ATTACHMENT III TO IPN-97-149

SAFETY EVALUATION OF
PROPOSED TECHNICAL SPECIFICATION CHANGES ASSOCIATED WITH
PRESSURE-TEMPERATURE AND OVERPRESSURE PROTECTION SYSTEM
LIMITS FOR UP TO 13 EFFECTIVE FULL POWER YEARS

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

SAFETY EVALUATION RELATED TO PROPOSED TECHNICAL SPECIFICATION CHANGES ASSOCIATED WITH PRESSURE-TEMPERATURE AND OVERPRESSURE PROTECTION SYSTEM LIMITS FOR UP TO 13 EFFECTIVE FULL POWER YEARS

Section I - Description of Changes

This application for amendment seeks to revise Sections 1, 3.1, 3.3, 4.3, and 6 of Appendix A of the Indian Point 3 Technical Specifications. These revisions extend the Heatup-Cooldown limits from 11 to 13 effective full power years (EFPYs), provide the corresponding Overpressure Protection System (OPS) limits, relocate the new pressure temperature limit curves and low temperature OPS limits to the pressure temperature limit report (PTLR) (in accordance with Reference 1) 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 2) 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 (including 13 EFPYs) was prepared in accordance with RG 1.99, Revision 2 for Indian Point 3. This report was previously submitted to the NRC by Reference 3, and approved as part of Amendment 109 (Reference 4). The new heatup and cooldown pressure-temperature curves contained in Attachment VI are effective up to 13 EFPYs and are based upon the data presented in that report.

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 methodology submitted to the NRC by Reference 3 and the use of ASME Code Case N-514. (Attachment I 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 four reactor coolant 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.

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 accepted by the NRC (References 3, 4 and 5) and is revised only to include the effects of increased vessel lifetime burnup of 13 EFPYs.

Relocation of Pressure-Temperature and LTOPS Limits to the PTLR

All components of the reactor coolant system (RCS) are designed to withstand the effects of cyclic loads resulting from system pressure and temperature changes. These loads are introduced by heatup and cooldown operations, power transients, and reactor trips. In accordance with Appendix G to 10 CFR 50, Technical Specifications limit the pressure and temperature changes during RCS heatup and cooldown within the design assumptions and the stress limits for cyclic operation. These limits are defined by pressure-temperature limit curves for heatup, cooldown, LTOP, and inservice leak and hydrostatic testing. Each curve defines an acceptable region for normal operation. The curves are used for operational guidance during heatup and cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LTOP system controls RCS pressure at low temperatures so that the integrity of the reactor coolant pressure boundary is not compromised by violating 10 CFR 50, Appendix G. This system includes two power operated relief valves which automatically open in the event of an impending overpressure condition. The LTOP system is reevaluated each time the pressure-temperature limit curves are revised to ensure that it meets its intended function.

The Technical Specification changes in Attachment II relocate the pressure-temperature and OPS limits to a newly created PTLR and are in accordance with the guidance in Reference 1, as follows.

1.) The definitions section of the Technical Specifications was modified (Section 1.19) to include a definition of the PTLR to which the figures, values, and parameters for pressure-temperature and LTOP system limits will be relocated in accordance with the methodology provided in Reference 3 and ASME Code Case N-514. As noted in the definition, plant operation within these limits is addressed by individual specifications.

    A definition of the OPS enable temperature was also added as Specification 1.20. This definition states that the OPS enable temperature is the temperature at or below which the OPS system is required to be in service. The numerical value of the temperature will be defined in the PTLR, as it is dependent on vessel age. Similarly, the numerical value for the OPS threshold temperature has been removed from the Technical Specifications and placed in the PTLR.

2.) Several technical specifications (3.1.A.1, 3.1.A.8, 3.1.B.1, 3.1.B.2, 4.3.A, associated Figures and Basis sections) were revised to replace the pressure-temperature and LTOP system limits with a reference to the PTLR that provides these limits.

3.) Technical Specification Section 6.9.1.7 was added to the reporting requirements of the administrative controls section. This specification requires that the PTLR be submitted to the NRC within 45 days of issuance. The 45 day time frame for submittal of the PTLR is slightly different from the Standard Technical Specification wording which states that the PTLR will be submitted to the NRC 'upon issuance.' The insertion of the 45 day time period provides a finite time frame for submittal of the PTLR and allows adequate time for the preparation of the NRC submittal. This revision to the Standard Technical Specification wording provided in Generic Letter 96-03 (Reference 1) has no impact on plant safety or operation.

The PTLR provides the explanations, figures, values, and parameters of the pressure-temperature and LTOP system limits for the applicable effective period. Furthermore, this specification requires that the figures, values, and parameters be established using the methodology approved by the NRC for this purpose. Section 6.9.1.7 lists the ASME Code Case N-514 as NRC approved methodology for Indian Point 3. Request for approval to use this code case is presented in Attachment I of this submittal.

Relocation of the pressure-temperature curves and LTOP parameters does not eliminate the requirement to operate in accordance with the limits specified in 10 CFR 50, Appendix G or Regulatory Guide 1.99, Revision 2. The requirement to operate within the limits in the PTLR is specified in and controlled by the Technical Specifications. Only the figures, values, and parameters associated with the pressure-temperature limits and LTOP parameters are relocated to the PTLR. The methodology for generation of the pressure-temperature curves has been previously submitted to the NRC in Reference 3. The methodology used to create the LTOP curves is based upon Reference 3 and ASME Code Case N-514. A calculation (Reference 8) was performed to document the use of these methodologies to generate the curves included in the PTLR.

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. This statement 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 9) 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 bring the Technical Specifications into agreement with 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%. 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 Figures 2.5 and 2.6 of the PTLR 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%.

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 Reference 3 and approved as Amendments 109 and 121 (References 4 and 5). 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 methodology contained in Reference 3 and ASME Code Case N-514.

The relocation of the pressure-temperature and LTOPS limits from the Technical Specifications to the PTLR does not eliminate the requirement to operate in accordance with the limits specified in 10 CFR 50, Appendix G. The requirement to operate within the limits in the PTLR is specified in and controlled by the Technical Specifications.

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 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 relocation of the pressure-temperature and OPS limits from the Technical Specifications to the PTLR does not alter the requirements associated with these limits.

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 and ASME Boiler and Pressure Vessel Code Section XI, Appendix G. 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 Reference 3 and ASME Code Case N-514. The relocation of the pressure-temperature and OPS limits to the PTLR does not alter the requirements associated with these limits.

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.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 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," dated January 31, 1996.

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

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

4. NRC letter, F. Williams to R. Beedle, "Issuance of Amendment for Indian Point 3 (TAC No. 71505)", Amendment 109, dated August 28, 1991.

5. NRC letter, N. Conicella to R. Beedle, “Issuance of Amendment for Indian Point Nuclear Generating Unit No. 3 (TAC No. M82062),” Amendment 121, dated June 18, 1992.

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. IP3-CALC-RCS-02444, “Generation of All Curves in the PTLR,” dated October 24, 1997.

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

ATTACHMENT IV TO IPN-97-149

MARK-UP OF 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 EFFECTIVE FULL POWER YEARS

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
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DPR-64

1.16 REPORTABLE EVENT

A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR 50.

1.17 CORE OPERATING LIMITS REPORT

The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.6. Plant operation within these operating limits is addressed in individual specifications.

1.18 SHUTDOWN MARGIN

SHUTDOWN MARGIN (SDM) is the instantaneous amount of negative reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full-length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

1.19 PRESSURE TEMPERATURE LIMITS REPORT (PTLR)

The PTLR is the document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 6.9.1.7. Plant operation within these operating limits is addressed in Sections 3.1.A.1, 3.1.A.8, 3.1.B and 4.3.A.

