTECTION SYSTEM (OPS)

a. ~~When the RCS temperature is below 332°F and the RCS is not depressurized and vented with an equivalent opening of at least 2.00 square inches, the OPS shall be "armed" and "operable". Both OPS PORVs shall have lift settings not to exceed those given by Curve III (OPS PORV setpoint limit curve) on Fig. 3.1.A-3.~~ **When the RCS temperature is below 319°F,**

(1) the OPS shall be armed and operable. Both OPS PORVs shall have lift settings not to exceed those given in Figure 3.1.A-2, or

(2) the RCS must be vented with an equivalent opening of 2.00 square inches.

b. ~~The requirements of 3.1.A.8.a. may be modified to allow one PORV and/or its series MOV to be inoperable for a maximum of seven (7) consecutive days. If the single PORV and/or its series MOV are not restored to meet the requirements of 3.1.A.8.a. within the seven (7) day period, or if both PORVs and/or their series MOVs are inoperable when required to be operable by 3.1.A.8.a., then one of the following actions shall be performed.~~ **The requirements of 3.1.A.8.a may be modified to allow one PORV and/or its series block valve to be inoperable for a maximum of seven (7) consecutive days.**

c. If the requirements of 3.1.A.8.a or 3.1.A.8.b cannot be met, then one of the following actions shall be completed within 8 hours.

(1) The RCS must be depressurized and vented with an equivalent opening of at least 2.00 square inches;

Or

(2) The RCS must be heated in accordance with **the restrictions of** ~~Specifications~~ **Specification** 3.1.A.1.h(3) ~~and (4)~~ and maintained above ~~370°F~~ **411°F;**

Or

(3) Restrict pressurizer level as per the curves ~~referenced below:~~ **on Figures 3.1.A-5 and 3.1.A-6.**

~~For up to 1 charging pump in operation with no SI pumps energized and aligned to feed the Reactor Coolant System, see Fig. 3.1.A-5.~~

Amendment No. ~~63~~, ~~67~~, ~~121~~,

~~For up to 3 charging pumps in operation concurrent with up to 1 SI pump energized and aligned to feed the Reactor Coolant System, see Fig. 3.1.A-6.~~

~~e~~d. In the event the PORV's or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2.j within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or vent(s) on the transient and any corrective action necessary to prevent recurrence.

The requirement that 150 kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at hot shutdown.

The power operated relief valves (PORVs) 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 ensure the ability to seal off possible RCS leakage paths.

Reactor vessel head vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one reactor vessel head vent path ensures that capability exists to perform this function.

The valve redundancy of the reactor coolant system vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve power supply or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the reactor coolant system vent systems are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November, 1980.

The OPS is designed to relieve the RCS pressure for certain unlikely incidents to prevent the peak RCS pressure from exceeding the ~~10 CFR 50, Appendix G,~~ limits **established in Reg. Guide 1.99, Revision 2. The OPS is considered to be operable when the minimum number of required channels (per Table 3.5-3) are available to open the PORVs upon receipt of a high pressure signal which is based upon RCS T_{cold}, as shown in Figure 3.1.A-2. (The happy face icon contained on this and other Technical Specification figures indicates the side of the applicable curve in which operation is permissible. Conversely, the sad face icon indicates the side of the applicable curve in which operation is prohibited.) The OPS setpoint is based upon a comparative analysis of References 5 and 9, with allowances for metal/fluid temperature differences (as described below) and for the static head due to elevation differences and dynamic head effect of the operation of the reactor coolant and RHR pumps**. "Arming" means that the motor operated **block** valve (MOV) is in the open position. This can be accomplished either automatically by the OPS when the RCS temperature is less than or equal to ~~332°F~~ **319°F** or manually by the control room operator.

The start of an RCP is allowed when the steam generators' temperature does not exceed the RCS and the OPS is operable ~~(i.e., both PORVs available)~~. During all modes of operation, the steam generator temperature may be measured using the Control Room instrumentation or, as a backup, from a contact reading off the steam generator's shells.

Most start-ups will satisfy these requirements as provided in Specification 3.1.A.1.h (1). In order to allow start of an RCP when the steam generators are hotter than the RCS, requirements for a pressurizer bubble (gas or steam) are developed **(technical specification value for pressurizer level includes an allowance for instrument uncertainty)**. During this Heat Input initiation event the RCS fluid temperature rise is considerably more rapid than the reactor vessel metal temperature rise. Since OPS utilizes a setpoint curve (~~Fig. 3.1.A-3, curve III~~ **Figure 3.1.A-2**) and the temperature measured is the fluid temperature, and not the reactor vessel metal, it is necessary to shift to the right the OPS setpoint curve **by 50°F** to ensure the pressure does not exceed the allowable ~~(appendix G)~~ values for the vessel. For the conditions when the OPS is inoperable, additional requirements are developed for the pressurizer bubble, RCS pressure and temperature.

Due to the rate of energy transferred to the RCS, when the RCP is started, the resultant rate of temperature rise and the pressure increase are strongly dependent on the temperature difference between the RCS and the steam generators. The presence of a pressurizer bubble provides for a more moderate pressure increase. The bubble size is sufficient to prevent the RCS from going water solid for 10 minutes during which time operator action will terminate the pressure transient. Pressurizer level refers to indicated level and includes instrument uncertainty. The preventive measures for a Mass Input initiating event (i.e., ~~SI pump or~~ **up to three** charging pumps **and/or one SI pump**) as well as the Heat Input initiating event are described in References (3), (4) and (5). (Also refer to ~~Specification~~ **Specifications** 3.3.A.8, **3.3.A.9, and 3.3.A.10.)** ~~Safety Injection and Residual Heat Removal Systems).~~ The OPS need not be operable when the RCS temperature is less than ~~332°F~~ **319°F** if the RCS is depressurized and vented with an equivalent opening of at least 2.00 square inches. **One PORV, blocked fully open, also satisfies this vent area requirement.** This opening is adequate to relieve the worst case analyzed. **It should be noted that the analysis of record (Reference 5) is based upon a minimum vent area of 1.4 square inches, which for the sake of conservatism has been rounded up to 2.00 square inches.**

The OPS ~~arming~~ **enable** temperature of ~~332°F~~ **319°F** permits the performance of an RCS hydrostatic test (see Fig. 4.3-1) without activating the OPS.

Upon OPS inoperability, the RCS may be heated above ~~370°F~~ **411°F.** This temperature is that value for which the RCS heatup and cooldown curves (Figures 3.1-1 and 3.1-2) permit pressurization to the setting of the pressurizer safety valves. Accordingly, with an inoperable OPS and an RCS temperature ~~370°F~~ **above 411°F**, the pressurizer safety valves will preclude violation of the 10 CFR 50, Appendix G, curves. In addition, the OPS need not be operable upon satisfying the conditions of Specification ~~3.1.A.8.b(3)~~ **3.1.A.8.c(3)** which requires the presence of a pressurizer bubble to preclude RCS overpressurization during inadvertent mass inputs. Specification ~~3.1.A.8.b(3)~~ **3.1.A.8.c(3)** also places restrictions on the number of **charging and** SI pumps capable of feeding the RCS (see ~~Specification~~ **Specifications** 3.3.A.8, **3.3.A.9, and 3.3.A.10**). ~~An SI~~ **Any** pump can be rendered incapable of feeding the RCS if, for example, its switch is in the trip pull-out position, or if at least one valve in the flow path ~~from the SI pump~~ to the

RCS is closed and locked (if manual) or de-energized (if motor operated). This section has also been revised in accordance with the results of tests conducted on the capsule "T", **Y, and Z** specimens (~~Reference~~ **References** 6, **7 and 8**).

