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FIGURE 3.1.A-1

Secondary Side Limitations for RCP Start With Secondary Side Hotter Than Primary, 16.2 EFPY

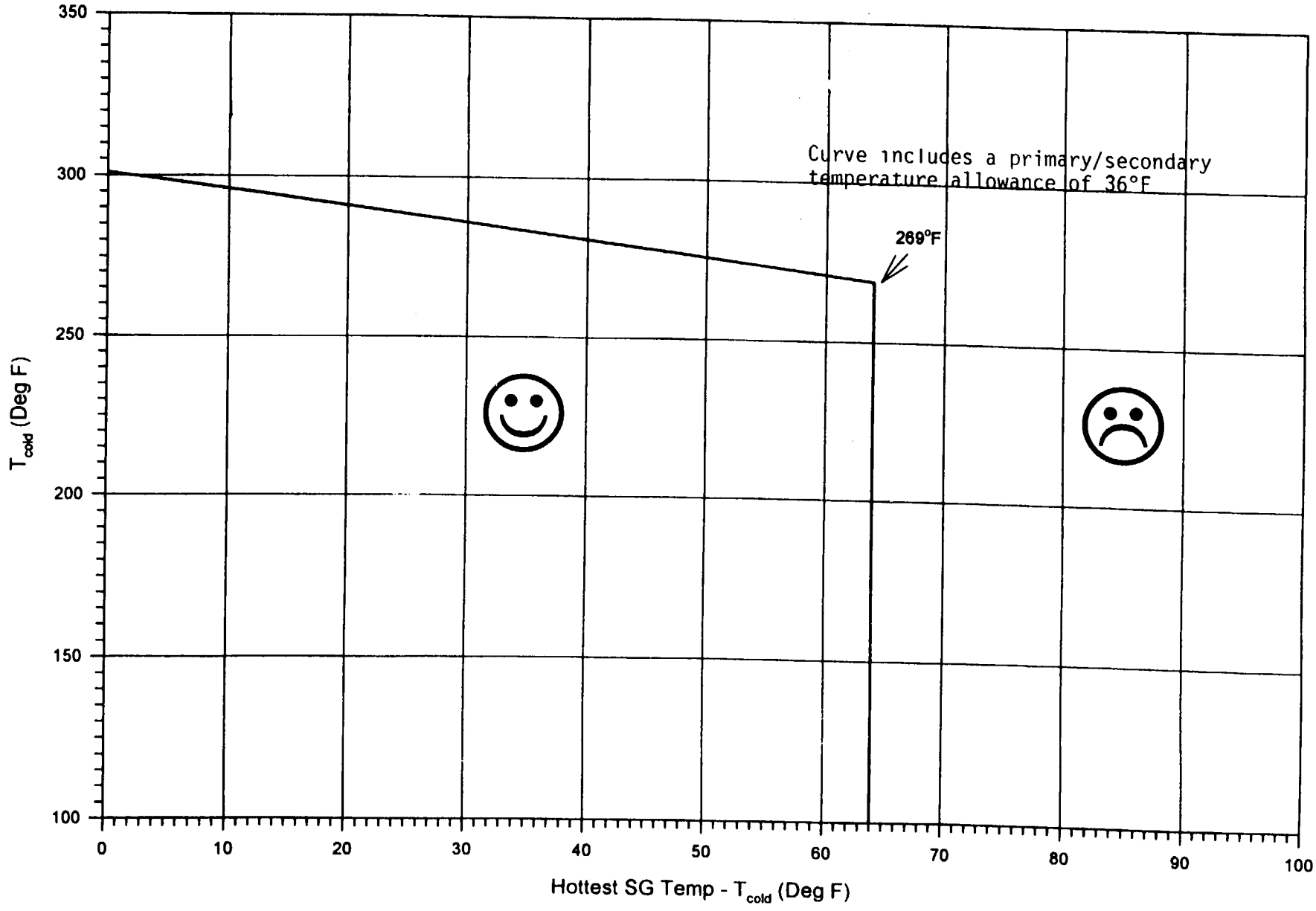

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FIGURE 3.1.A-2

Maximum Allowable Nominal PORV Setpoint for the Low Temperature Overpressure Protection System (OPS), 16.2 EFPY