1.20 OVERPRESSURIZATION PROTECTION SYSTEM (OPS) ENABLE TEMPERATURE

The OPS ENABLE TEMPERATURE is the temperature at or below which the Overpressurization Protection System is required to be in service. This temperature is controlled and defined in the PTLR.

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~~ **the OPS enable temperature,** ~~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}. **Following the start of one or more RCPs and prior to reaching the OPS enable temperature, the RCS pressure shall not exceed that given by Figure 2.3 in the PTLR;**

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~~ **2.4 in the PTLR;**

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~~ **73** percent, and the RCS pressure does not exceed that given by ~~Curve I on Fig. 3.1.A-2;~~ **Figure 2.3 in the PTLR. The pressurizer level must be restricted per Figures 2.5 and 2.6 of the PTLR if operation below the OPS enable temperature 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~~, ~~95~~, ~~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.~~

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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.~~ **When the RCS temperature is below the OPS enable temperature,**

1. **the OPS shall be armed and operable. Both OPS PORVs shall have lift settings not to exceed those given in Figure 2.3 of the PTLR, or**

2. **the RCS must be vented with an equivalent opening of 2.0 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~~ **the maximum OPS**

threshold temperature defined in Figure 2.3 of the PTLR;

Or

(3) Restrict pressurizer level as per the curves ~~referenced below:~~ **on Figures 2.5 and 2.6 of the PTLR.**

~~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.~~

~~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~~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 2.3 of the PTLR. 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 four reactor coolant 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~~ **the OPS enable temperature** 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 2.3 in the PTLR**) 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 **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~~ **the OPS enable temperature** 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 2.0 square inches.**

The OPS ~~arming~~ **enable** temperature ~~of 332°F~~ permits the performance of an RCS hydrostatic test (see Fig. ~~4.3-1~~ **2.7 in the PTLR**) without activating the OPS.

Upon OPS inoperability, the RCS may be heated above ~~370°F. This~~ **the** temperature ~~is~~ that **is the** value for which the RCS heatup and cooldown curves (~~Figures 3.1-1 and 3.1-2~~ **Figures 2.1 and 2.2 in the PTLR**) permit pressurization to the setting of the pressurizer safety valves. ~~Accordingly, with an inoperable OPS and an RCS temperature 370°F,~~ **At these conditions**, 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**).

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 1993.**
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

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

DELETED

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

Figure 3.1.A-2

~~MAXIMUM PERMISSIBLE RCS PRESSURE FOR RCP START WITH OPS INOPERABLE~~

DELETED

FIGURE 3.1.A-3

~~RCS PRESSURE LIMITS FOR LOW TEMPERATURE OPERATION~~

DELETED

FIGURE 3.1.A-4

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

DELETED

FIGURE 3.1.A-5

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

DELETED

FIGURE 3.1.A-6

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

DELETED

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)~~ **Figures 2.1 and 2.2 in the PTLR**. 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~~ **PTLR Figures 2.1 and 2.2** 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~~ **the service life identified in all Figures of the PTLR.**

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~~ **for the analyzed service life is identified in Figures 2.1 and 2.2 of the PTLR. The limiting plate in the reactor vessel is Plate B2803-3, which is also more limiting than all reactor vessel welds.**

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} at the end of ~~11.00~~ **the analyzed** 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].

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~~ **2.2 of the PTLR** 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.

FIGURE 3.1-1

~~Reactor Coolant System Heatup Limitations - Indian Point 3~~

DELETED

FIGURE 3.1-2

~~Reactor Coolant System Cooldown Limitations -- Indian Point 3~~

DELETED

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 the OPS enable temperature, 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%. (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 $\geq$35% and $\leq$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%) 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.

<u>References</u>

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 11.00 EFPYs of operations~~ **2.7 in the PTLR for the applicable service period.** ~~Figure 4.3-1~~ **This figure** will be recalculated periodically. Allowable pressures during cooldown from the leak test temperature shall be in accordance with Figure ~~3.1-2~~ **2.2 in the PTLR.**

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~~ **2.7 of the PTLR**. 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).~~

RT_{NDT} predictions for the applicable service period are shown on PTLR Figure 2.7, with the accompanying margins to safety limits.

The heatup limits specified ~~on the heatup curve, Figure 4.3-1~~ **in Figure 2.7 of the PTLR**, 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 **PTLR** Figure ~~3.1-2~~ **2.2** must not be exceeded. ~~Figures 4.3-1 and 3.1-2~~ **PTLR Figures 2.2 and 2.7** 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~~

FIGURE 4.3-1

~~Pressure/Temperature Limits for Hydrostatic Leak Test~~

DELETED

Amendment No. 28, 109, 121,

3f. WCAP-12610, "VANTAGE+ Fuel Assembly Report," (W Proprietary).

(Methodology for Specification 3.10.2 - Heat Flux Hot Channel Factor).

6.9.1.6.c The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety limits are met.

6.9.1.6.d The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

6.9.1.7 Pressure and Temperature Limits Report (PTLR)

a. **RCS pressure and temperature limits for heatup, cooldown, low temperature operation, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:**

1. **OPS limits for specification 3.1.A.8,**

2. **Heatup and cooldown limits for specification 3.1.B, and**

3. **Pressure/temperature limits for the RCS leak test for specification 4.3.A.**

b. **The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:**

1. **"Final Report on the Pressure-Temperature Limits for Indian Point 3 Nuclear Power Plant," July 1990. (Approved as part of Amendment 109.)**

(Methodology for specifications 3.1.A.8, 3.1.B, and 4.3.B)

2. **ASME Code Case N-514, "Low Temperature Overpressure Protection," February 1992. (Approved by NRC for use at IP3 through an exemption to 10 CFR 50.60.)**

(Methodology for specification 3.1.A.8.)

c. **The PTLR shall be provided to the NRC within 45 days of issuance for each reactor vessel fluence period and for any revision or supplement thereto.**

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator-Region 1 within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification;

a. Sealed source leakage on excess of limits (Specification 3.9)

b. Inoperable Seismic Monitoring Instrumentation (Specification 4.10)

c. Seismic event analysis (Specification 4.10)

d. Inoperable plant vent sampling, main steam line radiation monitoring or effluent monitoring capability (Table 3.5-4, items 5, 6 and 7)

e. The complete results of the steam generator tube inservice inspection (Specification 4.9.C)

f. Deleted

g. Release of radioactive effluents in excess of limits (Appendix B Specifications 2.3, 2.4, 2.5, 2.6)

h. Inoperable containment high-range radiation monitors (Table 3.5-5, Item 24)

i. Radioactive environmental sampling results in excess of reporting levels (Appendix B Specification 2.7, 2.8, 2.9)

j. Operation of Overpressure Protection System (Specification ~~3.1.A.8.c~~ **3.1.A.8.d**).

k. Operation of Toxic Gas Monitoring Systems (Specification 3.3.H.3.)

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspection, repair and replacements of principal items of equipment related to nuclear safety.

c. ALL REPORTABLE EVENTS submitted to the Commission.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of changes made to Operating Procedures.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all source material of record.

i. Records of reactor tests and experiments.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Records of any drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

ATTACHMENT V TO IPN-97-149

COMMITMENTS FOR THE PROPOSED TECHNICAL SPECIFICATION CHANGES ASSOCIATED WITH PRESSURE-TEMPERATURE AND OVERPRESSURE PROTECTION SYSTEM LIMITS FOR UP TO 13 EFFECTIVE FULL POWER YEARS

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

COMMITMENTS ASSOCIATED WITH IPN-97-149

Comm. No.	Commitment Description	Due Date
IPN-97-149-01	Issue final version of PTLR.	Prior to amendment implementation.
IPN-97-149-02	Submit to the NRC the final PTLR.	45 days after issuance of PTLR.
IPN-97-149-03	Revise FSAR.	Next applicable FSAR update.

