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

June 18, 1992

Docket No. 50-286

Mr. Ralph E. Beedle
Executive Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Beedle:

SUBJECT: ISSUANCE OF AMENDMENT FOR INDIAN POINT NUCLEAR GENERATING UNIT NO. 3 (TAC NO. M82062)

The Commission has issued the enclosed Amendment No. 121 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated October 28, 1991.

The amendment revises Technical Specifications Section 3.1.A (Operational Components), Section 3.1.B (Heatup and Cooldown), Section 3.3 (Engineered Safety Features), and Section 4.3 (Reactor Coolant System Integrity Testing). These sections have been revised to extend the reactor coolant system Pressure-Temperature (P/T) limits to 11.0 effective full power years (EFPY) of operation and to provide for the corresponding Overpressure Protection System (OPS) limits. In addition, Section 3.1 has been retyped in its entirety for format consistency and to correct typographical errors.

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

Sincerely,

N. F. Conicella

Nicola F. Conicella, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 121 to DPR-64
2. Safety Evaluation

cc w/enclosures:
See next page

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Mr. Ralph E. Beedle
Power Authority of the State
of New York

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DATED: June 18, 1992

AMENDMENT NO. 121 TO FACILITY OPERATING LICENSE NO. DPR-64-INDIAN POINT UNIT 3

Docket File
NRC & Local PDRs
PDI-1 Reading
S. Varga, 14/E/4
J. Calvo, 14/A/4
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OGC-WF
D. Hagan, 3302 MNBB
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

POWER AUTHORITY OF THE STATE OF NEW YORK

DOCKET NO. 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 121
License No. DPR-64

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

   A. The application for amendment by Power Authority of the State of New York (the licensee) dated, October 28, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-64 is hereby amended to read as follows:

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(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 121, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Capra

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 18, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 121

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Revise Appendix A as follows:

Remove Pages	Insert Pages
ii	ii
3.1-1 through 3.1-27	3.1-1 through 3.1-35
3.3-5a	3.3-5a
4.3-1	4.3-1
4.3-2	4.3-2
4.3-3	4.3-3

Section	Title	Page
1.14	$\bar{E}$-Average Disintegration Energy	1-5
1.15	Dose Equivalent I-131	1-5
1.16	Reportable Event	1-6
1.17	Core Operating Limits Report	1-6
2	Safety Limits and Limiting Safety System Settings	2.1-1
2.1	Safety Limits, Reactor Core	2.1-1
2.2	Safety Limit, Reactor Coolant System Pressure	2.2-1
2.3	Limiting Safety System Settings, Protective Instrumentation	2.3-1
3	Limiting Conditions for Operation	3.1-1
3.1	Reactor Coolant System	3.1-1
3.1.A	Operational Components	3.1-1
3.1.B	Heatup and Cooldown	3.1-17
3.1.C	Minimum Conditions for Criticality	3.1-25
3.1.D	Primary Coolant Activity	3.1-26
3.1.E	Maximum Reactor Coolant Oxygen, Chloride and Fluoride Concentration	3.1-29
3.1.F	Leakage of Reactor Coolant	3.1-31
3.1.G	Secondary Coolant Activity	3.1-35
3.2	Chemical and Volume Control System	3.2-1
3.3	Engineered Safety Features	3.3-1
3.3.A	Safety Injection and Residual Heat Removal Systems	3.3-1
3.3.B	Containment Cooling and Iodine Removal Systems	3.3-5a
3.3.C	Isolation Valve Seal Water System (IVSWS)	3.3-7
3.3.D	Weld Channel and Penetration Pressurization System (WC & PPS)	3.3-8
3.3.E	Component Cooling System	3.3-9
3.3.F	Service Water System/Ultimate Heat Sink	3.3-10
3.3.G	Containment Hydrogen Monitoring Systems	3.3-11
3.3.H	Control Room Ventilation System	3.3-11
3.3.I	Electric Hydrogen Recombiner System	3.3-12
3.4	Steam and Power Conversion System	3.4-1

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3. LIMITING CONDITIONS FOR OPERATION

For the cases where no exception time is specified for inoperable components, this time is assumed to be zero.

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the Reactor Coolant System; operational components; heatup; cooldown; criticality; activity; chemistry and leakage.

Objective

To specify those limiting conditions for operation of the Reactor Coolant System which must be met to ensure safe reactor operation.

Specification

A. OPERATIONAL COMPONENTS

1. Coolant Pumps

a. When a reduction is made in the boron concentration of the reactor coolant, at least one reactor coolant pump or one residual heat removal pump (connected to the Reactor Coolant System) shall be in operation.

b. (1) When the reactor coolant system T_{avg} is greater than 350°F and electrical power is available to the reactor coolant pumps, and as permitted during special plant evolutions, at least one reactor coolant pump shall be in operation. All reactor coolant pumps may be de-energized 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.

(2) When the reactor is subcritical and reactor coolant system T_{avg} is greater than 350°F, control bank withdrawal shall be prohibited unless four reactor coolant pumps are operating.

c. When the reactor coolant system T_{avg} is greater than 200°F and less than 350°F, and as permitted during special plant evolutions, at least one reactor coolant pump or one residual heat removal pump (connected to the Reactor Coolant System) shall be in operation. All reactor coolant pumps may be de-energized with RHR not in 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.

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, 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 percent, and T_{cold} is as per Figure 3.1.A-1;

Or

(3) The 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 percent, and the RCS pressure does not exceed that given by Curve I on Fig. 3.1.A-2;

Or

(4) The OPS is <u>inoperable</u>, 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.

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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.

2. Safety Valves

a. At least one pressurizer code safety valve shall be operable, or an opening greater than or equal to the size of one code safety valve flange to allow for pressure relief, whenever the reactor head is on the vessel except for hydrostatically testing the RCS in accordance with Section XI of the ASME Boiler and Pressure Vessel Code.

b. All pressurizer code safety valves shall be operable whenever the reactor is above the cold shutdown condition except during reactor coolant system hydrostatic tests and/or safety valve settings.

c. The pressurizer code safety valve lift setting shall be set at 2485 psig with ±1% allowance for error.

3. Pressurizer Heaters

Whenever the reactor is above the hot shutdown condition, the pressurizer shall be operable with at least 150 kw of pressurizer heaters.

a. With less than 150 kw of pressurizer heaters operable, restore the required inoperable heaters within 72 hours or be in at least hot shutdown within an additional 6 hours.

4. Power Operated Relief Valves

Whenever the reactor coolant system is above 400°F, the power operated relief valves (PORVs) shall be operable or their associated block valves closed.

a. If the block valve is closed because of an inoperable PORV, the control power for the block valve must be removed.

b. If the above conditions cannot be satisfied within 1 hour, be in at least hot shutdown within 6 hours and in cold shutdown within the following 30 hours.

5. Power Operated Relief Block Valves

Whenever the reactor coolant system is above 400°F, the motor operated block valves shall be operable or closed.

a. If the block valve is inoperable, the control power is to be removed.

b. If the above conditions cannot be satisfied within 1 hour be in at least hot shutdown within the following 30 hours.

6. Reactor Coolant System T_{avg}

During steady state operation, the maximum indicated T_{avg} shall not exceed 578.3°F.

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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.

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:

(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 Specifications 3.1.A.1.h(3) and (4) and maintained above 370°F;

Or

(3) Restrict pressurizer level as per the curves referenced below:

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.

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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.

c. 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.

Basis

When the boron concentration of the Reactor Coolant System is to be reduced the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the primary system volume in approximately one half hour. The pressurizer is of no concern because of the low pressurizer volume and because the pressurizer boron concentration will be higher than that of the rest of the reactor coolant.

Heat transfer analyses show that reactor heat equivalent to 10% of rated power (P-7) can be removed with natural circulation only (1); hence, the requirement for one operating RCP above 350°F and two operating RCP's above 2% rated power provides a substantial safety factor. In addition, a single RCP or RHR pump (connected to the RCS) provides sufficient heat removal capability for removing decay heat.

The restriction on control bank withdrawal with less than four reactor coolant pumps operating when the reactor is subcritical and RCS T_{avg} is greater than 350°F is necessary to conform with the assumptions used in the transient analyses for the uncontrolled control rod withdrawal event from subcritical condition. The FSAR safety analysis for uncontrolled control rod assembly withdrawal from a subcritical condition assumes all four reactor coolant pumps to be operating within the temperature range of concern. Using this assumption the DNB design basis is satisfied for the combination of the two banks of the maximum combined worth withdrawn at maximum speed. Since there is no mechanism by which the control rods can be automatically withdrawn due to a control system error when T_{avg} is between 350°F and the no-load temperature, such an event can only be initiated as a result of human error during rod manipulation. Prohibiting control bank withdrawal with less than four RCPs operating provides assurance that the plant is operated within the accident analysis assumptions.

The reactor shall not be operated at power levels above 10% rated power with less than four (4) reactor coolant loops in operation until safety analyses for less than four loop operation have been submitted by the licensee and approval for less than four loop operation at power levels above 10% rated power has been granted by the Commission. (See license condition 2.C. (3))

Each of the pressurizer code safety valves is designed to relieve 420,000 lbs. per hr. of saturated steam at the valve set point.

If no residual heat were removed by the Residual Heat Removal System the amount of steam which could be generated at safety valve relief pressure would be less than half the capacity of a single valve. One valve therefore provides adequate protection for overpressurization.

The combined capacity of the three pressurizer safety valves is greater than the maximum surge rate resulting from complete loss of load (2) without a direct reactor trip or any other control.

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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.