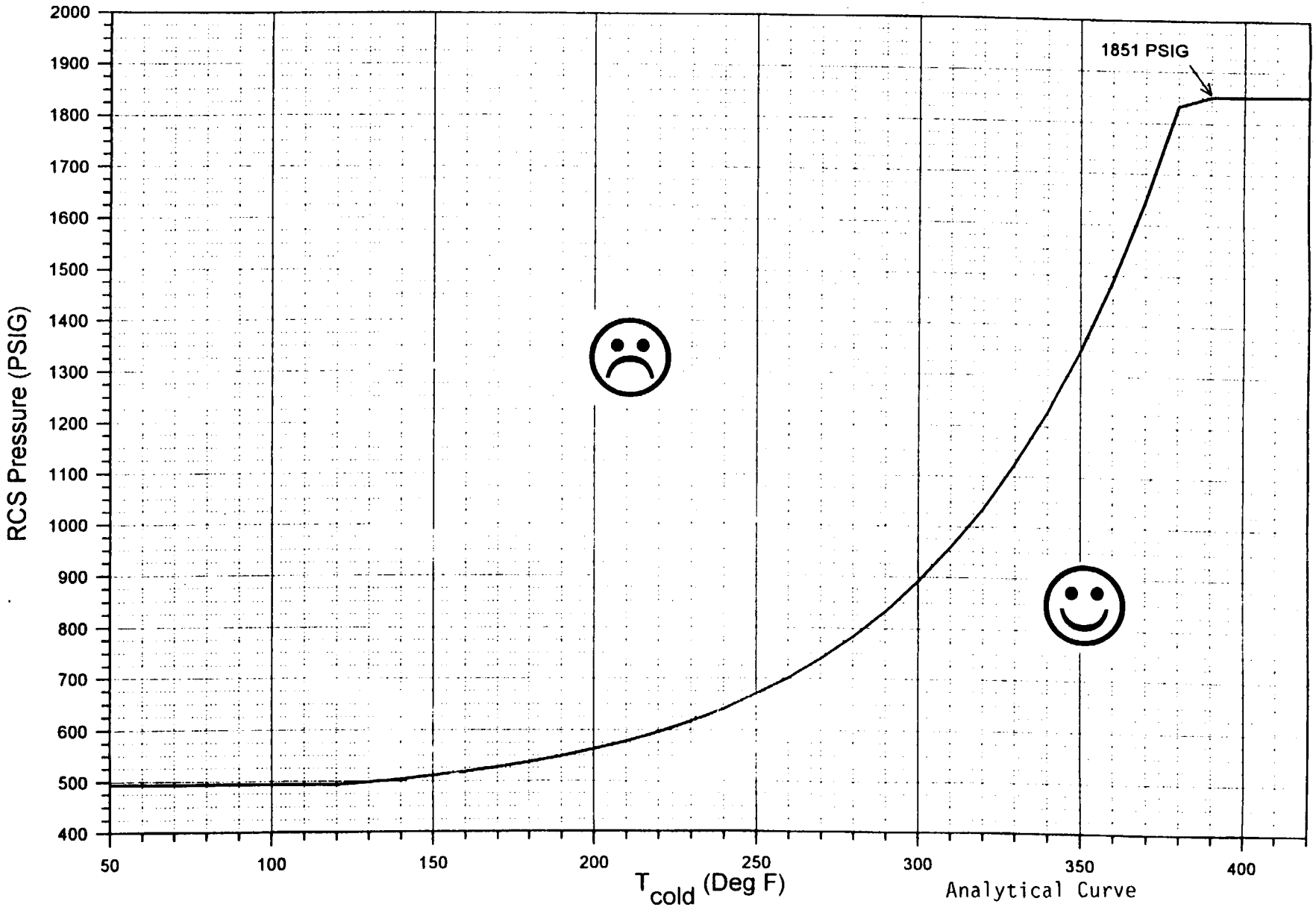

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FIGURE 3.1.A-5

Pressurizer Limitations for OPS Inoperable (Up to one charging pump capable of feeding RCS), 16.2 EFPY