ATTACHMENT VI TO IPN-97-149

INDIAN POINT 3 PRESSURE TEMPERATURE LIMITS REPORT

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

INDIAN POINT 3

PRESSURE AND TEMPERATURE LIMITS REPORT

REVISION 0 - DRAFT

Table of Contents

List of Figures

List of Tables

1.0 Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)

This PTLR for IP3 has been prepared in accordance with the requirements of Technical Specification (TS) 6.9.1.7. The following TS sections are addressed in this report:

3.1.A	Reactor Coolant System Operational Components;
3.1.B	Heatup and Cooldown; and
4.3.A	Reactor Coolant System Integrity Testing.

2.0 Operating Limits

The parameter limits for the specifications listed in Section 1.0 are presented in the following subsections. The current limits were developed using a methodology (Reference 1) that is in accordance with the NRC approved methodology specified in TS 6.9.1.7 with one exception. The exception is the use of ASME Code Case N-514 for the establishment of the OPS enable temperature and setpoints (Reference 2). Future changes to these limits will be made in full compliance with NRC-approved methodology and/or exceptions approved by the NRC for IP3.

2.1 RCS Pressure and Temperature (P/T) Limits

2.1.1 The RCS temperature rate of change limits are:

a. Maximum heatup of 60°F/hour in any one hour period, and

b. Maximum cooldown of 100°F/hour in any one hour period, with procedural controls in place to further limit the maximum cooldown rate to 50°F/hour above 200°F and 40°F/hour below 200°F.

2.1.2 The RCS P/T limits for heatup and cooldown are specified by Figures 2.1 and 2.2, respectively.

2.2 Low Temperature Overpressure Protection System (OPS) Setpoints

The power operated relief valves (PORVs) shall each have lift settings in accordance with Figure 2.3.

The setpoints in combination with the relief capacity of the PORVs will protect the RCS from the limiting mass injection transient of one safety injection pump injecting into the RCS and the limiting heat input transient of starting a RCP with the RCS 100°F colder than the secondary coolant (64°F indicated).

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 valve flange or when the pressurizer level is low enough (indicating 0%) to ensure at least a ten minute operator response time on inadvertent SI actuation without the pressurizer completely filling.

Figure 2.1
Reactor Coolant System Heatup Limitations

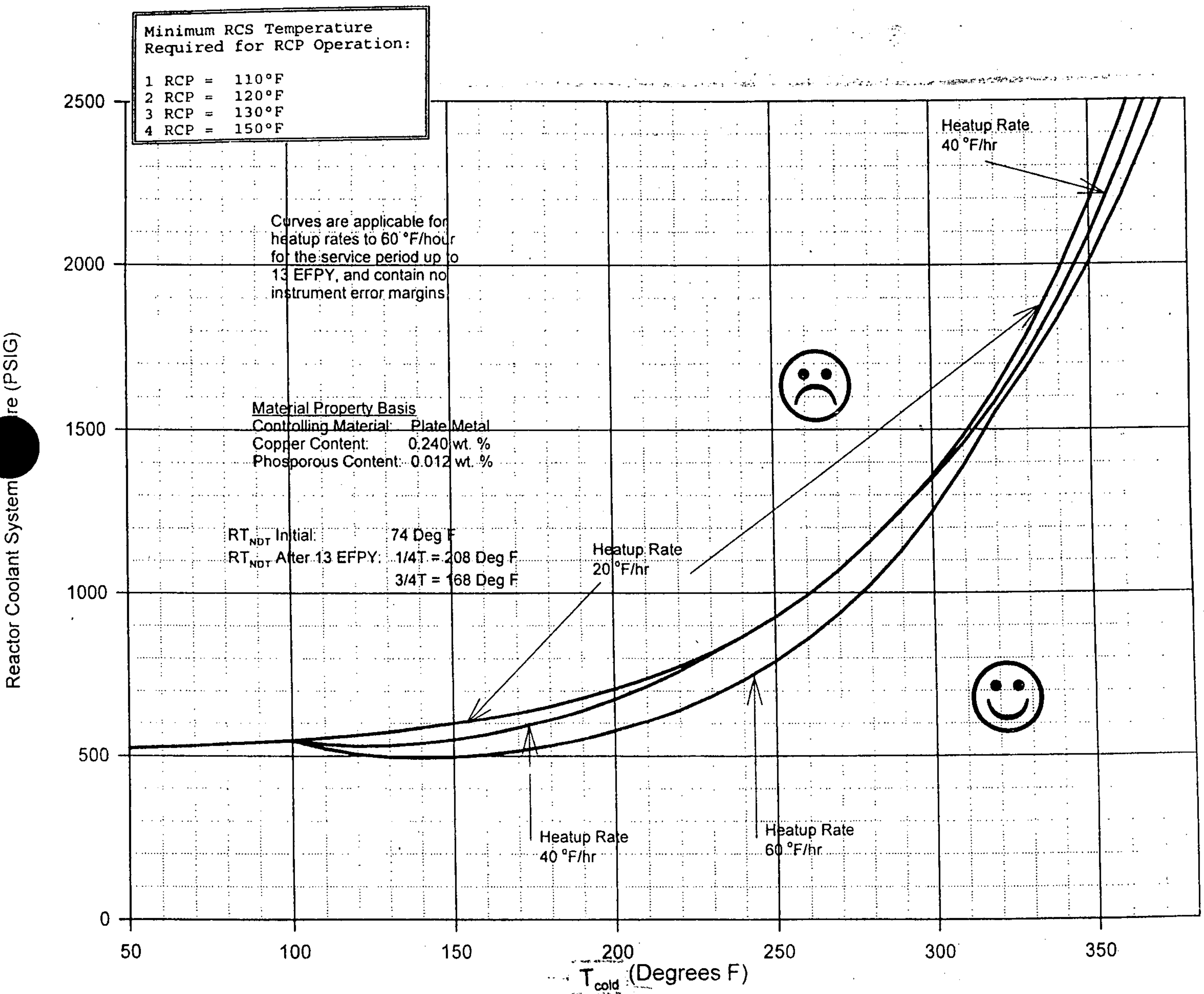


Table 2.1
Data Points for IP3 Reactor Coolant System Heatup Limitations

RCS TEMP DEG F	HEATUP 20 F/HR -ALLOWABLE (PSIG) 20 F/ HOUR	HEATUP 40 F/HR P-ALLOWABLE (PSIG) 40 F/ HOUR	HEATUP 60 F/HR P-ALLOWABLE (PSIG) 60 F/ HOUR
50	522.3	522.3	522.3
60	526.0	526.0	526.0
70	530.2	530.2	530.2
80	535.1	535.1	535.1
90	540.7	540.7	540.7
100	547.3	547.3	544.6
110	554.8	534.2	520.9
120	563.5	529.1	505.4
130	573.6	530.8	496.8
140	585.3	538.0	494.0
150	598.8	549.7	496.4
160	614.4	565.8	503.4
170	632.4	585.9	514.9
180	653.2	610.4	530.7
190	677.3	639.5	551.0
200	705.2	673.7	576.0
210	737.4	713.6	606.1
220	774.6	759.9	642.0
230	817.6	813.8	684.2
240	867.3	867.3	733.5
250	924.8	924.8	791.1
260	991.3	991.3	858.0
270	1068.1	1068.1	935.7
280	1156.9	1156.9	1025.7
290	1253.2	1256.2	1130.0
300	1358.7	1350.0	1250.7
310	1480.6	1458.4	1390.4
320	1621.6	1583.6	1552.0
329.9	-	-	-
329.9	-	-	-
330	1784.6	1728.5	1687.4
330.82	-	-	-
340	1973.0	1895.9	1836.2
350	2190.8	2089.5	2008.3
360	2442.6	2313.2	2207.1
361.97	2500.0	-	2437.0
370	2733.7	2571.9	2500.0
380	3000.0	2870.9	2702.7
390	-	3000.0	3000.0
400	-	-	-
401.97	-	-	-
410	-	-	-
420	-	-	-
		-	-