The limit on maximum indicated T_{avg} provides assurance that RCS temperatures are maintained within the normal steady state envelope of operation assumed in the safety analyses performed to support the Vantage 5 fuel reloads with asymmetric tube plugging among steam generators. A maximum full power T_{cold} of 547.9°F (including control deadband and measurement uncertainties) was assumed in these safety analyses. Restricting the maximum indicated T_{avg} to 578.3°F assures that a T_{cold} of 547.9°F is not exceeded at a measured flow of 332,240 gpm when considering asymmetric tube plugging among steam generators.

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. "Arming" means that the motor operated 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 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. 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) 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 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 charging pumps) as well as the Heat Input initiating event are described in References (3), (4) and (5). (Also refer to Specification 3.3.A.8. Safety Injection and Residual Heat Removal Systems). The OPS need not be operable when the RCS temperature is less than 332°F if the RCS is depressurized and vented with an equivalent opening of at least 2.00 square inches. This opening is adequate to relieve the worst case analyzed.

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

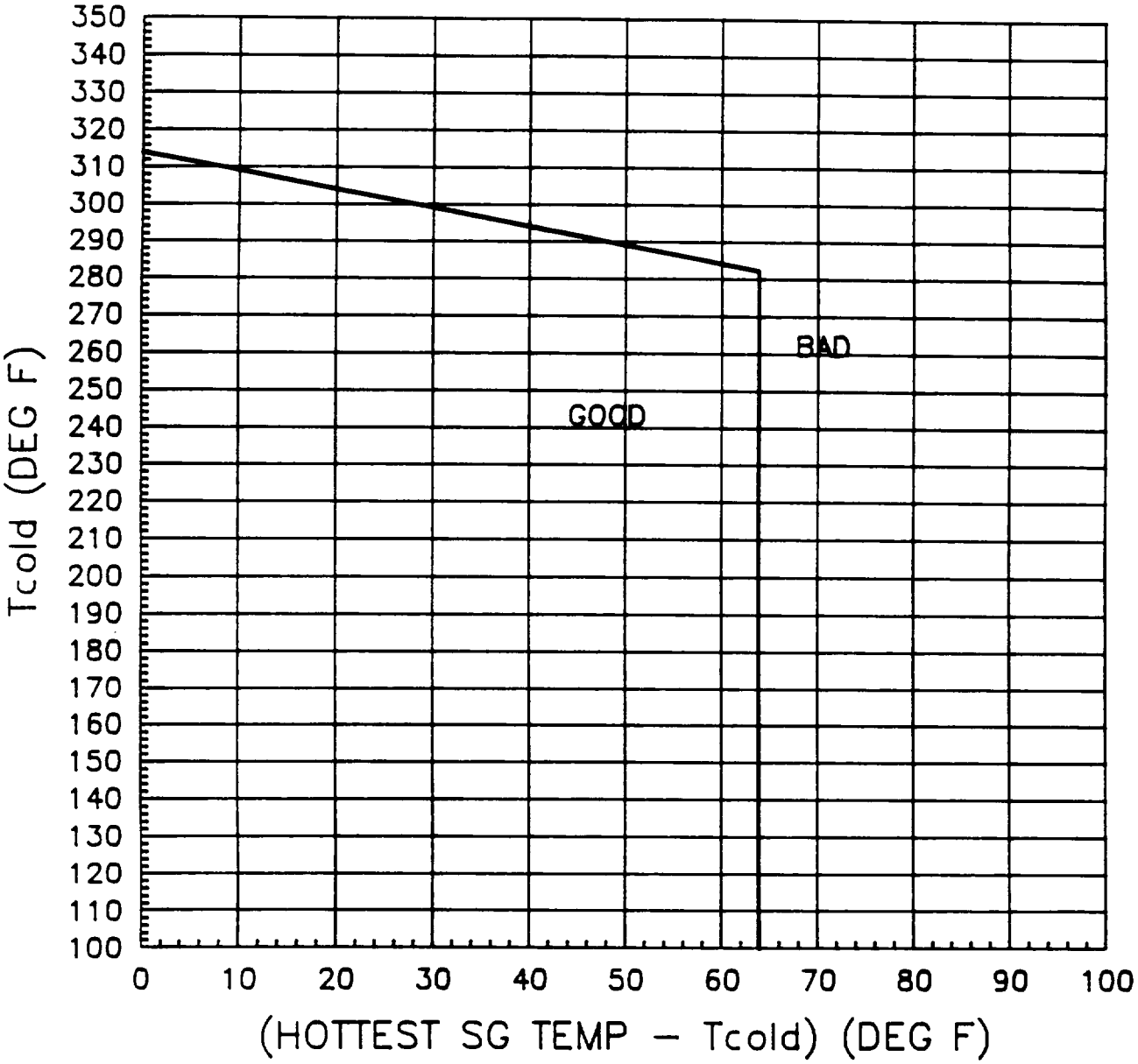
Upon OPS inoperability, the RCS may be heated above 370°F. This temperature is that value for which the RCS heatup and cooldown curves (Figures 3.1-1 and 3.1-2) permit pressurization to the setting of the pressurizer safety valves. Accordingly, with an inoperable OPS and an RCS temperature 370°F, the 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) which requires the presence of a pressurizer bubble to preclude RCS overpressurization during inadvertent mass inputs. Specification 3.1.A.8.b(3) also places restrictions on the number of SI pumps capable of feeding the RCS (see Specification 3.3.A.8). An SI 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" specimens (Reference 6).
Amendment No. ~~97~~, ~~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.

FIGURE 3.1.A-1

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



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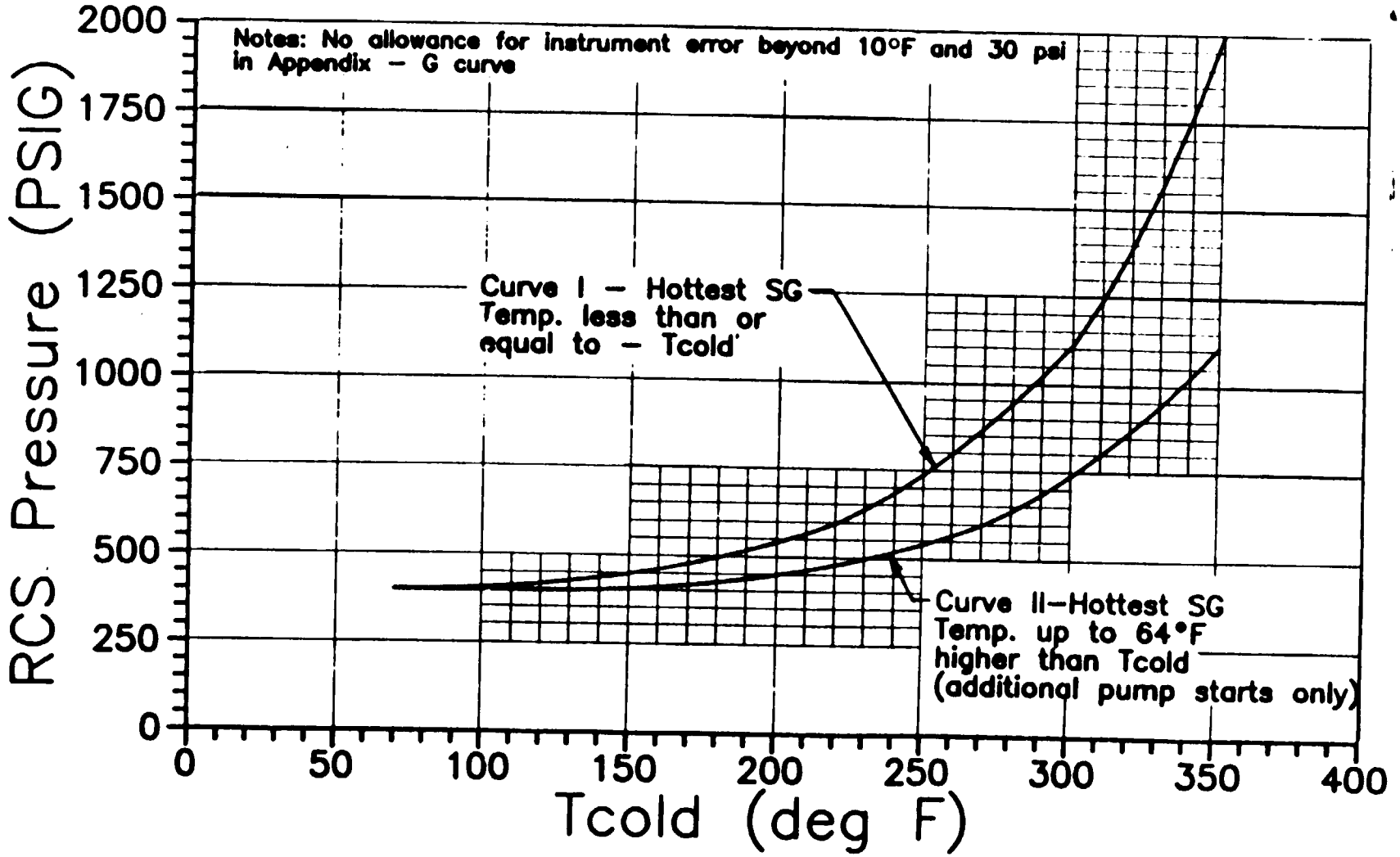