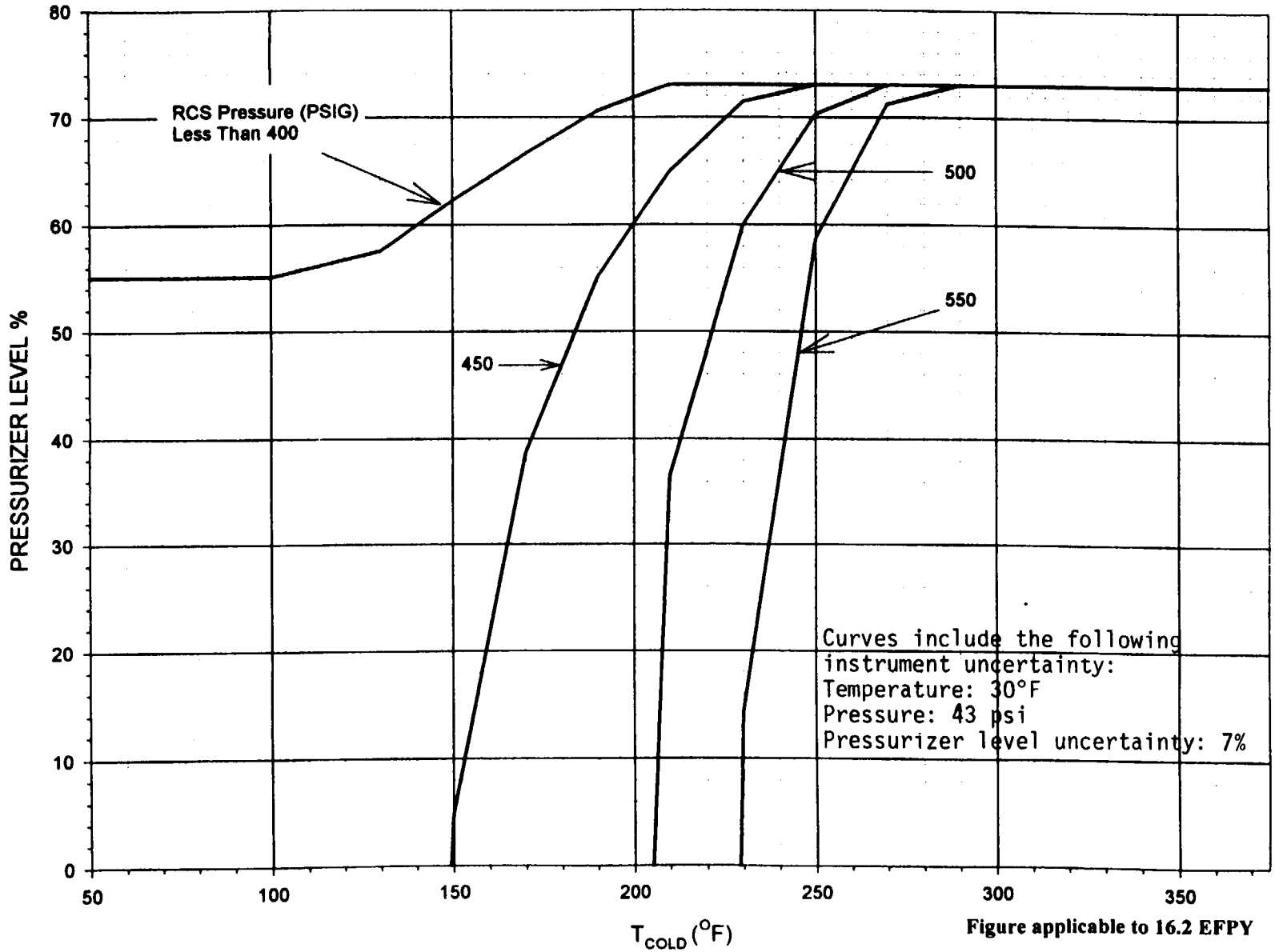

Figure applicable to 16.2 EFPY

Curves represent maximum allowable pressurizer level for the conditions defined.