Figure 2.2
Reactor Coolant System Cooldown Limititations

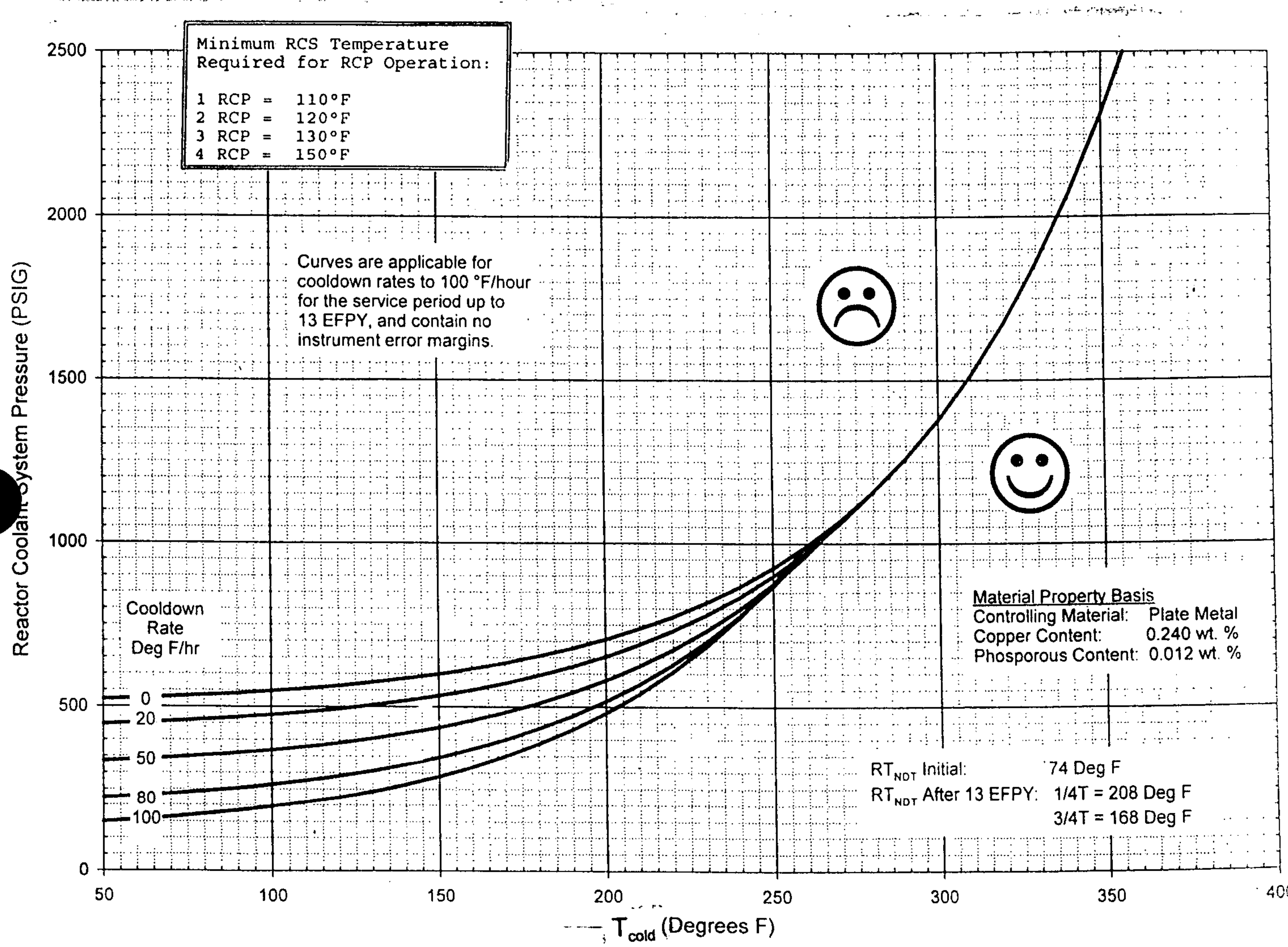
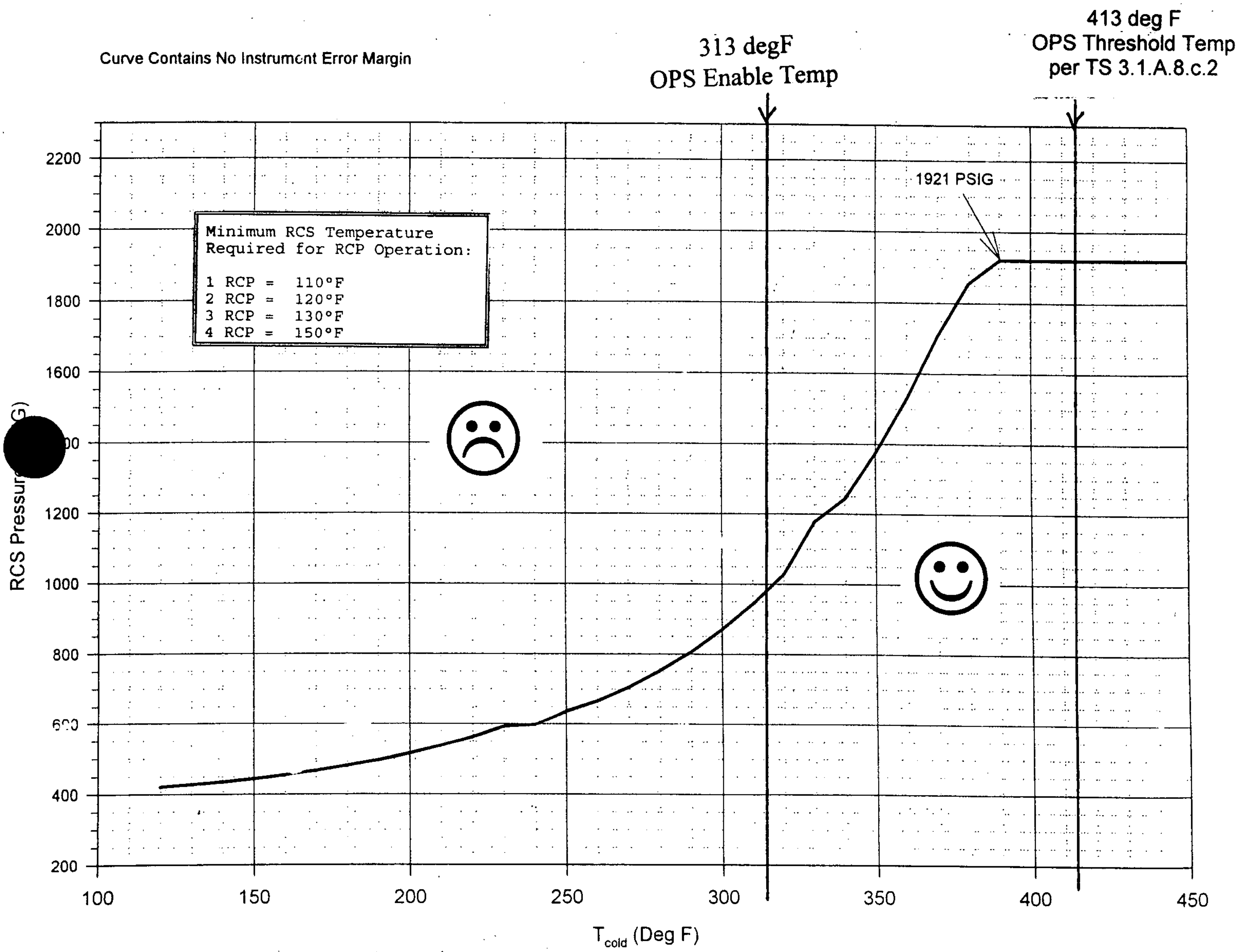