Amendment No. 67, 84, 121,

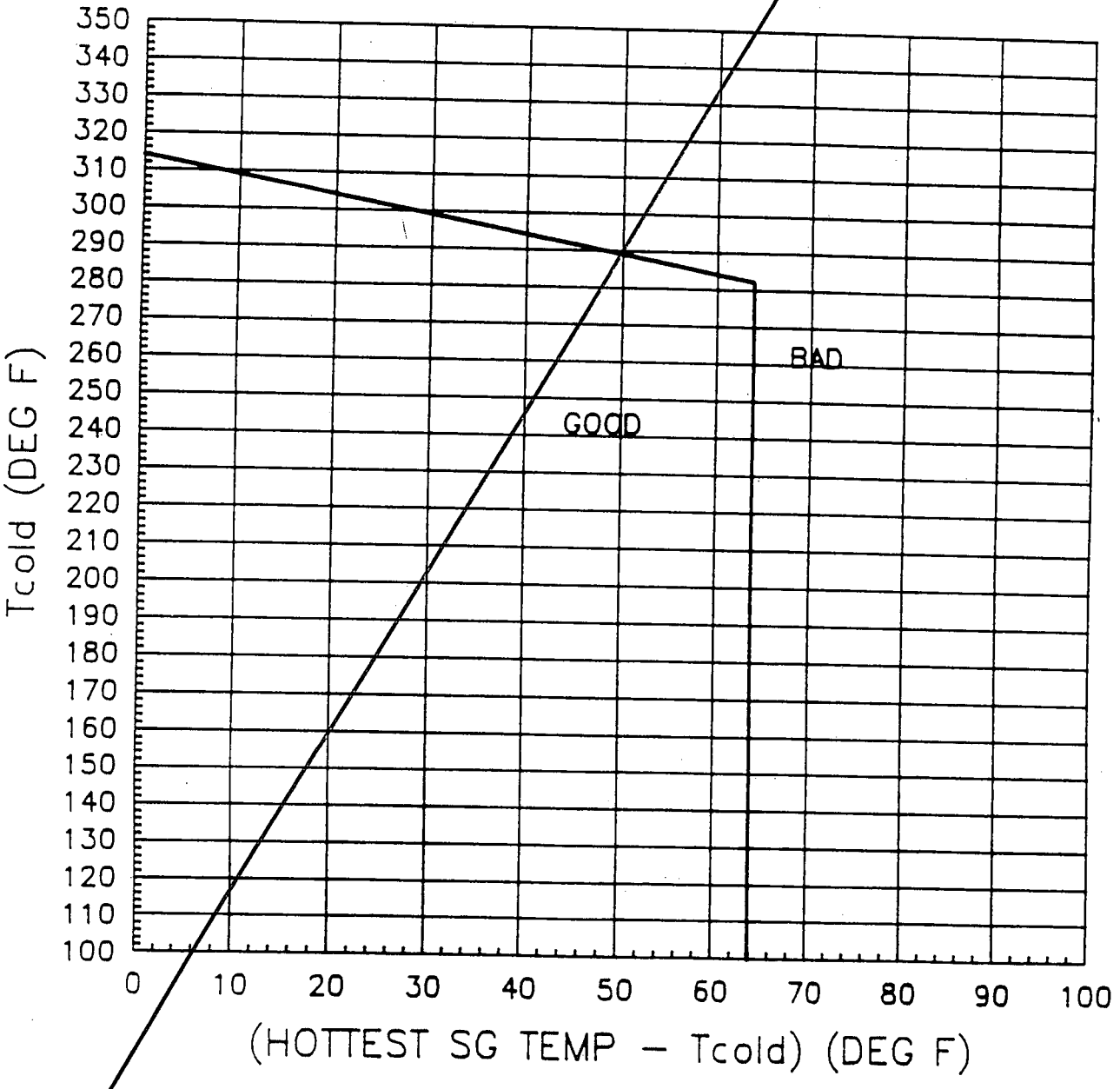
References

1) FSAR Section 14.1.6
2) FSAR Section 14.1.8
3) Letter dated 10/25/78 "Summary of Changes to IP-3 Plant Operating Procedures in Order to Preclude RCS Overpressurization"
4) Letter dated 2/28/76 "Conceptual Design of the Reactor Coolant Overpressure Protection System" and response to NRC questions.
5) IP-3 Low Temperature Overpressurization Protection System Analysis, NYPA Report dated 8/24/84.
6) WCAP-9491 "Analysis of Capsule T from IP-3 Reactor Vessel Radiation Surveillance Program", J.A. Davidson, S.L. Anderson, W.T. Kaiser, April 1979.
7) **WCAP-10300-1, "Analysis of Capsule Y from the Power Authority of the State of New York Indian Point Unit 3 Reactor Vessel Radiation Surveillance Program," S.E. Yanichko, S.L. Anderson, March 1983.**
8) **WCAP-11815, "Analysis of Capsule Z from the New York Power Authority Indian Point Unit 3 Reactor Vessel Radiation Surveillance Program," S.E. Yanichko, S.L. Anderson, L. Albertin, March 1988.**
9) **ASME Code Case N-514, "Low Temperature Overpressure Protection," February 12, 1992.**

REPLACE WITH ATTACHED PAGE

FIGURE 3.1.A-1

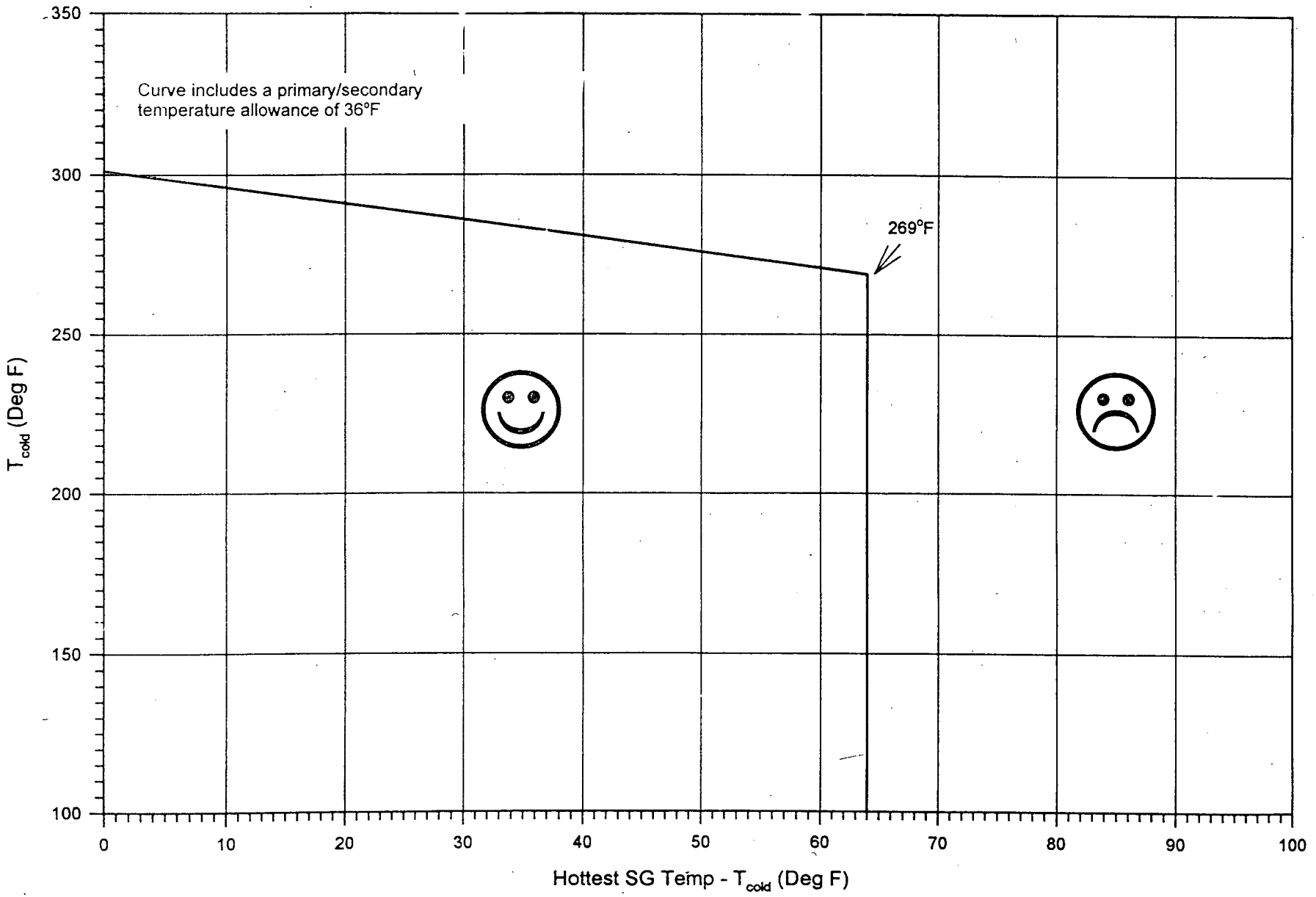
MAXIMUM PERMISSIBLE Tcold FOR FIRST RCP START (OPS OPERABLE, HOTTEST SG TEMP > Tcold)



Amendment No. ~~87~~, 121

FI 3.1.A-1

Secondary Side Limitations for RCP Start With
Secondary Side Hotter Than Primary, 13.3 EFPY