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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), 16.2 EFPY

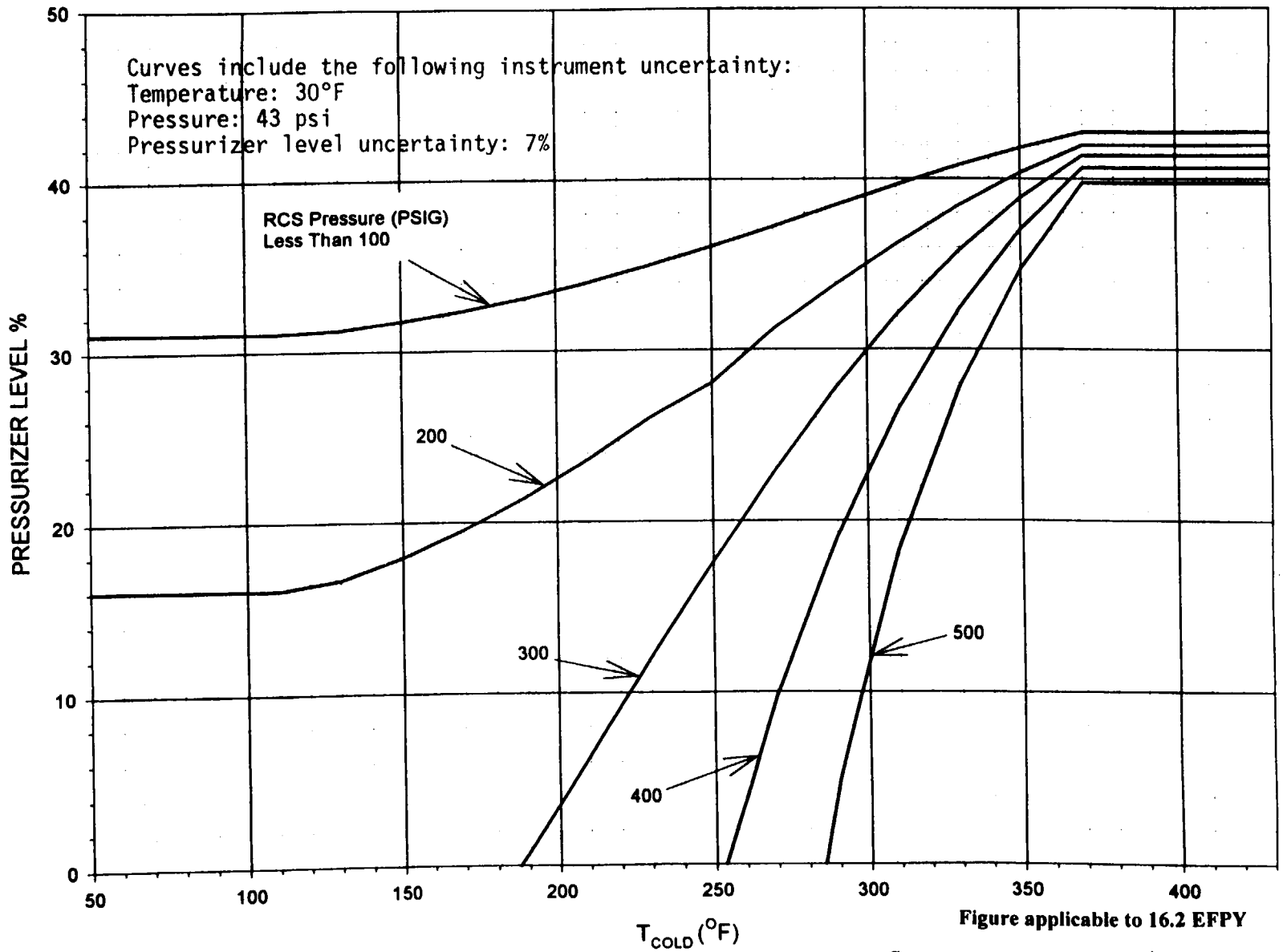

Figure applicable to 16.2 EFPY

Curves represent maximum allowable pressurizer level for the conditions defined.