Table 2.2
Data Points for IP3 Reactor Coolant System Cooldown Limitations

RCS TEMP DEG F	COOLDOWN P-ALLOWABLE (PSIG)					
	ISO THERMAL	20 F/ HOUR	50 F/ HOUR	60 F/ HOUR	80 F/ HOUR	100 F/ HOUR
50	522.3	446.9	334.6	297.4	223.3	149.8
60	526.0	451.1	339.5	302.6	229.2	156.5
70	530.2	455.8	345.2	308.6	235.9	164.2
80	535.1	461.3	351.8	315.6	243.8	173.1
90	540.7	467.7	359.4	323.6	252.8	183.3
100	547.3	475.0	368.1	332.9	263.4	195.0
110	554.8	483.5	378.3	343.7	275.4	208.8
120	563.5	493.3	390.0	356.1	289.5	224.7
130	573.6	504.7	403.6	370.5	305.6	243.0
140	585.3	517.8	419.3	387.1	324.5	264.0
150	598.8	533.0	437.3	406.3	346.0	288.2
160	614.4	550.5	458.3	428.6	371.2	316.7
170	632.4	570.8	482.5	454.2	399.9	349.4
180	653.2	594.3	510.5	483.9	433.6	387.1
190	677.3	621.4	542.9	518.2	472.0	430.5
200	705.2	652.7	580.2	557.9	517.0	480.4
210	737.4	688.9	623.6	603.8	568.2	539.0
220	774.6	730.8	673.5	656.8	628.4	606.5
230	817.6	779.2	731.3	718.1	696.9	684.1
240	867.3	835.2	798.2	789.0	777.3	773.3
250	924.8	899.9	875.2	870.9	868.7	875.8
260	991.3	974.6	964.8	965.6	976.0	991.3
270	1068.1	1061.1	1067.9	1068.1	1068.1	1068.1
280	1156.9	1156.9	1156.9	1156.9	1156.9	1156.9
290	1259.6	1259.6	1259.6	1259.6	1259.6	1259.6
300	1378.3	1378.3	1378.3	1378.3	1378.3	1378.3
310	1515.5	1515.5	1515.5	1515.5	1515.5	1515.5
320	1674.2	1674.2	1674.2	1674.2	1674.2	1674.2
330	1857.5	1857.5	1857.5	1857.5	1857.5	1857.5
340	2069.6	2069.6	2069.6	2069.6	2069.6	2069.6
350	2314.6	2314.6	2314.6	2314.6	2314.6	2314.6
356.54	2500.0	2500.0	2500.0	2500.0	2500.0	2500.0
360	2598.0	2598.0	2598.0	2598.0	2598.0	2598.0
370	2925.5	2925.5	2925.5	2925.5	2925.5	2925.5
380	3000.0	3000.0	3000.0	3000.0	3000.0	3000.0

Figure 2.3
Maximum Allowable Nominal PORV Setpoint for the
Low Temperature Overpressure Protection System (OPS), 13 EFPY

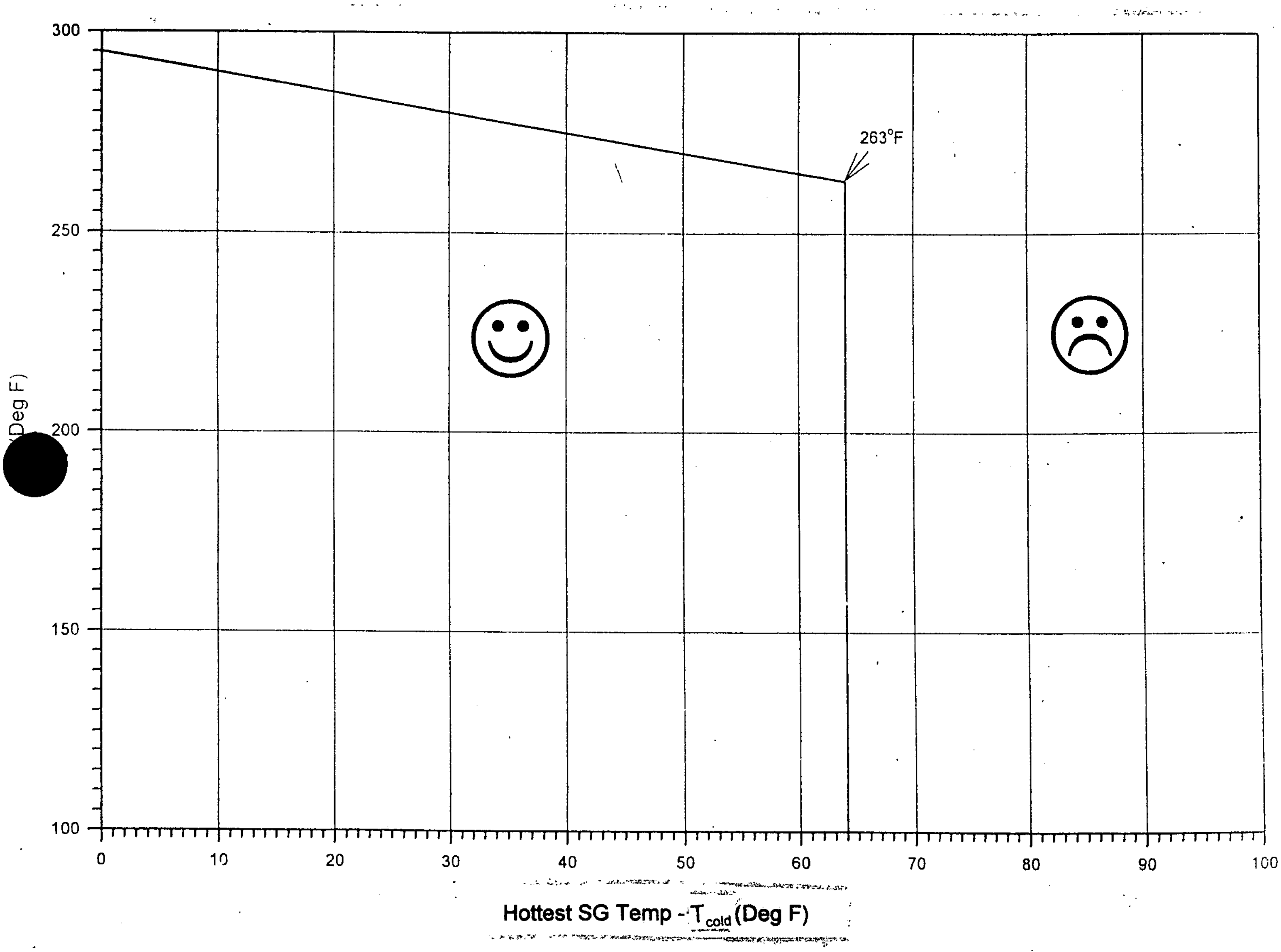


OPS Enable Temp and OPS Threshold Temp include a 30°F allowance for instrument uncertainties.