Amendment No. 67, 122,

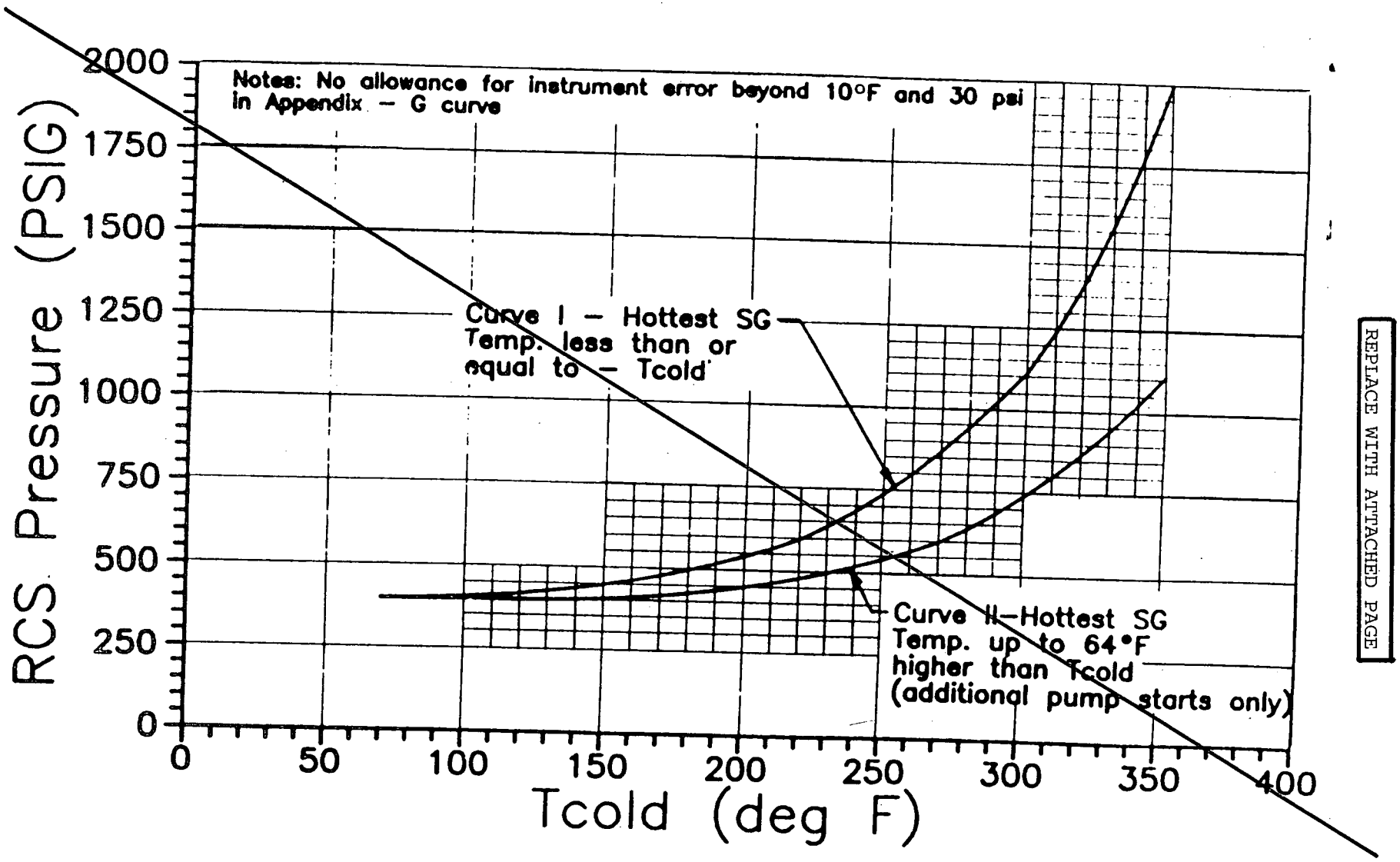


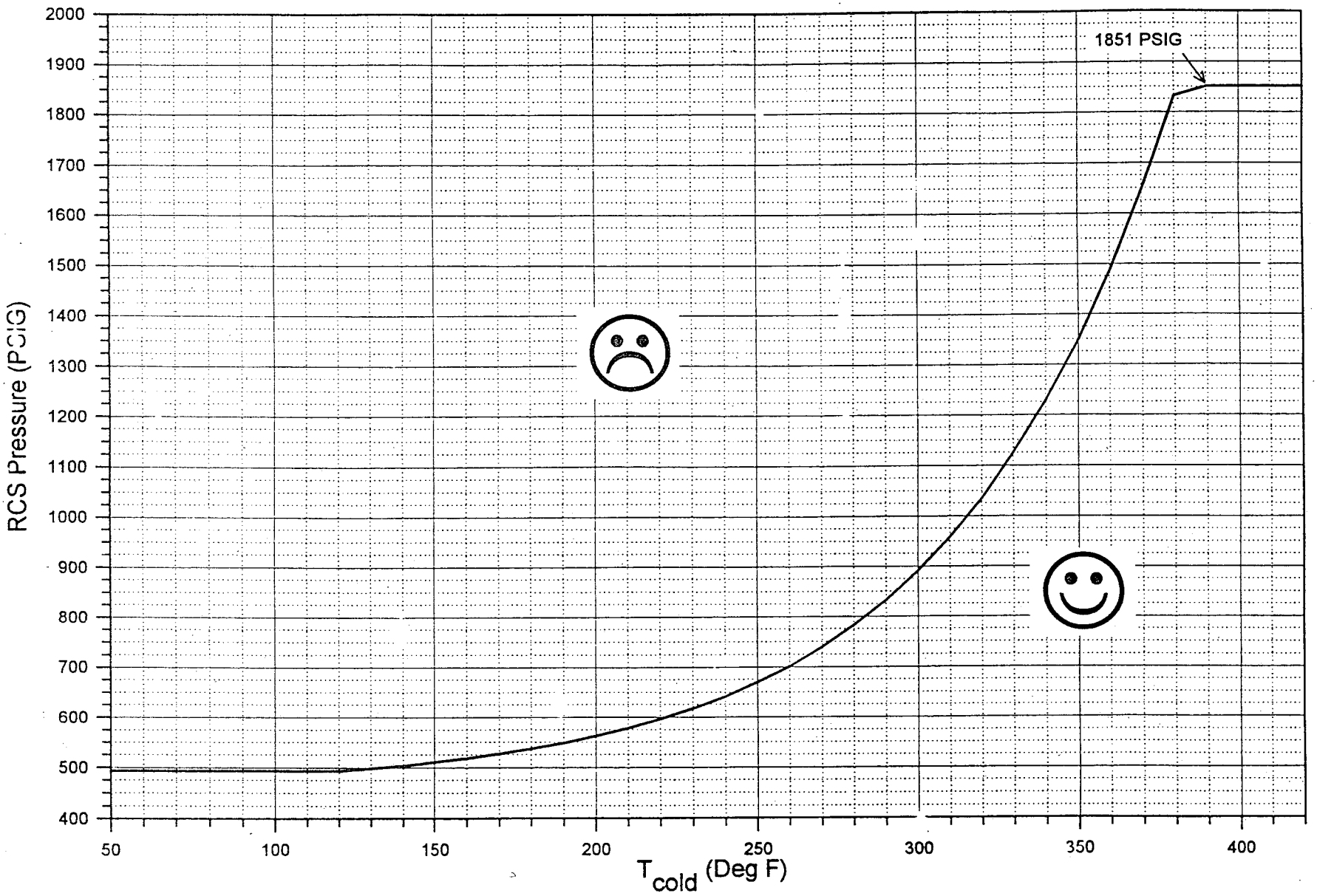
Figure 3.1.A–2 Figure applicable to 11 EFPY

Maximum Permissible RCS Pressure for RCP Start with OPS Inoperable

REPLACE WITH ATTACHED PAGE

Figure 3.1.A-2

Maximum Allowable Nominal PORV Setpoint for the
Low Temperature Overpressure Protection System (OPS), 13.3 EFPY



Analytical Curve

Amendment No. 67, 121,

FIGURE 3.1.A-3

~~RCS PRESSURE LIMITS FOR LOW TEMPERATURE OPERATION~~

DELETED

Amendment No. 67, 121, 154,

FIGURE 3.1.A-4

~~MAXIMUM PRESSURIZER LEVEL FOR OPS INOPERABLE AND FIRST RCP START (STEAM GENERATOR TEMPERATURE GREATER THAN Tcold)~~

DELETED

Amendment No. ~~67~~, ~~121~~,

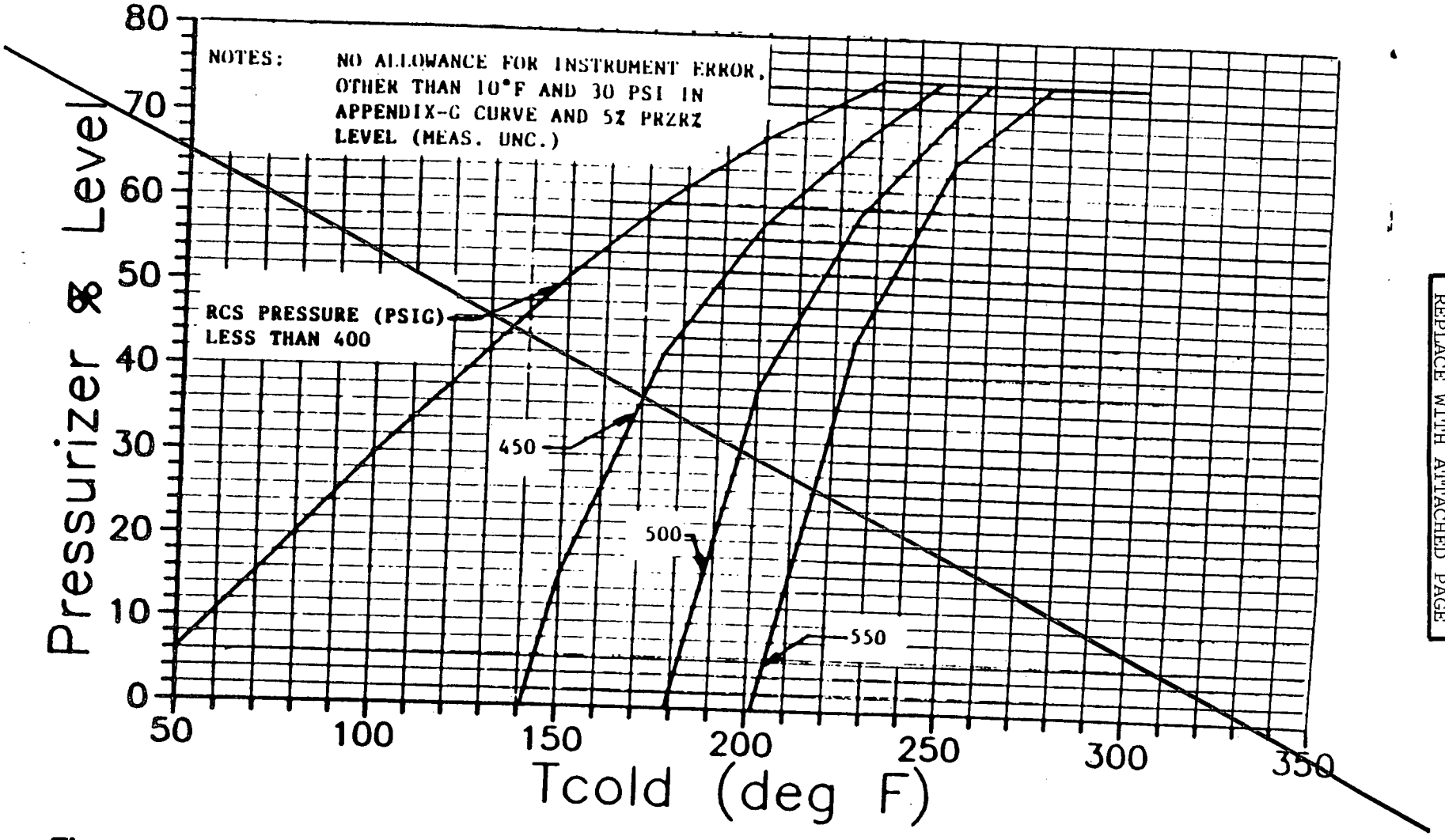


Figure 3.1.A-5

Figure applicable to 11 EFPY

MAXIMUM PRESSURIZER LEVEL WITH OPS INOPERABLE AND ONE (1) CHARGING PUMP ENERGIZED

Amendment No. 67, ~~XOX~~, 121

REPLACE WITH ATTACHED PAGE

FIG[illegible]3.1.A-5

Pressurizer Limitations for OPS Inoperable
(Up to one charging pump capable of feeding RCS), 13.3 EFPY

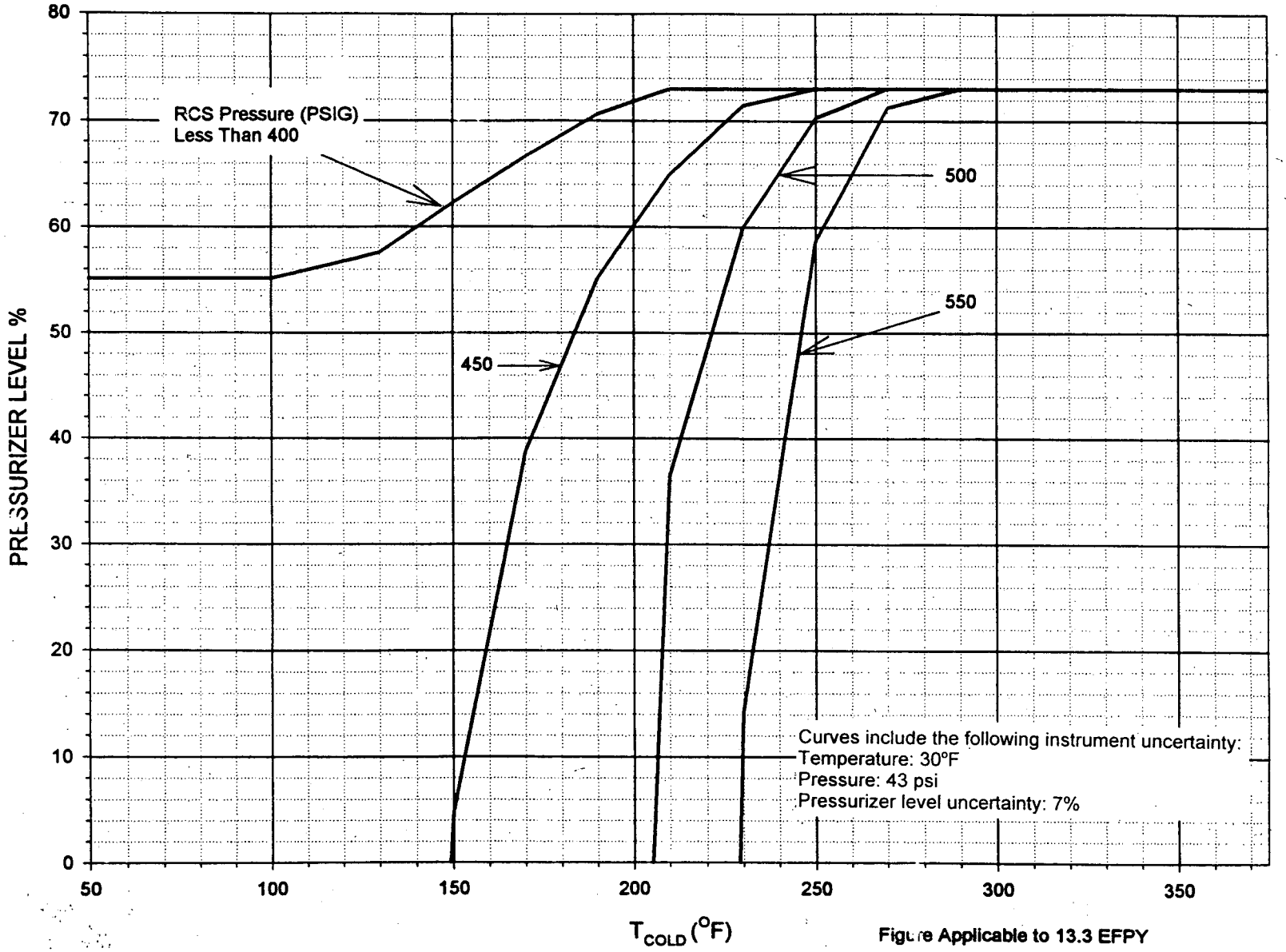


Figure Applicable to 13.3 EFPY

Curves represent maximum allowable pressurizer level for the conditions defined.