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B. HEATUP AND COOLDOWN

Specifications

1. The reactor coolant temperature and pressure and system heatup and cooldown rates averaged over one hour (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1-1 and Figure 3.1-2 for the service period up to 16.2 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 16.2 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 16.2 EFPYs of operation is projected to be 214^0F at the 1/4 T and 172^0F 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 13.3 EFPYs.

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} at the end of 16.2 years of service life. The 16.2 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

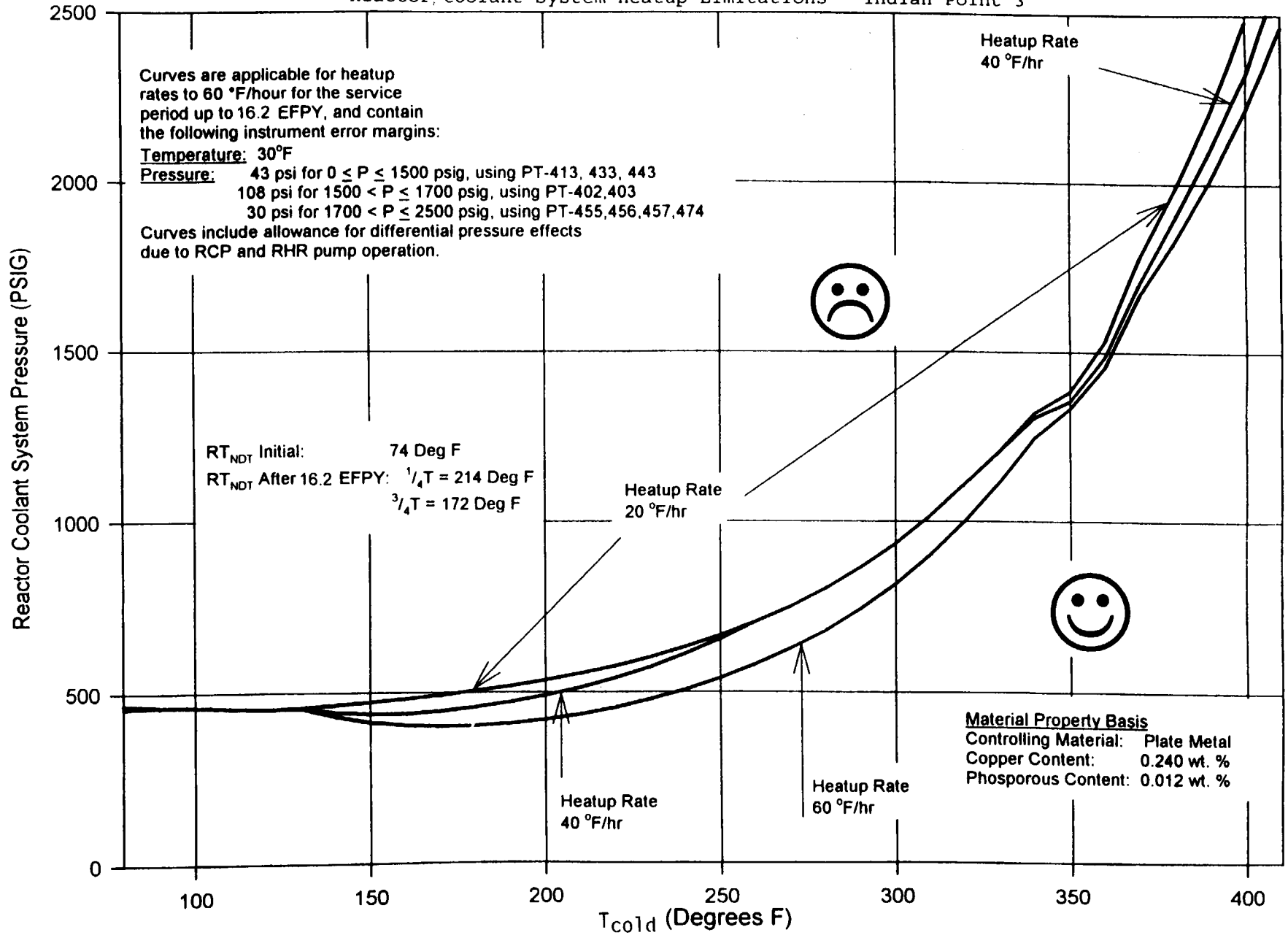

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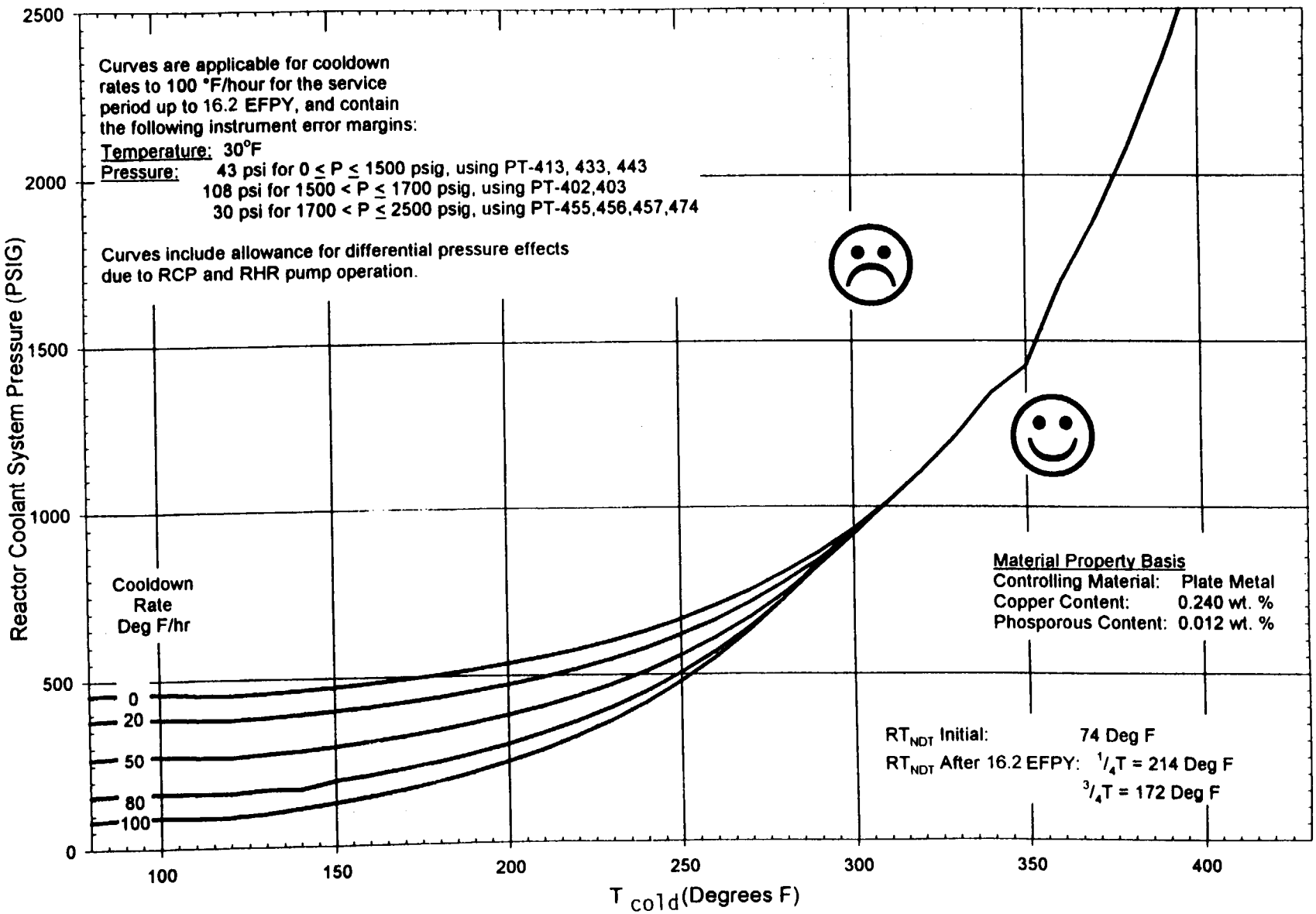
FIGURE 3.1-2

Reactor Coolant System Cooldown Limitations - Indian Point 3



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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 16.2 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 16.2 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 16.2 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.