Table 2.3
Data Points for IP3 PORV Setpoints, 13 EFPY

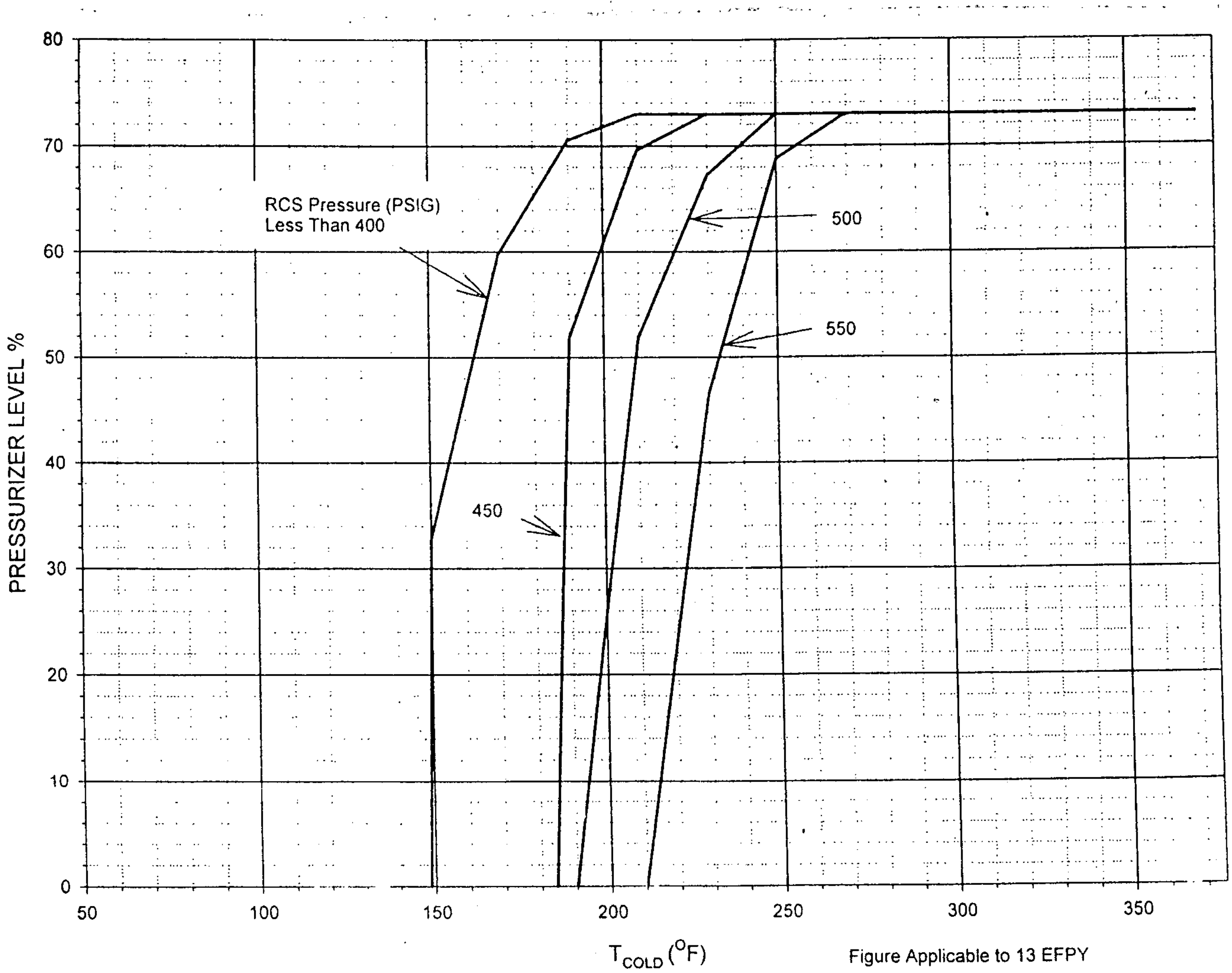
RCS Average T-Inlet (°F)	Limiting RCS Pressure for OPS (psig)	Comments
120	420.3	
130	427.1	
140	435.0	
150	444.1	
160	454.7	
170	466.8	
180	481.0	
190	497.3	
200	516.1	
210	537.9	
220	563.2	
230	592.2	
240	597.2	
250	638.2	
260	666.7	
270	706.2	
280	752.6	
290	806.8	
300	870.2	
310	943.8	
320	1029.3	
330	1178.3	
340	1243.0	
350	1375.8	
360	1529.4	
370	1707.2	
380	1856.1	Max OPS Press
390	1921.0	
400	1921.0	

Figure 2.4
Secondary Side Limitations for RCP Start with Secondary Side Hotter Than Primary, 13 EFPY



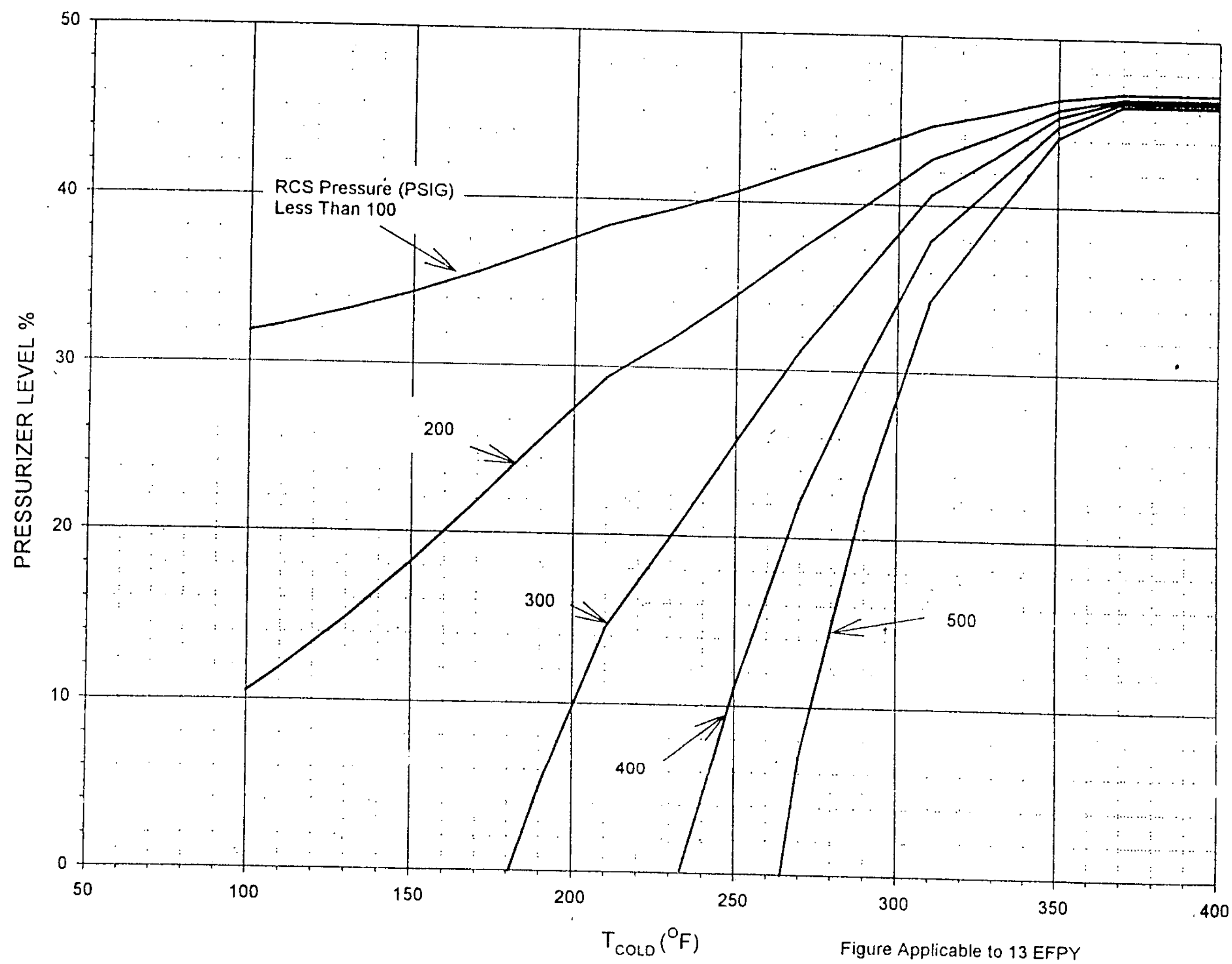
This curve contains an allowance for instrument error.