Amendment No. 67, 101, 121,

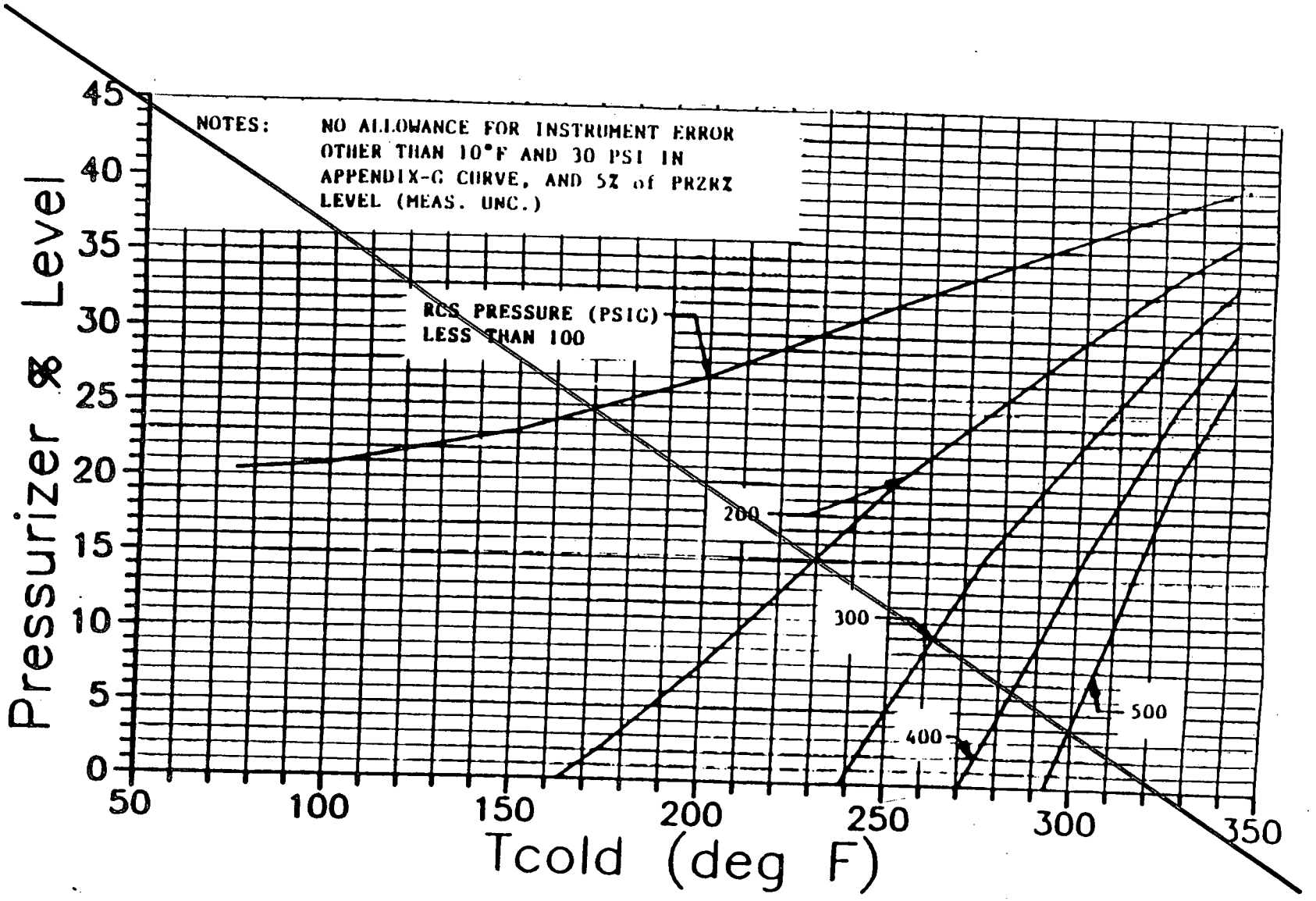


Figure 3.1.A-6

Figure applicable to 11 EFPY

MAXIMUM PRESSURIZER LEVEL WITH OPS INOPERABLE AND ONE (1) SAFETY INJECTION PUMP AND/OR THREE (3) CHARGING PUMPS ENERGIZED

Amendment No. 81, ~~102~~, 121

REPLACE WITH ATTACHED PAGE

FIGURE 3.1.A-6

Pressurizer Limitations for OPS Inoperable (up to three charging pumps and/or one safety injection pump capable of feeding RCS), 13.3 EFPY

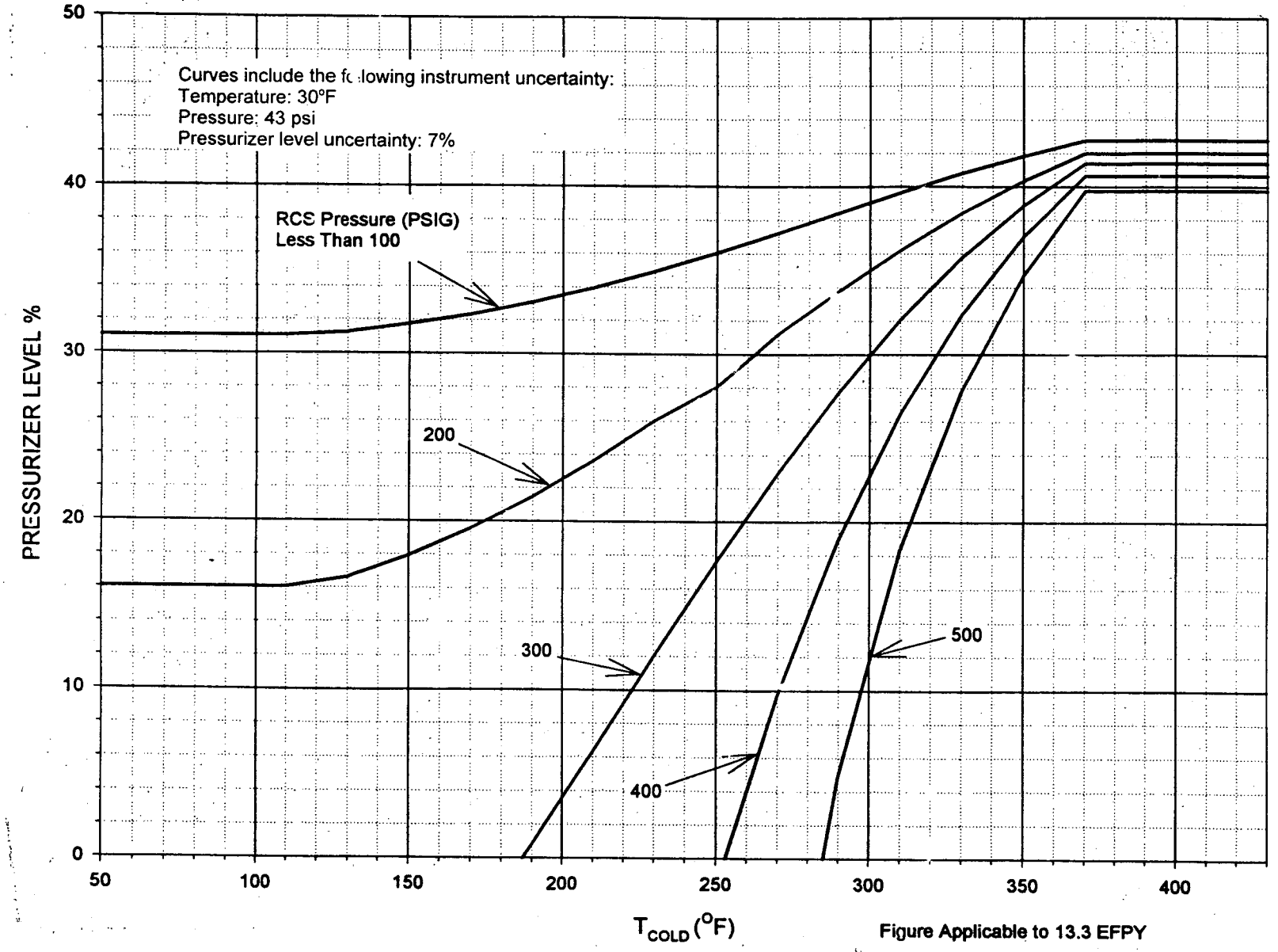


Figure Applicable to 13.3 EFPY

Curves represent maximum allowable pressurizer level for the conditions defined.

Amendment No. 67, 101, 121,

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 3.1-1 and Figure 3.1-2 for the service period up to ~~11.00~~ **13.3** effective full-power years (EFPYs). The heatup and cooldown rates shall not exceed 60°F/hr and 100°F/hr respectively.

   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.

2. The limit lines shown in Figure 3.1-1 and Figure 3.1-2 shall be recalculated periodically using methods discussed in the Basis and results of surveillance specimens as covered in Specification 4.2. The order of specimen removal may be modified based on the results of testing of previously removed specimens.

3. 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.

4. 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.

5. Reactor Coolant System integrity tests shall be performed in accordance with Section 4.3.

Basis

Fracture Toughness Properties

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the Summer 1965 Section III of the ASME Boiler and Pressure Vessel Code [6] and ASTM E185 [5] and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1972 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code [1], and the calculation methods described in WCAP-7924 [2].

The first reactor vessel material surveillance capsule was removed during the 1978 refueling outage. This capsule has been tested by Westinghouse Corporation and the results have been evaluated and reported [7]. Similar reports were prepared for the surveillance capsules [10, 8] removed in 1982 and 1987. Based on the Westinghouse evaluation, heatup and cooldown curves (Figures 3.1-1 and 3.1-2) were developed for up to ~~11.00~~ **13.3** EFPYs of reactor operation.

Generic Letter 88-11 requested that licensees use the methodology of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials", to predict the effect of neutron radiation on reactor vessel materials as required by paragraph V.A. of 10 CFR part 50, Appendix G. Capsule Z was analyzed [8] and new pressure-temperature curves were developed using this methodology.