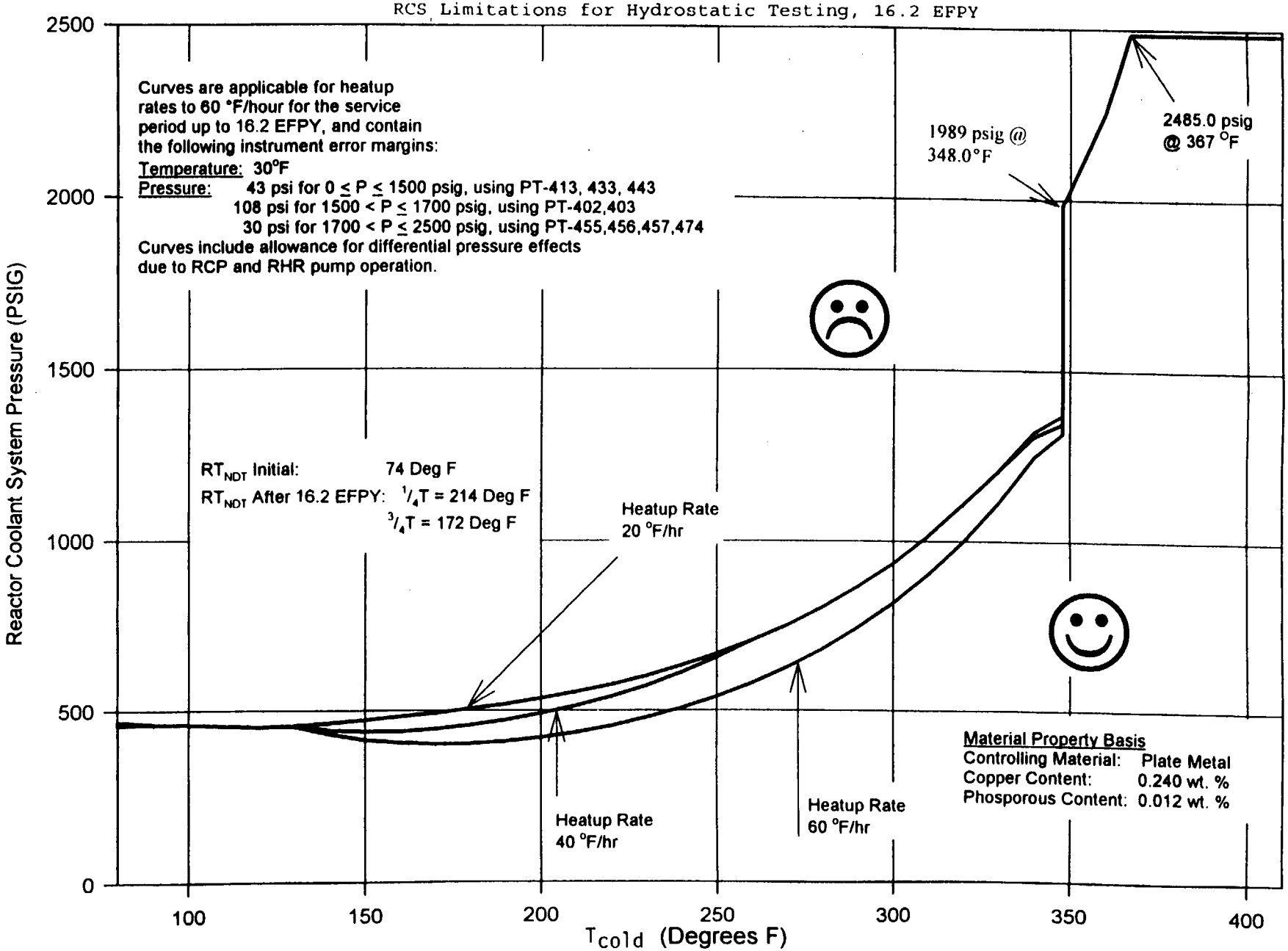
Reference

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, 16.2 EFPY



Amendment ~~28, 109, 121, 179,~~ 202