Figure 2.5
Pressurizer Limitations for OPS Inoperable
(up to one charging pump capable of feeding RCS), 13 EFPY



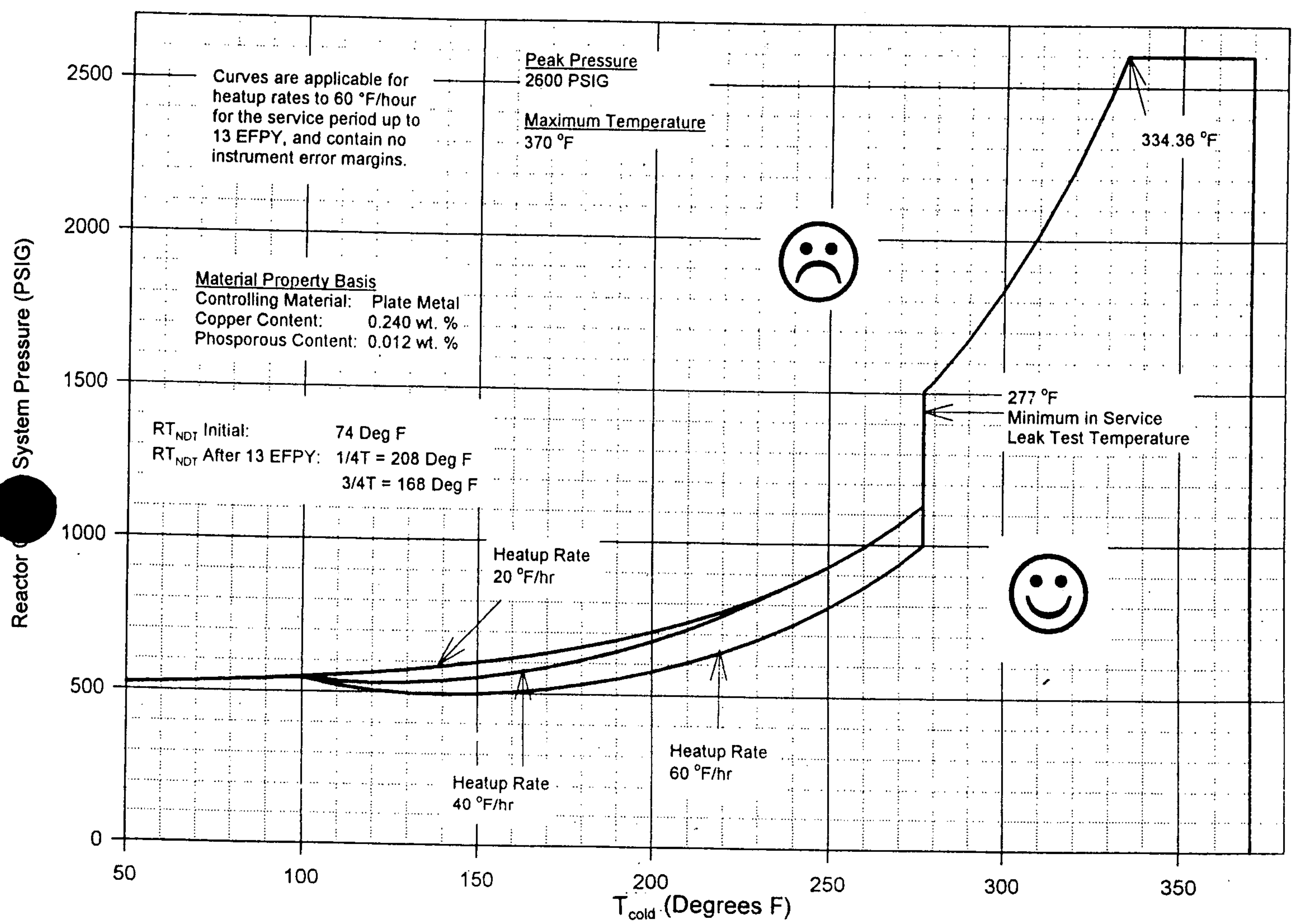
This curve contains an allowance for instrument error.

Figure 2.6
Pressurizer Limitations for OPS Inoperable (up to three charging pumps and/or one safety injection pump capable of feeding RCS), 13 EFPY



This curve contains an allowance for instrument error.

Figure 2.7
RCS Pressure Limitations for Hydrostatic Testing, 13 EFPY



Note: For the first 13.0 Effective Full Power Years, it is predicted that the highest RT_{NDT} in the core region taken at the 1/4 thickness will be 208 degrees F. The temperature determined by methods of ASME Code Section III for 2335 psig is 117 degrees F above this RT_{NDT} and for 2510 psig is 123 degrees F above this RT_{NDT}.

3.0 Supplemental Data Tables

Table 3.1 is a comparison of the measured surveillance material 30 ft-lb temperature increases and upper shelf energy decreases with Regulatory Guide 1.99, Revision 2 predictions.

Table 3.2 shows the calculation of the surveillance material chemistry factors using surveillance capsule data.

Table 3.3 provides a summary of the measured (Capsule Z Report - Reference 3) and extrapolated fluences on the IP3 reactor vessel.

Table 3.4 provides a summary of the adjusted reference temperatures (ARTs) of the IP3 limiting reactor vessel beltline material at 1/4T and 3/4T locations for 9 to 32 EFPY (lower shell plate B2803-3).

Table 3.5 provides a summary of fluences at key locations of the IP3 reactor vessel clad/metal interface (Capsule Z Report - Reference 3).

Table 3.6 provides RT_{PTS} for IP3 for a range of capacity factors, burnups and fluences.

Table 3.1

COMPARISON OF MEASURED VERSUS PREDICTED 30 ft-lb TEMPERATURE INCREASES AND UPPER SHELF DECREASES FOR INDIAN POINT UNIT 3 REACTOR VESSEL MATERIALS BASED ON THE PREDICTION METHODS OF REGULATORY GUIDE 1.99 PROPOSED REVISION 2

			30 ft-lb Transition Temp Increase (°F)		Upper Shelf Energy Decrease (ft-lb)	
Material	Capsule	Fluence (10^{19} n/cm^2)	Measured	R.G.1.99 Rev.2 Prediction(a)	Measured	R.G.1.99 Rev. 2 Prediction(b)
B2802-1 (Long.)	T	0.292	89	82	13	27
B2802-3 (Long.)	Z	1.07	150	133	22	37
B2803-3 (Long.)	T	0.292	137	106	9	25.5
B2803-3 (Long)	Z	1.07	170	163	23	35
B2803-3 (Trans.)	T	0.292	118	106	9	16.5
B2803-3 (Trans.)	Y	0.805	150	150	10	21
B2803-3 (Trans)	Z	1.07	155	163	11	22
Weld Metal	T	0.292	143	137	29	27.5
Weld Metal	Y	0.805	180	193	52	34
Weld Metal	Z	1.07	220	210	44	36.5
Correlation Mat'l	Y	0.805	140	96	25	26.5
HAZ Metal	Y	0.805	150	-	36	-

a) Based on average copper and nickel content reported in Table 4-1 and 4-2.
b) Based on average copper content reported in Table 4-1 and 4-2.