Figure 3.1.A–2 Figure applicable to 11 EFPY

Maximum Permissible RCS Pressure for RCP Start with OPS Inoperable

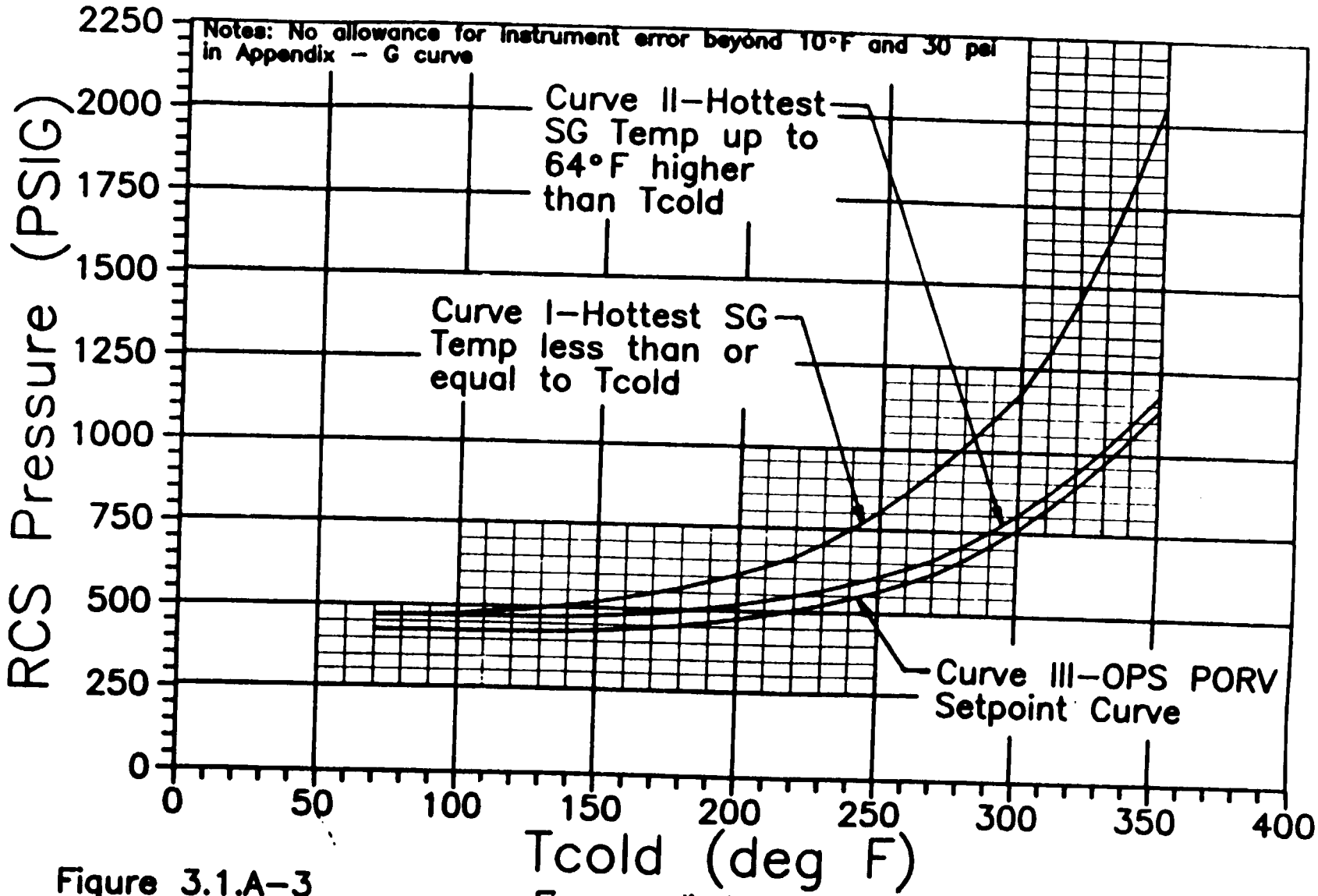


Figure 3.1.A–3

Figure applicable to 11 EFPY

RCS Pressure Limits for Low Temperature Operation

Amendment No. 67, 121

Notes: No allowance for instrument error beyond 10°F and 30 psi in Appendix-G, and 5% PRZR LEVEL (Meas. Unc.)
Figure applicable to 11 EFPY
P_{RCS} = Rcs Pressure (psig)

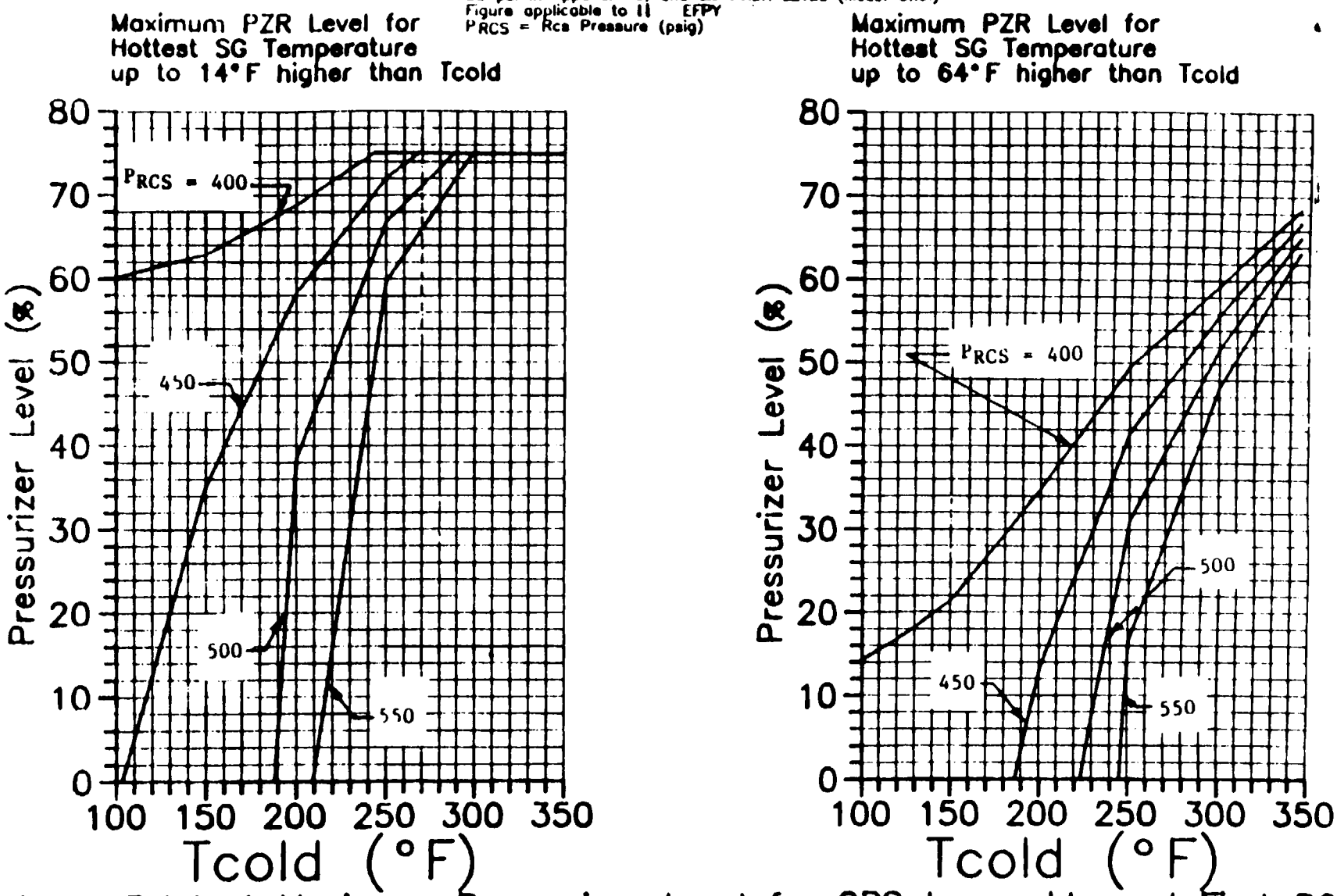


Figure 3.1.A-4 Maximum Pressurizer Level for OPS Inoperable and First RCP Start (Steam Generator Temperature Greater than Tcold)