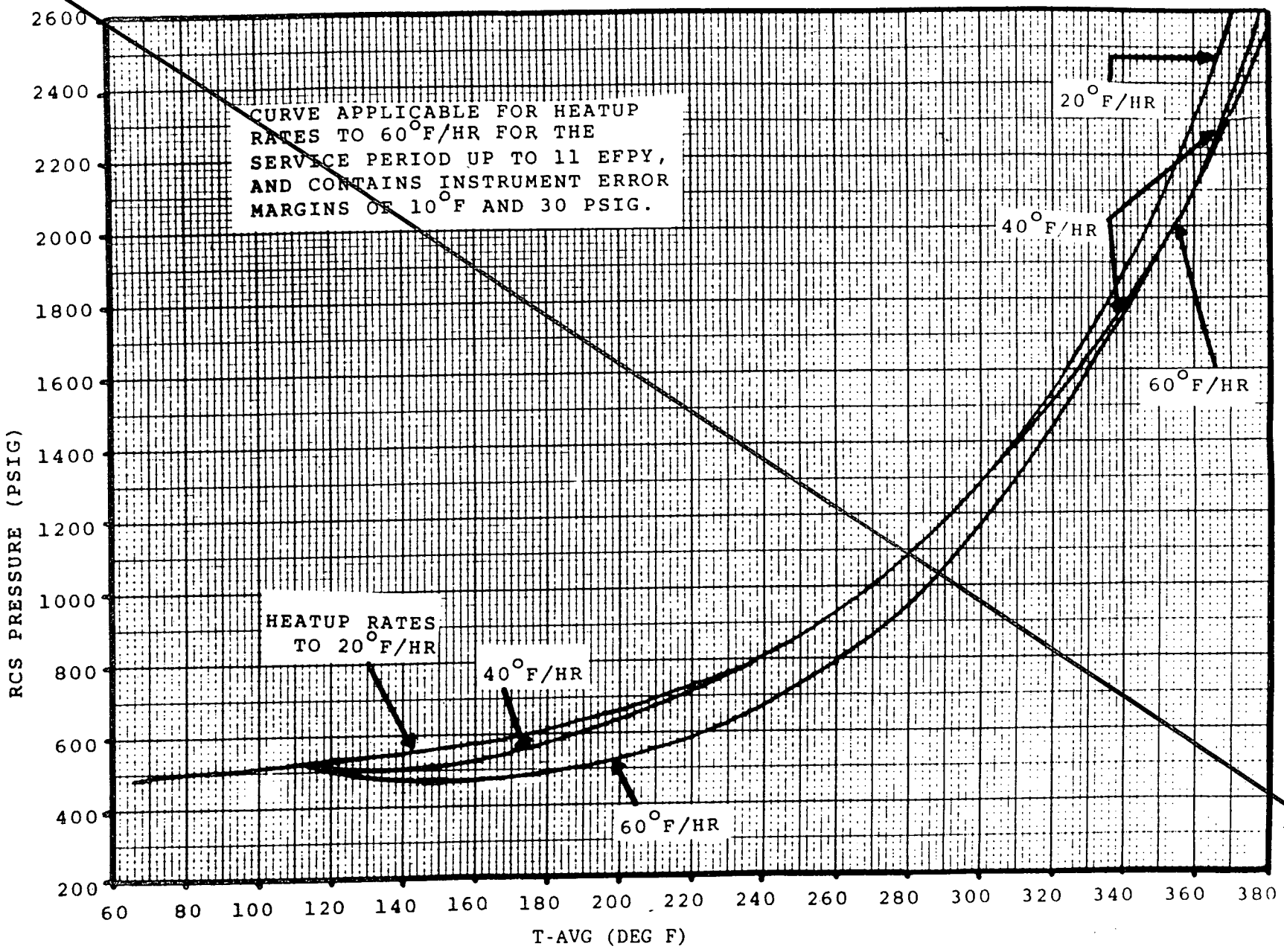
The maximum ~~shift~~ **value** in RT_{NDT} after ~~11.00~~ **13.3** EFPYs of operation is projected to be ~~202°F~~ **214°F** at the 1/4 T and ~~163°F~~ **172°F** at the 3/4 T vessel wall locations for Plate B2803-3 the controlling plate. Plate B2803-3 was also the controlling plate for the operating period up to ~~9.00~~ **11** EFPYs.

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} at the end of ~~11.00~~ **13.3** years of service life. The ~~11.00~~ **13.3** year service life period is chosen such that the limiting RT_{NDT} at the 1/4 T location in the core region is higher than the RT_{NDT} of the limiting unirradiated material. This service period assures that all components in the Reactor Coolant System will be operated conservatively in accordance with Code recommendations.

The highest RT_{NDT} of the core region material is determined by adding the radiation induced ΔRT_{NDT} for the applicable time period to the original RT_{NDT} shown in Table Q4.2-1 [3].

REPLACE WITH ATTACHED PAGE

FIGURE 3.1-1
Reactor Coolant System Heatup Limitations - Indian Point 3

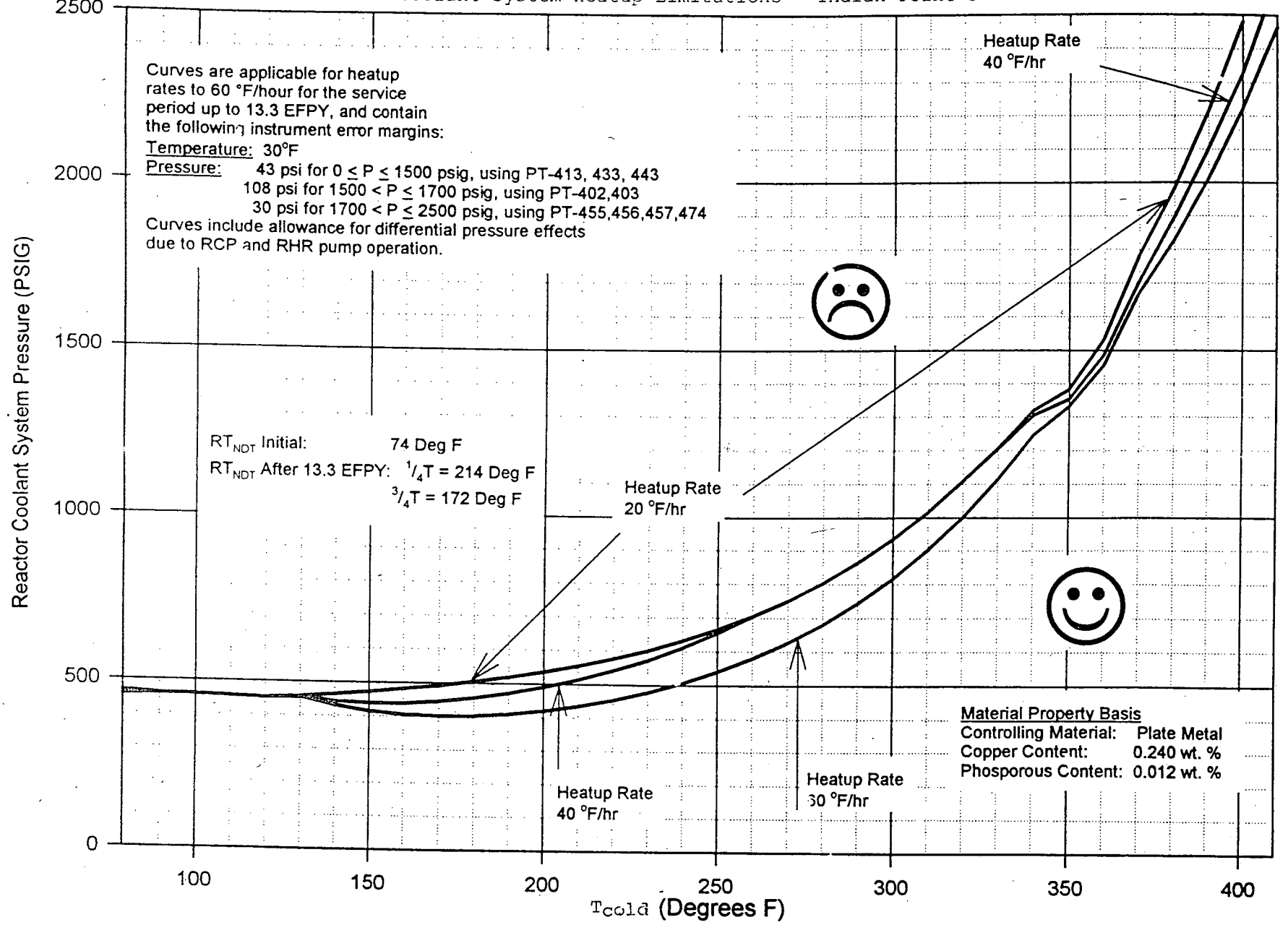


MATERIAL PROPERTY BASIS

Controlling Material:	Plate Metal	RT_{NDT} INITIAL:	74°F
Copper Content:	0.24 wt. %	RT_{NDT} AFTER 11 EFPY:	1/4T = 202°F
Phosphorous Content:	0.012 wt. %		3/4T = 163°F