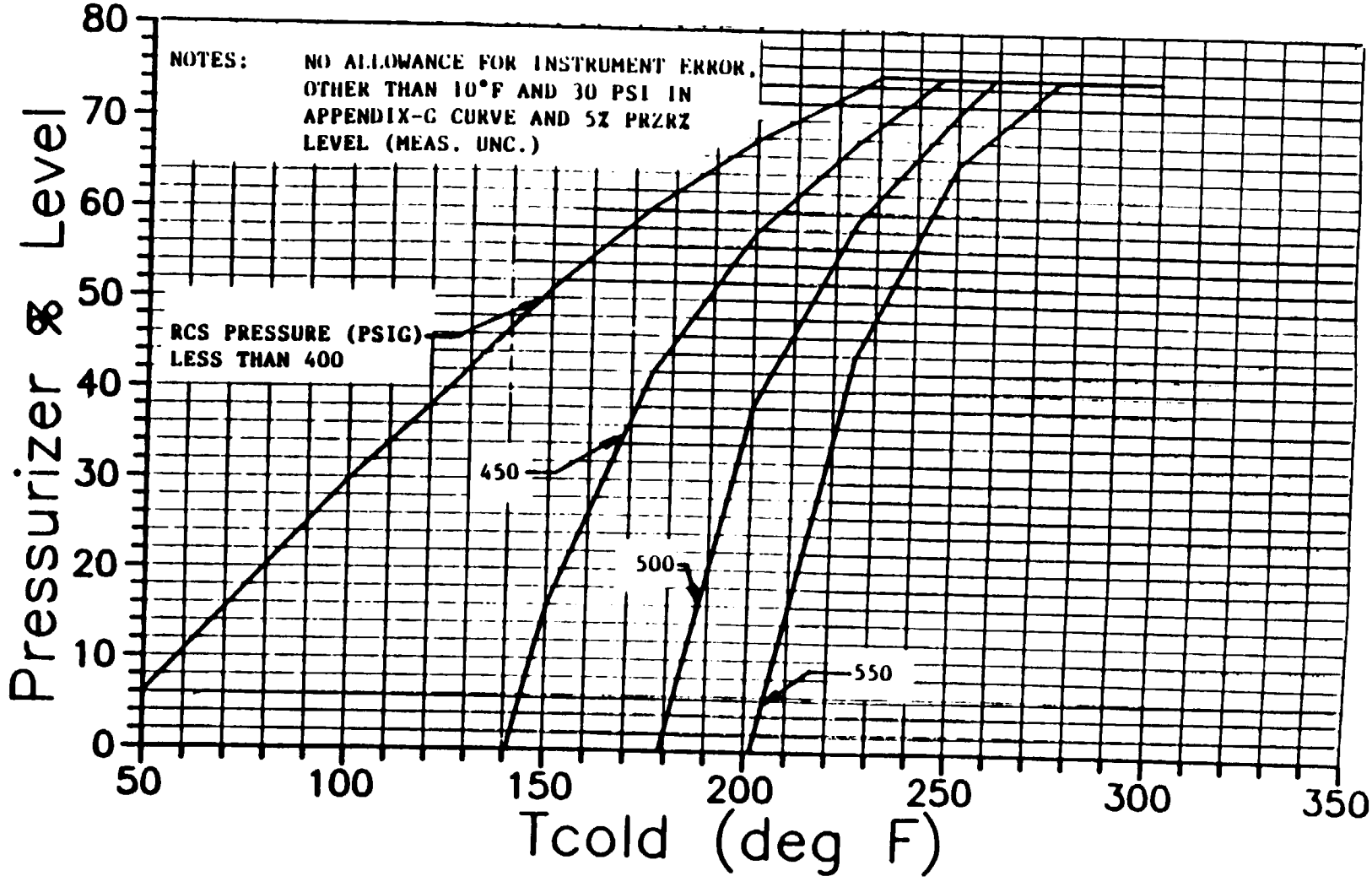


Figure 3.1.A–5

Figure applicable to 11 EFPY

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

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

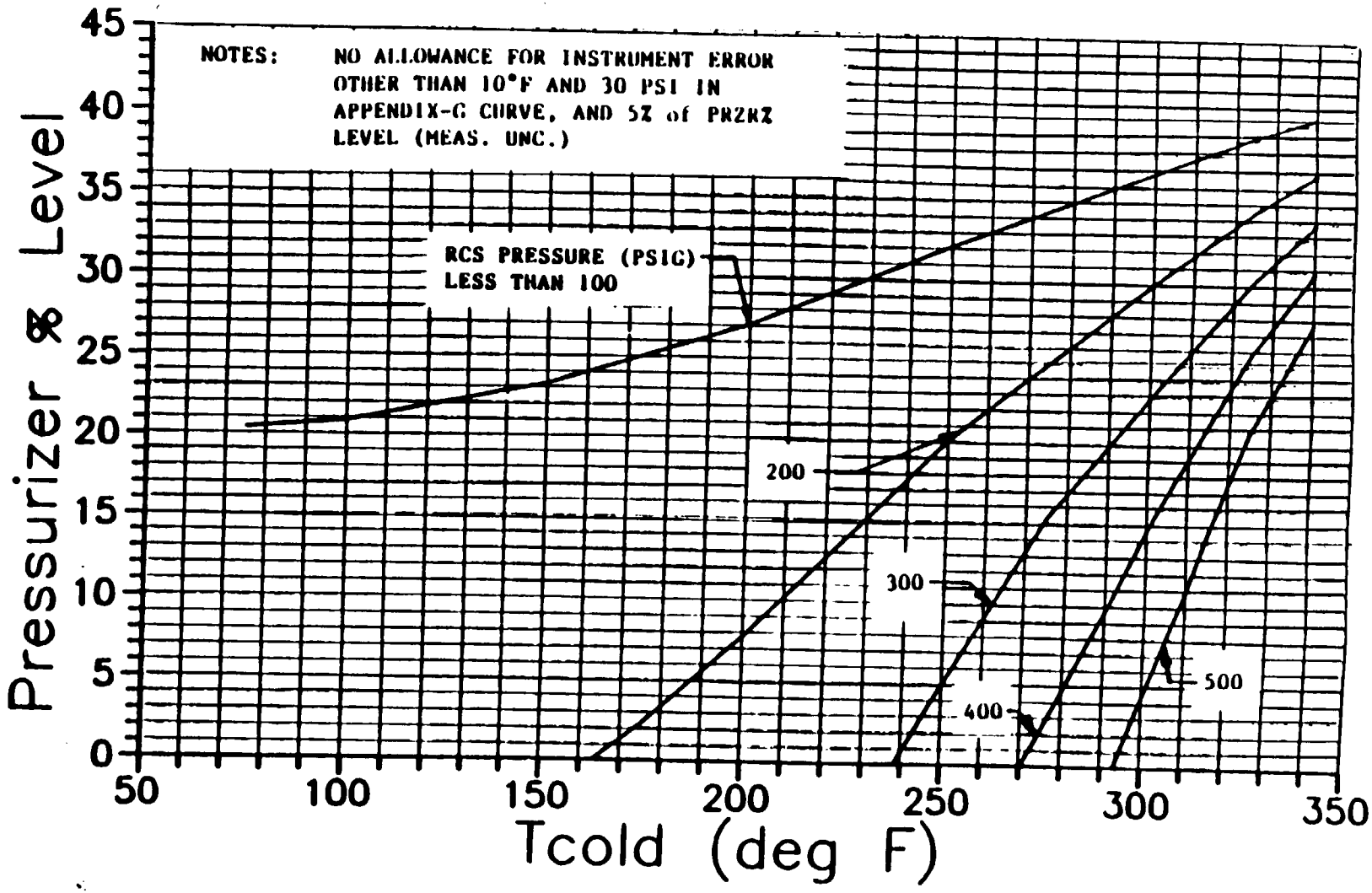


Figure 3.1.A–6

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. 67, 100, 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 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 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 in RT_{NDT} after 11.00 EFPYs of operation is projected to be 202°F at the 1/4 T and 163°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 EFPYs.

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} at the end of 11.00 years of service life. The 11.00 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].

Heatup and Cooldown Curves

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Non Mandatory Appendix G in Section III of the ASME Boiler and Pressure Vessel Code and discussed in detail in WCAP-7924. [2]

The approach specifies that the allowable total stress intensity factor (K_I) at any time during heatup or cooldown cannot be greater than that shown on the K_{IR} curve [1] for the metal temperature at that time. Furthermore, the approach applies an explicit safety factor of 2.0 on the stress intensity factor induced by pressure gradients. Thus, the governing equation for the heatup-cooldown analysis is:

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

where:

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

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

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

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

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

The second portion of the heatup analysis concerns the calculation of pressure temperature limitations for the case in which the 3/4 T location becomes the controlling factor. Unlike the situation at the 1/4 T location, at the 3/4 T position (i.e., the tip of the 1/4 T deep O.D. flaw) the thermal gradients established during heatup produce stresses which are tensile in nature; and, thus, tend to reinforce the pressure stresses present. These thermal stresses are, of course, dependent on both the rate of heatup and the time (or water temperature) along the heatup ramp. Furthermore, since the thermal stresses at 3/4 T are tensile and increase with increasing heatup rate, a lower bound curve similar to that described in the preceding paragraph cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

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

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

The cooldown analysis proceeds in the same fashion as that for heatup, with the exception that the controlling location is always at 1/4 T. The thermal gradients induced during cooldown tend to produce tensile stresses at the 1/4 T location and compressive stresses at the 3/4 T position. Thus, the ID flaw is clearly the worst case.

As in the case of heatup, allowable pressure temperature relations are generated for both steady state and finite cooldown rate situations. Composite limit curves are then constructed for each cooldown rate of interest. Again adjustments are made to account for pressure and temperature instrumentation error.

The use of the composite curve in the cooldown analysis is necessary because system control is based on a measurement of reactor coolant temperature, whereas the limiting pressure is calculated using the material temperature at the tip of the assumed reference flaw. During cooldown, the 1/4 T vessel location is at a higher temperature than the fluid adjacent to the vessel I.D. This condition is, of course, not true for the steady-state situation. It follows that the ΔT induced during cooldown results in a calculated higher allowable K_{IR} for finite cooldown rates than for steady state under certain conditions.

Because operation control is on coolant temperature, and cooldown rate may vary during the cooldown transient, the limit curves shown in Figure 3.1-2 represent a composite curve consisting of the more conservative values calculated for steady state and the specific cooling rate shown.

Details of these calculations are provided in WCAP-7924 [2]. Information on the specific calculations used to develop the current heatup-cooldown curves can be found in Reference 9.

- **Pressurizer Limits**

Although the pressurizer operates at temperature ranges above those for which there is reason for concern about brittle fracture, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, 1965 Edition and associated Code Addenda through the Summer 1966 Addendum.

REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section III, 1972 Summer Addenda.

2. WCAP-7924, "Basis for Heatup and Cooldown Limit Curves", W. S. Hazelton, S. L. Anderson, S. E. Yanichko, July 1972.

3. FSAR Volume 5, Response to Question Q4.2.

4. Intentionally deleted.

5. ASTM E185-70, Surveillance Tests on Structural Materials in Nuclear Reactors.

6. ASME Boiler and Pressure Vessel Code, Section III, Summer 1965.

7. WCAP-9491, "Analysis of Capsule T from the Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program", J. A. Davidson, S. L. Anderson, W. T. Kaiser, April 1979.

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

10. 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.

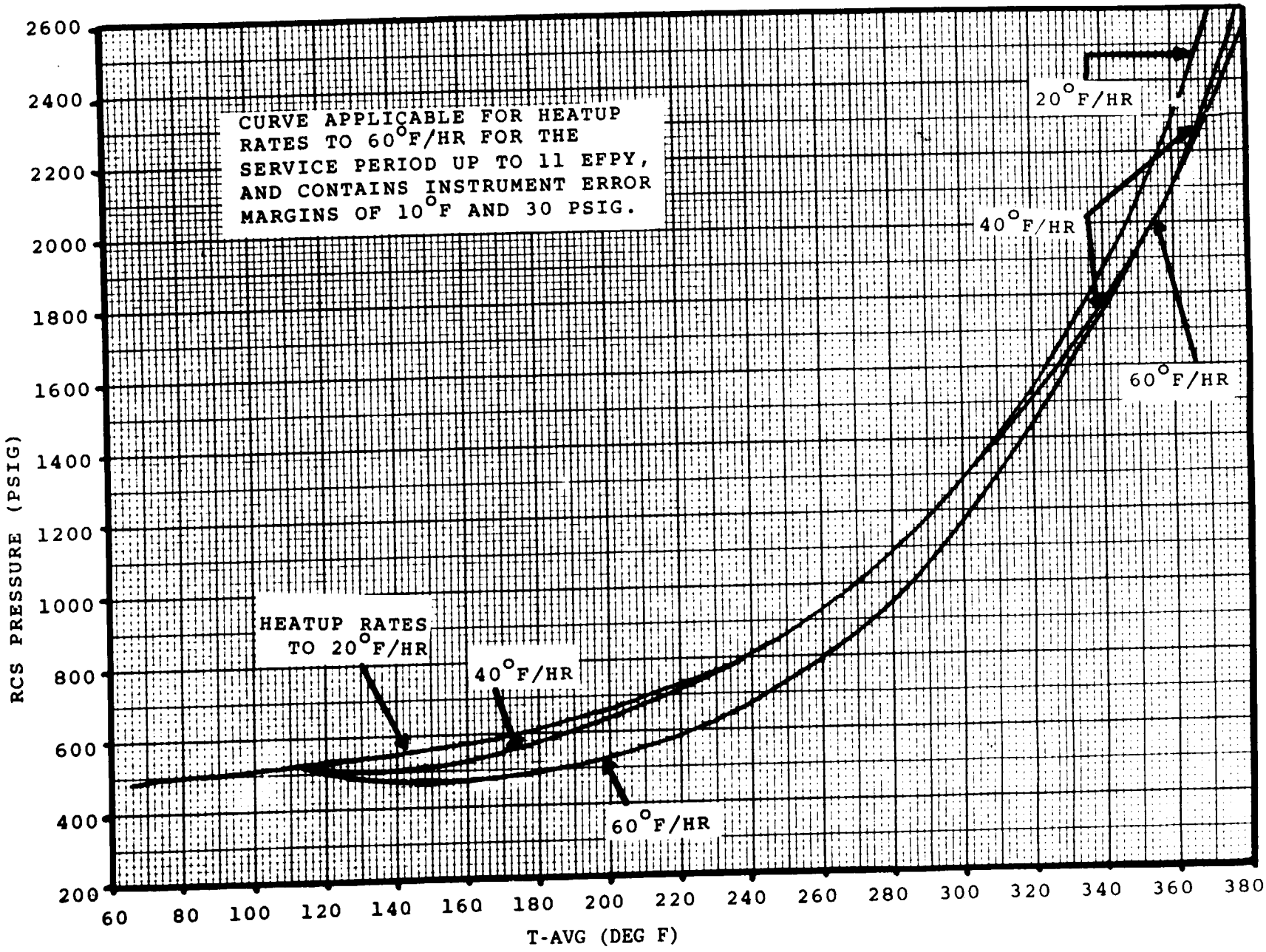
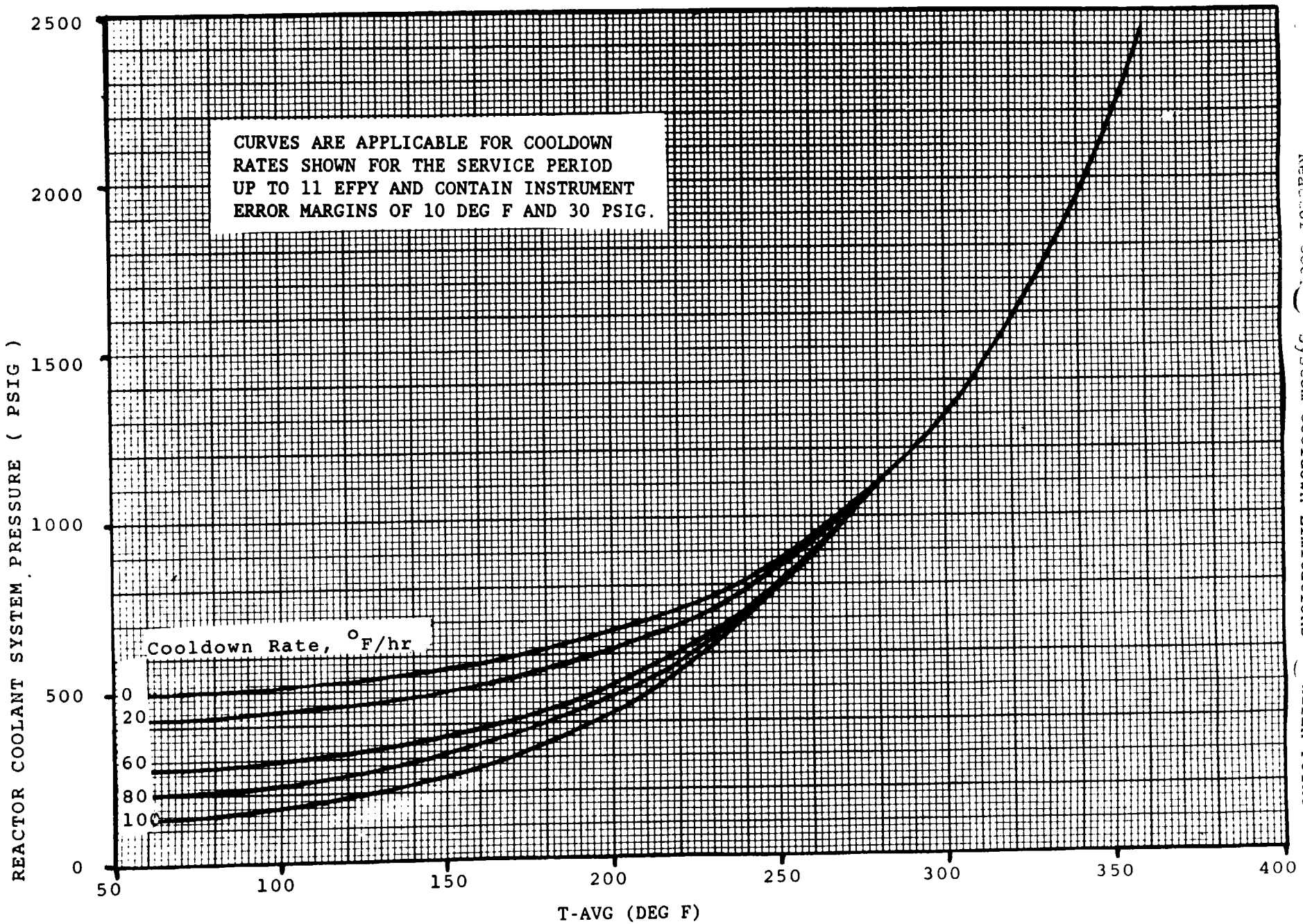


FIGURE 3.1-1
Reactor Coolant System Heatup Limitations - Indian Point 3

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: 1/4T = 202°F
3/4T = 163°F



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Phosphorous Content:	0.012 wt. %

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

C. MINIMUM CONDITIONS FOR CRITICALITY

1. Except during low power physics test, the reactor shall not be made critical at any temperature above which the moderator temperature coefficient is positive.

2. This section intentionally deleted.

3. At all times during critical operation, T_{avg} should be no lower than 450°F.

4. The reactor shall be maintained subcritical by at least 1% $\frac{\Delta k}{k}$ until normal water level is established in the pressurizer.

Basis

During the early part of the initial fuel cycle, the moderator temperature coefficient is calculated to be slightly positive at coolant temperatures below the power operating range. [1] [2] The moderator coefficient at low temperatures will be most positive at the beginning of life of the fuel cycle, when the boron concentration in the coolant is the greatest. Later in the life of the fuel cycle, the boron concentration in the coolant will be lower and the moderator coefficient will be either less positive or will be negative. At all times, the moderator coefficient is negative in the power operating range. [1] [2] Suitable physics measurements of moderator coefficient of reactivity will be made as part of the startup program to verify analytic predictions.

The requirement that the reactor is not to be made critical when the moderator coefficient is positive has been imposed to prevent any unexpected power excursion during normal operations as a result of an increase in moderator temperature. This requirement is waived during low power physics tests to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special operating precautions will be taken.

The requirement that the reactor is not to be made critical except when T_{avg} is $\geq$ 450°F provides increased assurance that an overpressure event will not occur whenever the reactor vessel is in the nil-ductility temperature range. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.

The requirement for bubble formation in the pressurizer when the reactor has passed the threshold of 1% subcriticality will assure that the reactor coolant not be solid when criticality is achieved.

References:

1. FSAR Table 3.2.1-1

2. FSAR Figure 3.2.1-9

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D. Primary Coolant Activity

Specification

1. Whenever the reactor is critical or the average reactor coolant temperature is >500°F, the specific activity of the primary coolant shall be limited to:

   a. ≤1.0 μCi/cc Dose Equivalent I-131,

   and

   b. ≤100/Ē μCi/cc for all noble gases with half-lives greater than 10 minutes.

2. If the specific activity of the primary coolant is >1.0 μCi/cc Dose Equivalent I-131 but within the allowable limit (below and to the left of the line) shown on Figure 3.1-3, operation may continue for up to 48 hours.