FIGURE 3.1-1

Reactor Coolant System Heatup Limitations - Indian Point 3



Amendment No. 28, 100, 121,

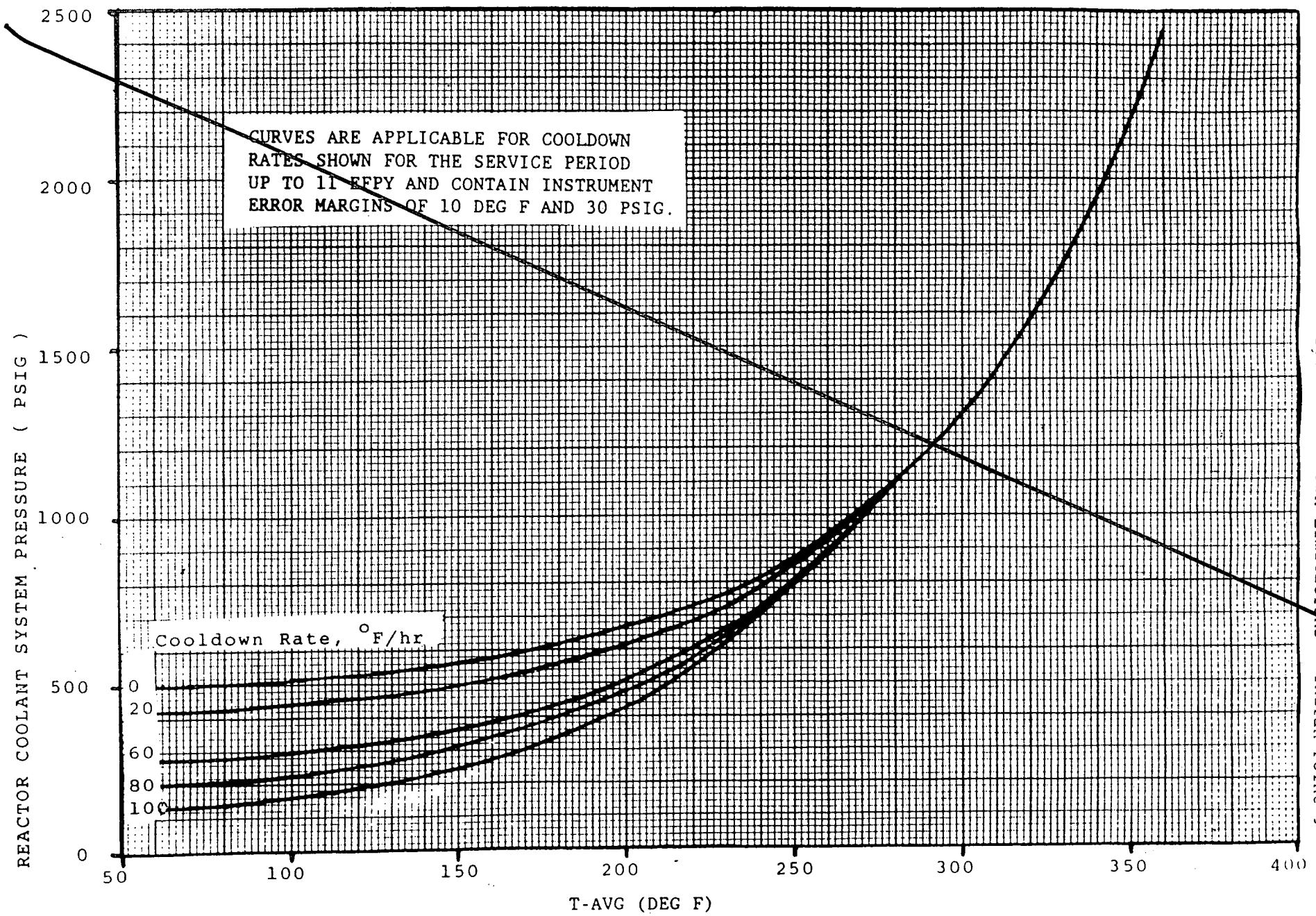


FIGURE 3.1-2
Reactor Coolant System Cooldown Limitations - Indian Point 3

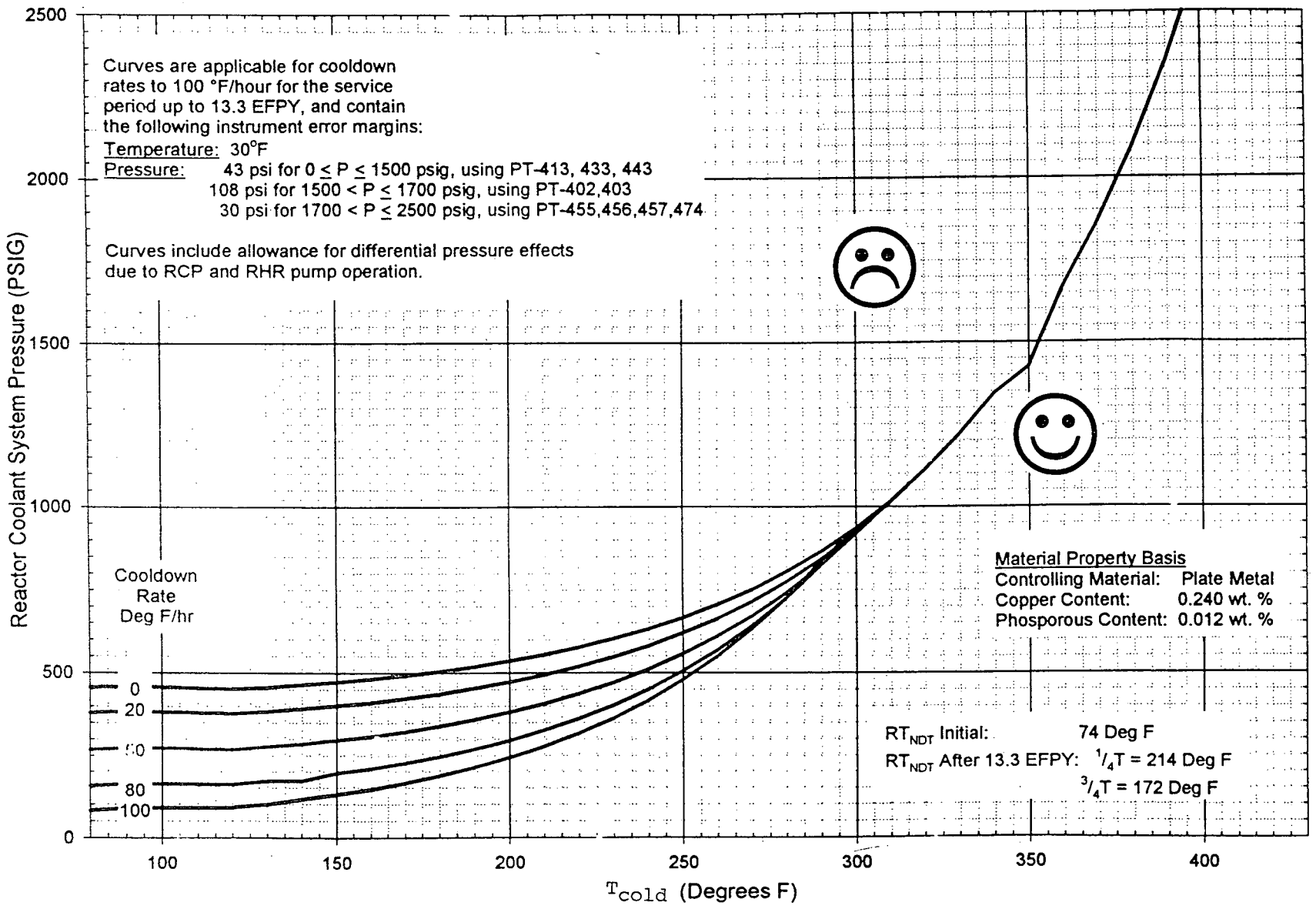
REPLACE WITH ATTACHED PAGE

MATERIAL PROPERTY BASIS

Controlling Material:	Plate Metal	RT_{NDT} INITIAL:	74°F
Copper Content:	0.24 wt. %	RT_{NDT} AFTER 11 EFPY:	1/4T = 202°F
Phosphorous Content:	0.012 wt. %		3/4T = 163°F

FIGURE 3.1-2

Reactor Coolant System Cooldown Limitations - Indian Point 3



Amendment No. 28, 109, 121,

d. One pressure and one level transmitter shall be operating per accumulator.

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

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

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

h. Valves 856B and 856G in the Safety Injection discharge headers shall be closed and their power supplies de-energized.

i. Valve 1810 in the suction line of the high-level SI pumps and valves 882 and 744 in the suction and discharge lines, respectively, of the residual heat removal pumps shall be open and their power supplies de-energized.

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

k. The refueling water storage tank low level alarms are operable and set to alarm between 10.5 feet and 12.5 feet of water in the tank.

l. Valve 883 in the RHR return line to the RWST is de-energized in the closed position.

m. Valves 1870 and 743 in the miniflow line for the Residual Heat Removal Pumps shall be open and their power supplies de-energized.

n. The RHR system is in the ESF alignment with the normal RHR suction line isolated from the RCS.

4. The requirements of 3.3.A.3 may be modified to allow any one of the following components to be inoperable at any one time:

2) RCS temperature and the source range detectors are monitored hourly;

and

3) no operations are permitted which would reduce the boron concentration of the reactor coolant system.

8. ~~When the RCS cold leg temperature (T_{cold}) is at or below 332°F, no more than one safety injection pump shall be energized and aligned to feed the RCS.~~

When the RCS average cold leg temperature (T_{cold}) is below 319°F, or when RHR is in service (i.e., not isolated from the RCS), no safety injection pumps shall be energized and aligned to feed the RCS.

9. **The requirements of 3.3.A.8 may be relaxed to allow one safety injection pump energized and aligned to feed the RCS under the following circumstances:**

   a. **emergency boration; OR**

   b. **for pump testing, for a period not to exceed 8 hours; OR**

   c. **loss of RHR cooling.**

10. **The requirements of 3.3.A.8 may be further relaxed when the RCS is < 200°F, such that two safety injection pumps may be energized and aligned to feed the RCS under the following circumstances:**

    a. **the RCS is vented with an opening greater than or equal to the size of one code pressurizer safety valve flange, OR**

    b. **indicated pressurizer level is at 0% and the plant is vented in accordance with Technical Specification 3.1.A.8.c.1. (Alternate methods and instrumentation may be used to confirm actual RCS elevation.)**

B. Containment Cooling and Iodine Removal Systems

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

   a. The spray additive tank contains a minimum of 4000 gallons of solution with a sodium hydroxide concentration ≥35% and ≤38% by weight.

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

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

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 their performance is to sufficiently limit any increase in clad temperature below a value where emergency core cooling objectives are met.[13]

During operating modes in the temperature range between 200°F and 350°F, a sufficient decay heat removal capability is provided by a reactor coolant pump with a steam generator heat sink or a residual heat removal loop. This redundancy ensures that a single failure will not result in a complete loss of decay heat removal. **Above 350°F, the normal RHR suction line is isolated from the RCS to protect RHR piping from overpressurization due to inadvertent SI pump actuation.**

During operating modes when the reactor coolant T_{avg} is less than 200°F, but not in the refueling operation condition, a sufficient decay heat removal capability is provided by a residual heat removal loop.

The containment cooling and iodine removal functions are provided by two independent systems: (a) fan-coolers plus charcoal filters and (b) 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).[4] In the event of a Design Basis Accident, any one of the following configurations will provide sufficient cooling to reduce containment pressure at a rate consistent with limiting off-site 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, any one of three configurations of fan-cooler units (with charcoal filters) and/or containment spray pumps (with sodium hydroxide addition) will reduce airborne organic and molecular iodine activities sufficiently to limit off-site doses to acceptable values.[5] Any one of these three configurations constitutes the minimum safeguards for iodine removal.

The combination of three fan-coolers and one containment spray pump is capable of being operated on emergency power with one diesel generator failing to start. Adequate power for operation of the redundant containment heat removal systems (i.e., five fan-cooler units or two containment spray pumps) is assured by the availability of off-site power or operation of all emergency diesel generators.