3. If the specific acitivity of the primary coolant is >1.0 μCi/cc Dose Equivalent I-131 for more than 48 hours during one continuous time interval or exceeds the limit line shown on Figure 3.1-3, the reactor shall be immediately brought to the hot shutdown condition with T_{avg} ≤500°F utilizing normal operating procedures.

4. If the specific activity of the primary coolant is >100/Ē μCi/cc for all noble gases with half-lives greater than 10 minutes, the reactor shall be immediately brought to the hot shutdown condition with T_{avg} ≤500°F utilizing normal operating procedures.

Bases

The limitations on the specific activity of the primary coolant insure that the resulting 2-hour doses at the site boundary will not exceed 1.5 rem to the thyroid and 0.5 rem whole body following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM and a resultant loss of offsite power. Accident meteorological conditions (5% X/Q) are assumed to exist.

The action statement permitting Power Operation to continue for limited time periods with the primary coolant's specific activity >1.0 μCi/cc Dose Equivalent I-131, but within the allowable limit shown on Figure 3.1-3, accommodates possible iodine spiking phenomenon which may occur following changes in Thermal Power.

- Reducing T_{avg} to <500°F prevents the release of activity, should a steam generator tube rupture, since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Increased surveillance for performing isotopic analyses for iodine is required whenever the Dose Equivalent I-131 exceeds 1.0 μCi/cc and following a significant change in power level to monitor possible iodine spiking phenomenon.

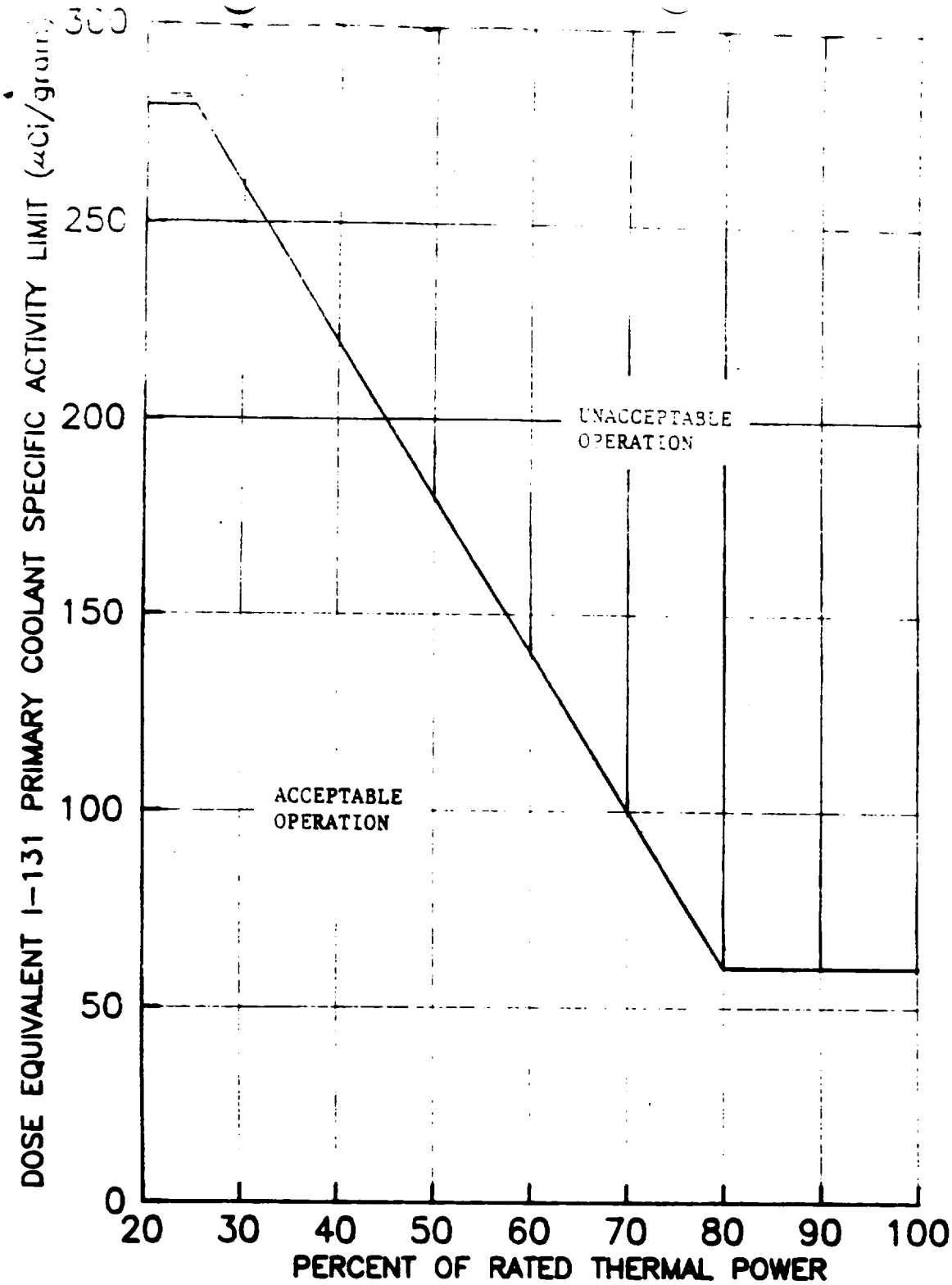

Figure 3.1-3

DOSE EQUIVALENT I-131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity > 1.0μCi/gram Dose Equivalent I-131

E. MAXIMUM REACTOR COOLANT OXYGEN, CHLORIDE AND FLUORIDE CONCENTRATION

Specification

1. Concentrations of contaminants in the reactor shall not exceed the following limits when the reactor coolant is above 250°F:

Contaminant	Normal Steady-State Operation (PPM)	Transient Not To Exceed 24 Hours (PPM)
a. Oxygen	0.10	1.00
b. Chloride	0.15	1.50
c. Fluoride	0.15	1.50

2. If any of the normal steady-state operating limits as specified in 3.1.E.1, above, are exceeded, or if it is anticipated that they may be exceeded, corrective action shall be taken immediately.

3. If the concentrations of any of the contaminants cannot be controlled within the limits of Specification 3.1.E.1, namely, steady-state limit not restored within 24 hours or transient limit exceeded, the reactor shall be brought to the cold shutdown condition, utilizing normal operating procedures, and the cause of the out-of-specification operation ascertained and corrected. The reactor may then be restarted and operation resumed if the maximum concentration of any of the contaminants did not exceed the permitted transient values. Otherwise, a safety review is required before startup.

4. Concentrations of contaminants in the reactor coolant shall not exceed the following maximum limits when the reactor coolant temperature is below 250°F:

Contaminant	Normal Concentration (PPM)	Transient Not To Exceed 48 Hours (PPM)
a. Oxygen	Saturated	Saturated
b. Chloride	0.15	1.50
c. Fluoride	0.15	1.50

If the limits above are exceeded, namely, normal concentration limits not restored within 48 hours or the transient limits exceeded, the reactor shall be immediately brought to the cold shutdown condition and the cause of the out-of-specification condition ascertained and corrected.

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5. For the purposes of correcting the contaminant concentrations to meet Specifications 3.1.E.1 and 3.1.E.4 above, increase in coolant temperature consistent with operation of reactor coolant pumps for a short period of time to assure mixing of the coolant shall be permitted. This increase in temperature to assure mixing shall in no case cause the coolant temperature to exceed 250°F.

Basis

By maintaining the oxygen, chloride and fluoride concentrations in the reactor coolant below the limits as specified in 3.1.E.1 and 3.1.E.4, the integrity of the reactor coolant system is assured against stress corrosion cracking under all operating conditions. [1]

If these limits are exceeded, measures can be taken to correct the condition, e.g., replacement of ion exchange resin or adjustment of the hydrogen concentration in the volume control tank [2] during power operation. Because of the time dependent nature of any adverse effects arising from oxygen, chloride, and fluoride concentration in excess of the limits, it is unnecessary to shut down immediately, as the condition can be corrected. Thus, the periods of either 24 hours or 48 hours for corrective action to restore concentrations within the limits have been established. If the corrective action has not been effective at the end of the proper period (24 hours or 48 hours), then the reactor will be brought to the cold shutdown condition and the corrective action will continue.

The effects of contaminants in the reactor coolant are time and temperature dependent. It is consistent, therefore, to permit a transient concentration to exist for a longer period of time and still provide the assurance that the integrity of the primary coolant system will be maintained.

In order to restore the contaminant concentrations to within specification limits in the event such limits were exceeded, mixing of the primary coolant with the reactor coolant pumps may be required. This will result in a small heatup of short duration and will not increase the average coolant temperature above 250°F.

References

1) FSAR Section 4.2

2) FSAR Section 9.2

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F. LEAKAGE OF REACTOR COOLANT

Specification

1. If leakage of reactor coolant is indicated by the means available such as water inventory balance, monitoring equipment or direct observation a follow-up evaluation of the safety implications shall be initiated as practicable but no later than within 4 hours. Any indicated leak shall be considered to be a real leak until it is determined that the indicated leak cannot be substantiated by direct observation or other indication.

2. If the leakage rate, excluding controlled leakage sources such as the Reactor Coolant Pump Controlled Leakage Seals and Leakage into Closed Systems, exceeds 1 gpm and the source of leakage is not identified, reduce the leakage rate to within limits within four hours or be in hot shutdown within the next six hours and in cold shutdown within the following 30 hours.

3. If the sources of leakage are identified and the results of the evaluation are that continued operation is safe, operation of the reactor with a total leakage, other than from controlled sources or into closed systems, not exceeding 10 gpm shall be permitted except as specified in 3.1.F.4 below.

4. If it is determined that leakage exists through a non-isolable fault which has developed in a Reactor Coolant System Component Body, pipe wall (excluding steam generator tubes), vessel wall or pipe weld, the reactor shall be brought to the cold shutdown condition within twenty-four hours.

5. If the total leakage, other than from controlled sources or into closed systems, exceeds 10 gpm, the reactor shall be placed in the hot shutdown condition within four hours and the cold shutdown condition within an additional twenty-four hours.

6. The reactor shall not be restarted following shutdown as per items 3.1.F.2, 3, 4, or 5, above, until the leak is repaired or until the problem is otherwise corrected.

7. Whenever the reactor is shutdown, or a steam generator removed from service, in order to investigate steam generator tube leakage and/or to plug or otherwise repair a leaking tube, the Authority shall inform the NRC before the reactor is brought critical.

8. Primary to secondary leakage through the steam generator tubes shall be limited to 0.3 gpm (432 gpd) per steam generator and the total leakage through all four steam generators shall be limited to 1.0 gpm (1440 gpd). With any steam generator tube leakage greater than this limit the reactor shall be placed in the hot shutdown condition within four hours and the cold shutdown condition within an additional twenty-four hours.

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9. If leakage from two or more tubes in the steam generators in any 20-day period is observed or determined, the reactor shall be brought to the hot shutdown condition within four hours and the cold shutdown condition within an additional twenty-four hours and Nuclear Regulatory Commission approval shall be obtained before resuming reactor operation. If two steam generator tube leaks attributable to the tube denting phenomena are observed after the reactor is in cold shutdown Nuclear Regulatory Commission approval shall be obtained before resuming reactor operation.

10. When the reactor is critical and above 2% power, two reactor coolant leak detection systems of different principles capable of detecting leakage into containment shall be in operation, with one of the two systems sensitive to radioactivity. The system sensitive to radioactivity may be out-of-service for 48 hours, provided two other systems are available.

Basis:

Water inventory balances, monitoring equipment, radioactive tracing, boric acid crystalline deposits, and physical inspections can disclose reactor coolant leaks. Any leak of radioactive fluid, whether from the reactor coolant system primary boundary or not can be a serious problem with respect to in-plant radioactivity contamination and cleanup or it could develop into a still more serious problem; and therefore, first indications of such leakage will be followed up as soon as practicable.

Although some leak rates on the order of GPM may be tolerable from a dose point of view, it must be recognized that small leaks through any of the walls of the primary system could be indicative of materials failure such as by stress corrosion cracking. If depressurization, isolation and/or other safety measures are not taken promptly, these small leaks could develop into much larger leaks. Therefore, the nature of the leak, as well as the magnitude of the leakage must be considered in the safety evaluation.

The distinction between identified and unidentified leakage in the specification is made because once the leakage source is identified, the seriousness can be easily evaluated. The strict limit of 1 gallon per minute for unidentified leakage is adopted because in the worst case the leakage source may increase with time or the coolant may impinge on or accumulate in a critical component.

When the source of leakage has been identified, the situation can be evaluated to determine if operation can safely continue. This evaluation will be performed by the Watch Force. Under these conditions, an allowable primary system leakage rate of 10 gpm has been established. This explained leakage rate of 10 gpm is also well within the capacity of one charging pump and makeup would be available even under the loss of off-site power condition.

Controlled sources of reactor coolant system leakage are sources which are designed to leak at a controlled rate. For example, the reactor coolant pump seals are controlled leakage sources. Leakage through a valve packing or a closed valve is not considered as controlled leakage. Leakage into closed systems is that leakage which can be accounted for and contained by a system not directly connected to the atmosphere. Leakage past the pressurizer safety valve seats and steam generator tube leakage are examples of reactor coolant system leakage into closed systems.

If leakage is to the containment, it may be identified by one or more of the following methods:

a. The containment air particulate monitor (R-11).

b. The containment radiogas monitor (R-12).

c. The containment humidity detectors.

d. A leakage detection system which determines leakage losses from all water and steam systems within the containment. This system collects and measures moisture condensed from the containment atmosphere by cooling coils of the main air recirculation units.

The most sensitive and rapid method for detecting small amounts of Reactor Coolant System leakage is the monitoring of the containment airborne radioactivity. Containment gaseous and particulate activity is continuously, automatically monitored. The leakage rate can be determined by the relationship of the airborne activity to the reactor coolant activity.

Measurement of the leakage rate to the containment atmosphere is also possible through humidity detection and condensation collection and measurement. However, it is expected that the containment activity method will give the initial indication of coolant leakage. The other methods will be employed primarily to confirm that leakage exists, to indicate the location of the leakage sources, and to measure the leakage rate.

As described above, the four reactor coolant leak detection systems are based on three different principles, i.e., activity, humidity and condensate flow measurements. Two systems of different principles provide, therefore, diversified ways of detecting leakage to the containment.

Total reactor coolant leakage can be determined by means of periodic water inventory balances. If leakage is into another closed system, it will be detected by the plant radiation monitors and/or inventory control.

Four hours is allowed from the time of leakage detection to identify the leakage source and to measure the leakage rate. This time period is required since identification and quantification of leakage sources of less than ten gallons per minute require a careful gathering and evaluation of data and/or a visual inspection of the reactor coolant system.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those limits found to result in negligible corrosion of the steam generator tubes. If stress corrosion cracking occurs, the extent of cracking during plant operation would be limited by the limitation of steam generator leakage between the primary coolant system and the secondary coolant system. Cracks having a primary-to-secondary leakage less than 500 gallons per day during operation will have an adequate margin of safety against failure due to loads imposed by design basis accidents. The 500 gallon per day per steam generator limit is also consistent with the assumptions used to develop the Technical Specification limit on secondary coolant activity. Operating plants have demonstrated that primary-to-secondary leakage as low as 0.1 gpm will be detected. Leakage in excess of 432 gallons per steam generator or 1 gpm total for all four steam generators will require plant shutdown and an unscheduled eddy current inspection, during which the leaking tubes will be located and plugged.

References

FSAR Sections 11.2.3 and 14.2.4

G. Secondary Coolant Activity

Specification

1. Whenever the average reactor coolant temperature is $\geq$ 350°F, the specific activity of the secondary coolant system shall be $\leq$ 0.10 μCi/gram of Dose Equivalent I-131.

2. If the specific activity of the secondary coolant system exceeds 0.10 μCi/gram of Dose Equivalent I-131, the reactor shall be immediately brought to the hot shutdown condition with T_{avg} <350°F utilizing normal operating procedures.

Basis

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10CFR Part 100 limits in the event of a steam line rupture. The restriction of 0.1 μCi/gram Dose Equivalent I-131 in the secondary system limits the two-hour thyroid exposure dose to 1.5 rem at the site boundary under these accident conditions. This accident analysis also includes the effects of a coincident 500 gallons per day primary to secondary tube leak in the steam generator of the affected steam line and considers the effect of a coincident iodine spike. Accident meteorological conditions are assumed (5% X/Q) and a decontamination factor of 10 is applied between the water and steam phases.

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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.

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:

4.3 REACTOR COOLANT SYSTEM INTEGRITY TESTING

Applicability

Applies to test requirements for Reactor Coolant System integrity.

Objective

To specify tests for Reactor Coolant System integrity after the system is closed following normal opening, modification or repair.

Specification

a) When the Reactor Coolant System is closed after it has been opened, the system will be leak tested at not less than 2335 psig and in accordance with NDT requirements for temperature.

b) When Reactor Coolant System modifications or repairs have been made which involve new strength welds on components, the new welds will meet the requirements of ASME Section XI.

c) The reactor coolant system leak test temerature-pressure relationship shall be in accordance with the limits of Figure 4.3-1 for heatup for the first 11.00 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

For normal opening, the integrity of the system, in terms of strength, is unchanged. If the system does not leak at 2335 psig (Operating pressure + 100 psi :± 100 psi is normal system pressure fluctuation), it will be leak tight during normal operation.

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

The inservice leak 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 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. The temperature determined by methods of ASME Code Section III for 2335 psig is 133°F above this RT_{NDT} and for 2510 psig (maximum) is 143°F above this RT_{NDT}. The minimum inservice leak test temperature requirements for periods up to 11.00 effective full power years are shown on Figure 4.3-1[2].