These toxic gas monitoring systems are designed to alarm in the control room upon detection of the short term exposure limit (STEL) value. The operability of the toxic gas monitoring systems provides assurance that the control room operators will have adequate time to take protective action in the event of an accidental toxic gas release. Selection of the gases to be monitored are based on the results described in the Indian Point Unit 3 Habitability Study for the Control Room, dated July, 1981. The alarm setpoints will be in accordance with industrial ventilation standards as defined by the American Conference of Governmental Industrial Hygienists.[16]

~~The OPS has been designed to withstand the effects of the postulated worse case Mass Input (i.e., single safety injection pump) without exceeding the 10 CFR 50, Appendix G curve. Curve III on Figure 3.1.A-3 provides the setpoint curve of the OPS PORVs which is sufficiently below the Appendix G curve such that PORVs overshoots would not exceed the allowable Appendix G pressures. Therefore, only one safety injection pump can be available to feed water into the RCS when the OPS is operable. The other pumps must be prevented from injecting water into the RCS. This may be accomplished, for example, by placing the SI pump switches in the trip pull-out position, or by closing and locking (if manual) or de-energizing (if motor operated) at least one valve in the flow path from these pumps to the RCS. For conditions when the OPS is inoperable, additional restrictions are imposed on the RCS temperature, and pressurizer pressure and level. See Specification 3.1.A.8.b.(3).~~

The RHR suction line is required to be isolated from the RCS when temperature is above 350°F. This protects the RHR system from overpressurization when the SI system is required to be in service. The requirement to prevent safety injection pumps from being able to feed the RCS under specific conditions prevents overpressurization of the RHR system or the RCS beyond the capacity of the OPS to mitigate. These conditions include when OPS is required to be in service and when RHR is in service. Special allowances are made for pump testing, loss of RHR cooling (during which time an SI pump may be required to recirculate coolant to the core), or emergency boration. Two SI pumps may be energized and aligned to feed the RCS when situations prevail that could not result in overpressurization. This is satisfied when the RCS is vented with an opening greater than or equal to the size of one code pressurizer safety flange or when the pressurizer level is low enough (indicating 0%) and the plant is vented in accordance with Technical Specification 3.1.A.8.c.1 to ensure at least a ten minute operator response time on inadvertent SI actuation without the pressurizer completely filling. Alternate methods and instrumentation may be used to confirm actual RCS elevation. Methods to ensure that an SI pump is unable to feed the RCS include placing the SI pump switches in the trip pull-out position, or by closing and locking (if manual) or de-energizing (if motor operated) at least one valve in the flow path from these pumps to the RCS.

References

1) FSAR Section 9
2) FSAR Section 6.2
3) FSAR Section 6.2
4) FSAR Section 6.3
5) FSAR Section 14.3.5

6) FSAR Section 1.2
7) FSAR Section 8.2
8) FSAR Section 9.6.1
9) FSAR Section 14.3
10) FSAR Section 6.8
11) FSAR Section 6.5
12) Response to Question 14.6, FSAR Volume 7
13) FSAR Appendix 14C
14) Response to Question 9.35, FSAR Volume 7
15) WCAP-12313, "Safety Evaluation for an Ultimate Heat Sink Temperature Increased to 95° at IP-3"
16) American Conference of Governmental Industrial Hygienists 1982 Industrial Ventilation, 19th Edition
17) NYPA calculation IP3-CALC-SI-00725, Rev. 0, "Instrument Loop Accuracy/Setpoint Calc./RWST Level."
18) Nuclear Safety Evaluation 93-3-162-SI, Rev. 0, Adequate Post-LOCA Coolant Inventory.

Amendment No. ~~67~~, ~~94~~, ~~98~~, ~~108~~, ~~145~~, ~~154~~,

4.3 REACTOR COOLANT SYSTEM (RCS) TESTING

A. 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.

Specification

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 ASME Section XI.

b) Testing of repairs, replacements or modifications for the Reactor Coolant System shall meet the requirements of ASME Section XI.

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 ~~11.00~~ **13.3** EFPYs of operations. Figure 4.3-1 will be recalculated periodically. Allowable pressures during cooldown from the leak test temperature shall be in accordance with Figure 3.1-2.

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 test temperatures are shown on Figure 4.3-1. The temperatures are calculated in accordance with ASME Code Section III, 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 ~~11.00~~ **13.3** effective full power years, it is predicted that the highest RT_{NDT} in the core region taken at the 1/4 thickness will be ~~202°F~~ **214°F**. The temperature determined by methods of ASME Code Section III for ~~2335~~ **1989** psig is ~~133°F~~ **134°F** above this RT_{NDT} and for ~~2510~~ **2485** psig (maximum) is ~~143°F~~ **153°F** above this RT_{NDT}. The minimum inservice leak test temperature requirements for periods up to ~~11.00~~ **13.3** effective full power years are shown on Figure 4.3-1[2].

The heatup limits specified on the heatup curve, Figure 4.3-1, must not be exceeded while the reactor coolant system is being heated to the inservice leak test temperature. For cooldown from the leak test temperature, the limitations of Figure 3.1-2 must not be exceeded. Figures 4.3-1 and 3.1-2 are recalculated periodically, using methods discussed in the Basis for Specification 3.1.B and results of surveillance specimens, as covered in Specification 4.2.

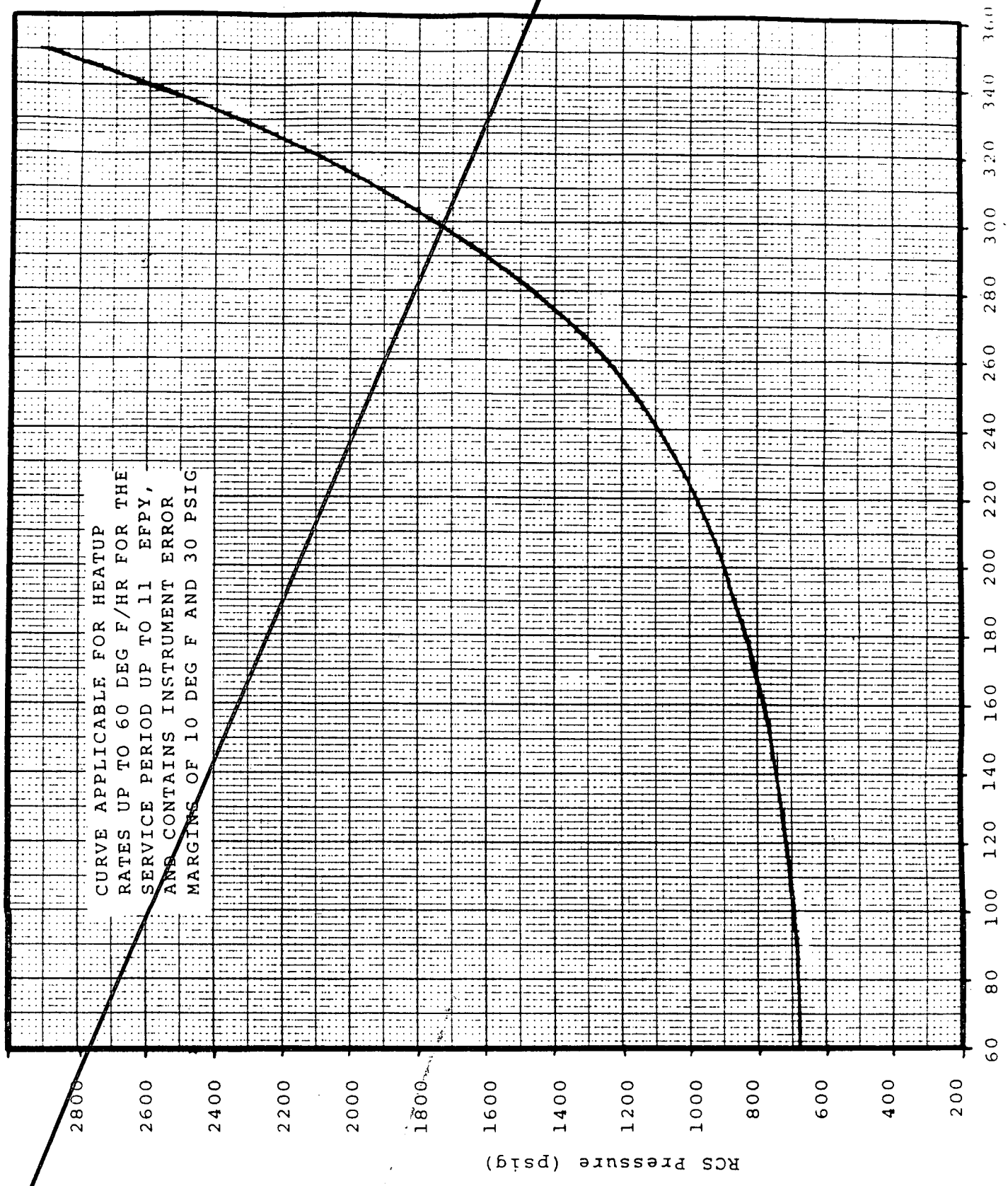
Reference

1. FSAR, Section 4.

2. "Indian Point Unit 3 Final Report on Appendix G Reactor Vessel Pressure-Temperature Limits" ABB-Combustion Engineering, July 24, 1990

REPLACE WITH ATTACHED PAGE

FIGURE 4.3-1
Pressure/Temperature Limits for Hydrostatic Leak Test



MATERIAL PROPERTY BASIS

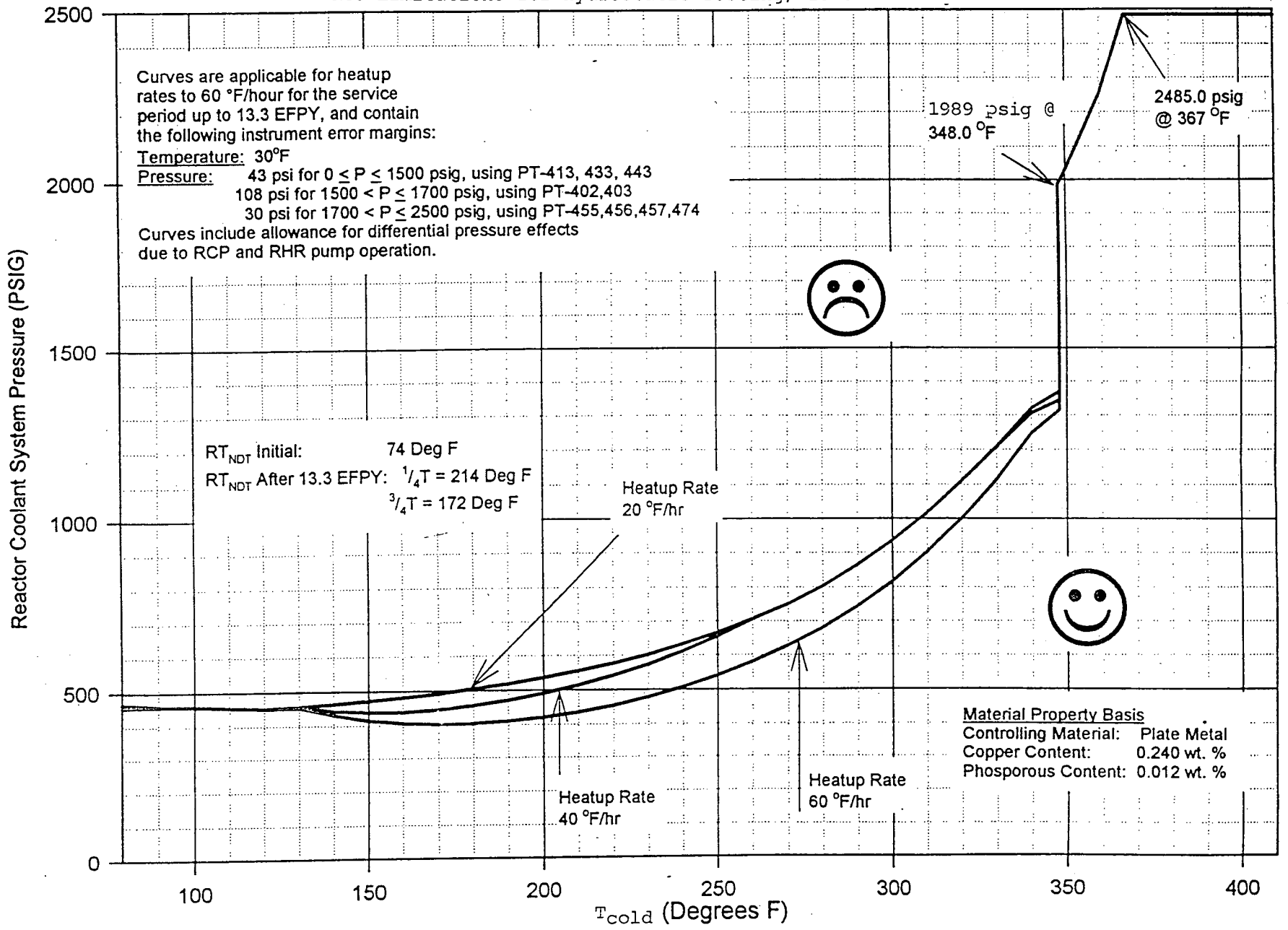
Controlling Material:	Plate Metal
Copper Content:	0.24 wt. %
Phosphorous Content:	0.012 wt. %

RT_{NDT} INITIAL:	74°F
RT_{NDT} AFTER 11 EFPY:	1/4 T = 202°F
	3/4 T = 163°F

Amendment No. 28, 109, 121

FIGURE 1

RCS Limitations for Hydrostatic Testing, 13.3 EFPY



Amendment No. 28, 109, 121,

ATTACHMENT IV TO IPN-98-024

COMMITMENTS FOR THE PROPOSED TECHNICAL SPECIFICATION CHANGES ASSOCIATED WITH PRESSURE-TEMPERATURE AND OVERPRESSURE PROTECTION SYSTEM LIMITS FOR UP TO 13.3 EFFECTIVE FULL POWER YEARS

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

COMMITMENTS ASSOCIATED WITH IPN-98-024

Comm. No.	Commitment Description	Due Date
IPN-98-024-01	Revise FSAR.	Next applicable FSAR update.

ATTACHMENT V TO IPN-98-024

PROPOSED EXEMPTION REQUEST FROM REQUIREMENTS OF 10 CFR 50.60

(Submitted as part of IPN-97-149 - Attached here for information only)

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

JUSTIFICATION FOR EXEMPTION FROM THE REQUIREMENTS OF 10 CFR 50.60

In accordance with 10 CFR 50.12(a), the Authority requests an exemption from the regulations of 10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Light-Water Nuclear Power Reactors For Normal Operation." This exemption would allow Indian Point 3 to determine the low temperature overpressurization system (LTOP) parameters using the American Society of Mechanical Engineers (ASME) Code Case N-514 in lieu of the safety margins required by 10 CFR 50, Appendix G. The Code Case limits the OPS curve to no greater than 110% of the pressure determined to satisfy Appendix G, paragraph G-2215 of ASME Code, Section XI, Division 1, further reduced to allow for static head due to elevation differences and dynamic head effect of the operation of the four reactor coolant pumps. In addition, the Code Case allows the OPS enable temperature to be the coolant temperature corresponding to a reactor vessel metal temperature less than RT_{NDT} + 50°F for the limiting beltline position, or 200°F, whichever is greater. (RT_{NDT} is defined by ASME Code Case N-514 to be the highest adjusted reference temperature for weld or base metal in the beltline region at a distance one-fourth of the vessel section thickness from the vessel inside surface, as determined by Regulatory Guide 1.99, Revision 2.)

10 CFR 50.60 states that all light-water nuclear power reactors must meet the fracture toughness and material surveillance program requirements for the reactor coolant boundary set forth in Appendices G and H of Part 50. Appendix G to 10 CFR 50 specifies fracture toughness requirements for ferritic materials of pressure retaining components of the reactor coolant pressure boundary to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests to which the pressure boundary may be subjected over its service lifetime. 10 CFR 50.60 specifically states that alternatives to the requirements described in Appendix G may be used when an exemption is granted by the NRC.

The provisions of ASME Code Case N-514 allow for a recalculation of LTOP parameters which results in greater operational flexibility than do the requirements of the version of ASME Section XI, Appendix G currently endorsed by 10 CFR 50.55(a). Code Case N-514 was approved by the ASME Code Case Committee and incorporated into Appendix G of Section XI of the ASME Code and published in the 1993 Addenda to Section XI. A draft revision to Regulatory Guide (RG) 1.147, published by the NRC in February 1997, endorses the use of this code case. However, since the approved versions of RG 1.147 and 10 CFR 50.55(a) do not currently endorse Code Case N-514 or the 1993 Addenda of the ASME Code, an exemption to 10 CFR 50.60 is requested.

Justification for Exemption

10 CFR 50.12(a) states that the NRC may grant exemptions from the requirements of the regulations contained in 10 CFR 50 provided that:

(1) the exemption is authorized by law;
(2) the exemption does not present an undue risk to the public health and safety;
(3) the exemption is consistent with the common defense and security; and
(4) special circumstances, as defined by 10 CFR 50.12(a)(2), are present.

(1) The requested exemption is authorized by law.

The NRC is authorized by law to grant this exemption. 10 CFR 50.60 states that the use of alternative methods to 10 CFR 50, Appendix G is acceptable when an exemption is granted by the NRC. Further, the NRC has granted similar exemptions to other nuclear facilities.

(2) The requested exemption does not present an undue risk to the public health and safety.

The requested exemption allows the determination of LTOP parameters using ASME Code Case N-514 in lieu of 10 CFR 50, Appendix G. The LTOP system controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary is not compromised. As stated in the Code Case, the LTOP systems shall be effective at coolant temperatures less than 200°F or at coolant temperatures corresponding to a reactor vessel metal temperature less than RT_{NDT} + 50°F (whichever is greater) and the maximum pressure in the vessel shall be limited to 110% of the pressure allowed by the NRC approved version of Appendix G of ASME Code, Section XI. The existing approved analysis supporting the heatup/cooldown and OPS limitations for Indian Point 3 (Reference 1) has been reviewed by Combustion Engineering (the author of the analysis) and confirmed to be applicable to the ASME code case (Reference 2). Therefore, the exemption does not present an undue risk to the public health and safety.

(3) The requested exemption will not endanger the common defense and security.

The common defense and security are not affected by this exemption request.

(4) Special circumstances are present which necessitate the request for an exemption.

10 CFR 50.12(a)(2) states that the NRC will not consider granting an exemption unless special circumstances are present. This exemption meets the special circumstances listed in 10 CFR 50.12(a)(2)(ii), 10 CFR 50.12(a)(2)(iii), and 10 CFR 50.12(a)(2)(v).

1. 10 CFR 50.12(a)(2)(ii) - Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule.

   The intent of the requirements of 10 CFR 50.60 is to provide protection of the reactor vessel against pressure transients at low temperatures. At Indian Point 3, the LTOP system provides this protection. ASME Code Case N-514 was approved by the ASME Code Case Committee as an acceptable alternative to the ASME Appendix G curves. This Code Case permits the recalculation of LTOP system parameters so as to limit the maximum pressure in the reactor vessel to 110% of the pressure determined to satisfy Appendix G of the ASME Code and to allow the LTOP enable temperature to equal 200°F or the coolant temperature corresponding to a reactor vessel metal temperature less than RT_{NDT} + 50°F (whichever is greater). Use of these parameters reduces the unnecessary actuation of the LTOP system due to normal pressure surges that occur during low temperature operation (i.e., surges resulting from starts of a reactor coolant pump or charging pump), while maintaining acceptable safety margins. Therefore, use of the Code Case meets the intent of 10 CFR 50.60.

2. 10 CFR 50.12(a)(2)(iii) - Compliance would result in undue hardship or other costs that are significantly in excess of those contemplated when the regulation was adopted, or that are significantly in excess of those incurred by others similarly situated.

   LTOP limits are designed to ensure that pressure transients at low temperatures do not cause fracture of the reactor vessel. As the vessel ages, the operating window created by the LTOP system is reduced because of the embrittling effects of neutron irradiation on the reactor vessel. This narrow operating window creates the potential of inadvertent actuation of the LTOP system. Inadvertent actuation of the LTOP system results in rapid pressure transients which place additional stresses on plant systems. This constitutes an unnecessary burden which can be alleviated by the use of ASME Code Case N-514. This Code Case permits the recalculation of the LTOP system setpoint with methodology which expands the operating window. The larger operating window does not significantly reduce the margin of safety and helps eliminate the unnecessary actuation of the LTOP system due to normal pressure

surges that occur during low temperature operation. For these reasons, compliance with 10 CFR 50.60 results in undue hardships which are in excess of those contemplated when the regulation was adopted.

3. 10 CFR 50.12(a)(2)(v) - The exemption would provide only temporary relief from the applicable regulation and the licensee or applicant has made good faith efforts to comply with the regulation.

   Indian Point 3 is currently in compliance with the regulations of 10 CFR 50.60. In order to maintain sufficient operating margin, an exemption is requested to use ASME Code Case N-514. Draft Regulatory Guide 1050, issued in February 1997, is a draft of Revision 12 to Regulatory Guide (RG) 1.147. This document proposes to endorse the use of Code Case N-514 in RG 1.147. Therefore, this exemption is needed only until the RG is amended to allow use of the Code Case.

Conclusion

The Authority concludes that continued compliance with the regulations of 10 CFR 50.60 results in an undue hardship. An exemption to allow the use of ASME Code Case N-514 to determine LTOP parameters would allow Indian Point 3 to meet the underlying purpose of the rule while gaining greater operational flexibility. This larger operating margin would help avoid unnecessary actuation of the LTOP system, and the associated rapid pressure transients, and still provide adequate margins against failure of the reactor pressure vessel.

References

1. ABB Combustion Engineering Report, "Final Report on Pressure-Temperature Limits for Indian Point 3 Nuclear Power Plant," dated July 1990.

2. ABB Combustion Engineering Calculation, "Indian Point Unit 3 Section XI LTOP Enable Temperatures for 13 & 15 EFPY (063-PENG-CALC-061, Revision 0)," dated August 14, 1997.