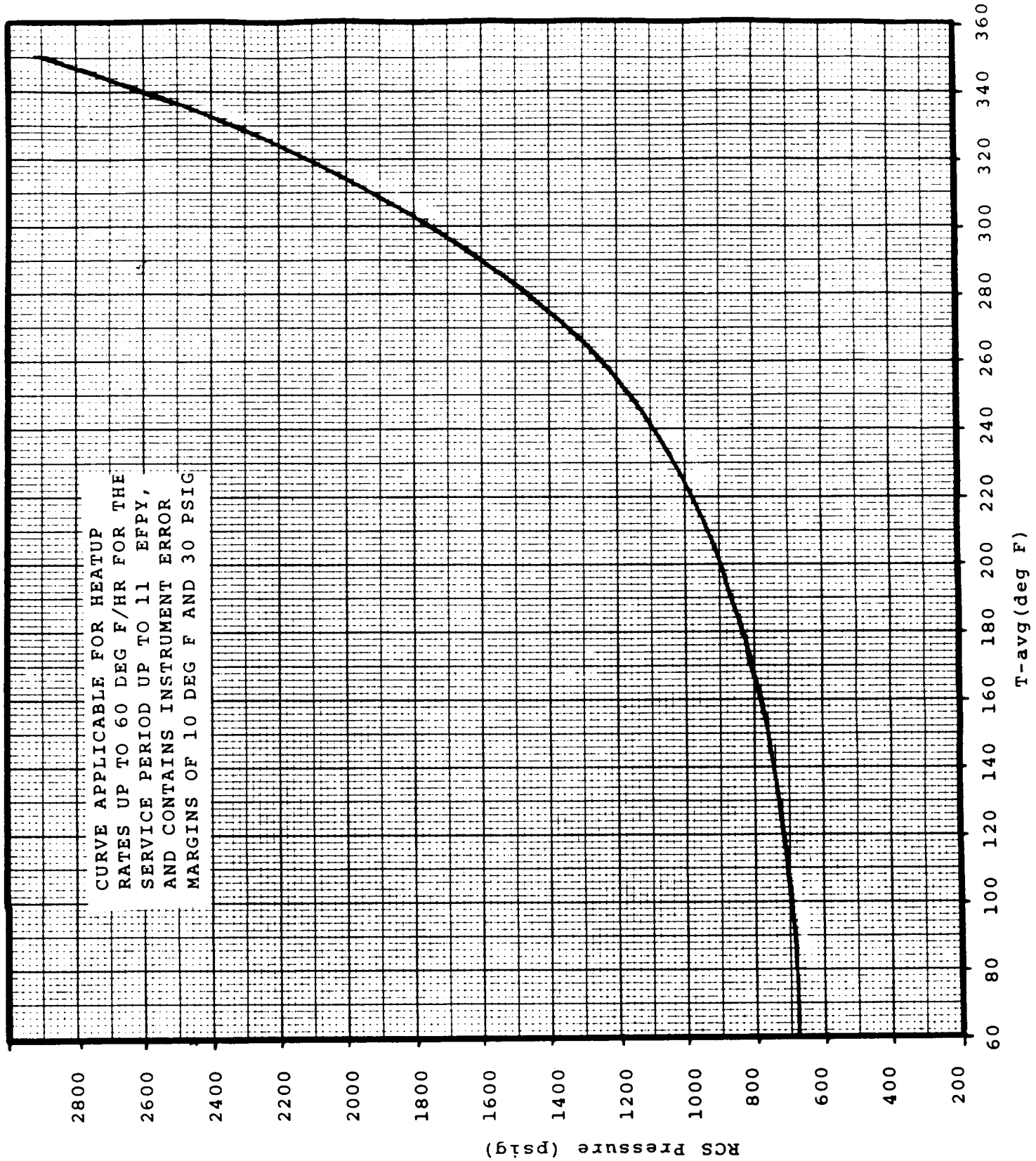
The heatup limits specified on the heatup curve, Figure 4.3-1, must not be exceeded while the reactor coolant system is being heated to the inservice leak test temperature. For cooldown from the leak test temperature, the limitations of Figure 3.1-2 must not be exceeded. Figures 4.3-1 and 3.1-2 are recalculated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

<u>Reference</u>

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

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 121 TO FACILITY OPERATING LICENSE NO. DPR-64

POWER AUTHORITY OF THE STATE OF NEW YORK

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

DOCKET NO. 50-286

1.0 INTRODUCTION

By letter dated October 28, 1991, the Power Authority of the State of New York (the licensee) submitted a request for changes to the Indian Point Nuclear Generating Unit No. 3 (IP3), Technical Specifications (TS). The requested changes would revise TS Section 3.1.A (Operational Components), Section 3.1.B (Heatup and Cooldown), Section 3.3 (Engineered Safety Features), and Section 4.3 (Reactor Coolant System Integrity Testing). These sections would be revised to extend the reactor coolant system Pressure-Temperature (P/T) limits to 11.0 effective full power years (EFPY) of operation and to provide for the corresponding Overpressure Protection System (OPS) limits. In addition, Section 3.1 would be retyped in its entirety for format consistency and to correct typographical errors.

2.0 EVALUATION

Generic Letter (GL) 88-11, "NRC Position On Radiation Embrittlement of Reactor Vessel Materials and its Impact On Plant Operations," requested that licensees use the methodology of Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," in calculating P/T limits, unless the use of different methods could be justified. The P/T limits provide for operation of the reactor coolant system during period of heatup, cooldown, criticality, and hydrostatic test. In response to GL 88-11, by letters dated August 31, 1990, and April 2, 1991, the licensee requested an amendment to the P/T limits using the recommended guidance provided in GL 88-11. At that time, the licensee requested P/T limits which would be valid for up to 9 EFPY of operation. The NRC staff found that amendment request acceptable and by letter dated August 28, 1991, issued Amendment No. 109 to the IP3 TS which allowed the licensee's requested P/T limits. The licensee's current submittal dated October 28, 1991, uses the same supporting data and evaluation methodologies which the licensee used in the submittal which the staff approved by issuance of Amendment No. 109.

2.1 Pressure Vessel Fluence Estimate

The information on which the current submittal is based is the same as that submitted for Amendment No. 109. The specific documents previously reviewed and found acceptable for up to 9 EFPY by the NRC staff were WCAP-11815,

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"Analysis of Capsule Z From The New York Power Authority Indian Point Unit 3 Reactor Vessel Radiation Surveillance Program," dated March 1988, and "Final Report on Pressure-Temperature Limits for Indian Point Unit 3 Nuclear Power Plant," dated July 1990. The licensee has shown that the fluence methods and results previously submitted can be extended to 11 EFPY. The staff has reviewed the licensee's analysis and finds the extension to 11 EFPY acceptable since the methods used are the same as previously approved.

2.2 P-T Operating Limits

The staff has evaluated the effect of neutron irradiation embrittlement on each beltline material in the IP3 reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff has determined that the material with the highest adjusted reference temperature (ART) at 11 EFPY was the lower shell course plate B2803-3 with 0.24% copper (Cu), 0.52% nickel (Ni), and an initial RT_{ndt} of 74 °F.

The licensee has removed three surveillance capsules (T, Y, and Z) from the IP3 reactor vessel. The results from the analysis of capsule Z (the most recently removed capsule) were published in WCAP-11815. The surveillance capsule contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and heat-affected-zone (HAZ) metal. For the limiting beltline material, lower shell course plate B2803-3, the staff calculated the adjusted reference temperature (ART) to be 203.5 °F at ¼T (T = reactor vessel beltline thickness) and 162.3 °F at ¾T for 11 EFPY. The staff used a neutron fluence of 3.04E18 n/cm² at ¼T and 1.05E18 n/cm² at ¾T. The ART was determined by the least squares extrapolation method using the surveillance data. The least squares method is described in Section 2.1 of RG 1.99, Rev. 2.

The licensee used the method in RG 1.99, Rev. 2, to calculate an ART of 202 °F at ¼T for the same limiting plate material at 11 EFPY. The staff finds the difference of 1.5 °F between the licensee's ART of 202 °F and the staff's ART of 203.5 °F acceptable. The staff used the more conservative ART of 203.5 °F as input for the equations of Standard Review Plan Section 5.3.2, "Pressure-Temperature Limits," to calculate P/T limits for heatup, cooldown, and hydrostatic test. These calculated P/T limits were compared to the licensee's proposed P/T limits and the staff concluded that the proposed P/T limits for heatup, cooldown, and hydrostatic test meet the beltline material requirements in Appendix G to 10 CFR Part 50.

In addition to beltline materials, Appendix G to 10 CFR Part 50 imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.A.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120 °F for normal operation and by 90 °F for hydrostatic pressure tests and leak tests. Based on the flange reference temperature of 38 °F, the staff concludes that the proposed P/T limits satisfy Section IV.A.2 of Appendix G.

Therefore, the staff concludes that the proposed P/T limits, which provide for operation of the reactor coolant system during period of heatup, cooldown, criticality, and hydrostatic test through 11 EFPY, are acceptable.

2.3 OPS Enable Temperature Setpoint

The licensee has proposed an OPS enable temperature of 332 °F for operation to 11 EFPY. This represents an 8 °F increase over the OPS enable temperature correlating to 9 EFPY. The licensee calculated the 332 °F OPS enable temperature based on the corresponding increase of RT_{ndt} at the ¼T location resulting from the increase from 9 to 11 EFPY of operation. The staff has reviewed the licensee's proposed OPS enable temperature for 11 EFPY, in accordance with Standard Review Plan Section 5.2.2, "Overpressure Protection," and concludes that the proposed change is acceptable.

2.4 Administrative Changes

The licensee has proposed to retype Section 3.1 in its entirety for format consistency and to correct typographical errors. The staff reviewed these administrative changes proposed by the licensee and finds them acceptable.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (56 FR 64661). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the

public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors:
L. Lois
S. Sheng

Date: June 18, 1992

Docket No. 50-286

June 18, 1992

Mr. Ralph E. Beedle
Executive Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Beedle:

SUBJECT: ISSUANCE OF AMENDMENT FOR INDIAN POINT NUCLEAR GENERATING UNIT NO. 3 (TAC NO. M82062)

The Commission has issued the enclosed Amendment No.121 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated October 28, 1991.

The amendment revises Technical Specifications Section 3.1.A (Operational Components), Section 3.1.B (Heatup and Cooldown), Section 3.3 (Engineered Safety Features), and Section 4.3 (Reactor Coolant System Integrity Testing). These sections have been revised to extend the reactor coolant system Pressure-Temperature (P/T) limits to 11.0 effective full power years (EFPY) of operation and to provide for the corresponding Overpressure Protection System (OPS) limits. In addition, Section 3.1 has been retyped in its entirety for format consistency and to correct typographical errors.

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

Sincerely,
Original Signed By:
Nicola F. Conicella, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 121 to DPR-64
2. Safety Evaluation

cc w/enclosures:
See next page

OFFICE	LA:PDI-1	PM:PDI-1	OGC - NLO	D:PDI-1	
NAME	CSVogan	NConicella:pc	MZOBLER	RACapra	
DATE	6/11 6/2/92 6/15	6/2/92 6/15/92	6/2/92	6/18/92	/ /92

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