Table 3.2

CHEMISTRY FACTOR DERIVATION
PLATE B2803-3 (TRANSVERSE)

Irradiation Capsule	ΔRTNDT[a], "A"(°F)	Neutron Fluence (10^{19} n/cm^2)	Fluence Factor[b],"B"	"A" X "B"	$(B)^2$
T	118	0.3226	0.689	81.31	0.4749
Y	150	0.805	0.939	140.87	0.8810
Z	155	1.07	1.019	157.93	1.0382
				380.11	2.3951

$$\frac{\Sigma \text{ A X B}}{\Sigma\ (B)^2} = \frac{380.11}{2.3951} = \underline{158.70.\ \text{Chemistry Factor}}$$

(a) Shift in reference temperature measured at 30-foot-pound level

(b) Fluence Factor = $f^{(0.28\ -\ 0.10\ \log f)}$
where f = neutron fluence in units of 10^{19} n/cm^2, E>1Mev

Table 3.3

SUMMARY OF FAST NEUTRON EXPOSURE PROJECTIONS FOR THE INDIAN POINT UNIT 3 REACTOR VESSEL

	5.55 EFPY		22.5 EFPY	
	Φ (E > 1.0 MeV) (n/cm^2)	(dpa)	Φ (E > 1.0 MeV) (n/cm^2)	(dpa)
Vessel IR	3.13×10^{18}	5.10×10^{-3}	1.08×10^{19}	1.75×10^{-2}
Vessel 1/4T	1.65×10^{18}	3.26×10^{-3}	5.69×10^{18}	1.11×10^{-2}
Vessel 1/2T	7.51×10^{17}	1.97×10^{-3}	2.60×10^{18}	6.75×10^{-3}
Vessel 3/4T	3.29×10^{17}	1.13×10^{-3}	1.14×10^{18}	3.87×10^{-3}
Vessel OR	1.32×10^{17}	5.41×10^{-4}	4.95×10^{17}	1.85×10^{-3}

Note: Data are based on the extrapolation of Capsule Z dosimetry results to vessel locations.

Table 3.4

ADJUSTED REFERENCE TEMPERATURE
PROJECTIONS FOR INDIAN POINT UNIT 3

(Lower Shell Plate B2803-3)

Time (EFPY)	Vessel Wall Location[a]	Neutron Fluence n/cm^2, 10^{18} n/cm^2)	Adjusted Reference Temperature (°F)
9.0	1/4 T	2.795	194
9.0	3/4 T	0.993	157
11.0	1/4 T	3.332	202
11.0	3/4 T	1.184	163
13.0	1/4 T	3.868	208
13.0	3/4 T	1.374	168
15.0	1/4 T	4.405	214
15.0	3/4 T	1.565	172
32.0	1/4 T	8.976	245
32.0	3/4 T	3.188	200

(a) Fraction of vessel wall thickness, T = 8.625 inches

Table 3.5

CALCULATED FAST NEUTRON EXPOSURE RATES AT THE
CLAD/BASE METAL INTERFACE - CYCLE 5

Location	ϕ(E > 1.0 MeV) (n/cm2-sec)	dpa/sec
45°	1.43×10^{10}	2.32×10^{-11}
30°	1.03×10^{10}	1.67×10^{-11}
15°	9.70×10^{9}	1.56×10^{-11}
0°	6.85×10^{9}	1.12×10^{-11}

Table 3.6

RT_{PTS} CALCULATIONS

Case	EOL Date	Avg ϕ_n beyond Cycle 9 ($n/cm^2 s$)	Cap F (%)	f	RT_{PTS} Base-plate (°F)	RT_{PTS} Longi-tudinal Weld (°F)	RT_{PTS} Girth Weld (°F)
1	12/12/15	1.30E10	80	1.141	255.55	213.69	193.61
2	12/12/15	1.30E10	75	1.099	253.89	211.54	191.68
3	**12/12/15**	**1.047E10**	**80**	**0.977**	**248.67**	**204.76**	**185.60**
4	**12/12/15**	**1.047E10**	**75**	**0.944**	**247.14**	**202.77**	**183.82**
5	12/12/25	1.30E10	80	1.470	266.64	228.10	206.53
6	12/12/25	1.30E10	75	1.407	264.74	225.63	204.31
7	**12/12/25**	**1.047E10**	**80**	**1.242**	**259.28**	**218.55**	**197.96**
8	**12/12/25**	**1.047E10**	**75**	**1.192**	**257.48**	**216.20**	**195.86**
9	12/12/30	1.30E10	80	1.634	271.19	234.01	211.85
10	12/12/30	1.30E10	75	1.561	269.23	231.47	209.55
11	**12/12/30**	**1.047E10**	**80**	**1.374**	**263.71**	**224.29**	**203.11**
12	**12/12/30**	**1.047E10**	**75**	**1.315**	**261.79**	**221.80**	**200.88**
13	12/12/35	1.30E10	80	1.797	275.23	239.26	216.53
14	12/12/35	1.30E10	75	1.714	273.23	236.66	214.20
15	**12/12/35**	**1.047E10**	**80**	**1.506**	**267.69**	**229.46**	**207.75**
16	**12/12/35**	**1.047E10**	**75**	**1.439**	**265.72**	**226.90**	**205.45**

Note: RT_{PTS} values in normal font were determined using the Bounding Method

RT_{PTS} values in boldface were determined using the Best Estimate Method

4.0 Reactor Vessel Material Surveillance Program

The reactor vessel material surveillance program is in compliance with 10 CFR 50, Appendix H. The reactor vessel surveillance program and capsule withdrawal schedule are described in Section 4.5.2 of the IP3 FSAR. The IP3 surveillance capsule reports are as follows:

1. 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.

2. 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 1993.

3. 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.

5.0 Reactor Vessel Surveillance Data Credibility

Regulatory Guide (RG) 1.99, Revision 2 presents five criteria by which surveillance data are judged to be credible (i.e., acceptable for determining adjusted reference temperature (ART) following RG 1.99). These criteria are addressed in Reference 1 and summarized below.

Criterion 1: Material in the capsules should be those judged most likely to be controlling with regard to radiation embrittlement according to the recommendations of RG 1.99.

This criterion was addressed by calculating the ART for each of the beltline materials following RG 1.99. The IP3 material exhibiting the highest ART is plate B2803-3, and is, therefore, the controlling beltline material. This plate is included in the surveillance capsules, thus satisfying the first credibility criterion.

Criterion 2: Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions should be small enough to permit the determination of the 30-foot-pound temperature and the upper-shelf-energy unambiguously.

Charpy impact tests are performed before and after irradiation to develop an average curve of impact energy versus temperature from which values of the 30-foot-pound index temperature and upper-shelf energy are obtained. The pre-irradiation and post-irradiation test results for the IP3 surveillance materials were reviewed, and no significant scatter in Charpy impact test results was observed. Each data set was adequate for extracting reasonable values from the mean curve, thus satisfying the second credibility criterion.

Criterion 3: When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Position 2.1 normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail this criterion for use in shift calculations, they may be credible for determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM-185-82.

This criterion provides a means for judging whether the trend exhibited by the irradiated materials is consistent with other similar vessel materials and is within acceptable limits. Two or more irradiated data points are available for three surveillance materials from IP3. Each set was evaluated using Regulatory Position 2.1. (In the case of Capsule T data, an updated fluence of 3.226×10^{18} n/cm^2 was employed versus the originally reported value of 2.92×10^{18} n/cm^2. The fluence update was performed by the Hanford Engineering Development Laboratory for the NRC as part of the Light Water Reactor Pressure Vessel Surveillance Dosimetry Improvement Program.) The computed chemistry factor (CF) and the maximum difference between predicted and actual shift for each set is as follows:

Material	CF	Maximum Difference
Plate B2803-3 (transverse)	158.7	9°F
Plate B2803-3 (longitudinal)	176.87	15°F
Surveillance Weld	205.3	13°F

In each case, the maximum difference between the predicted and actual shift is less than 17°F for base metal and 28°F for weld metal, thus satisfying the third credibility criterion.

Criterion 4: The irradiation temperature of the Charpy specimens in the capsule should match vessel wall temperature at the cladding/base metal interface within ±25°F.

The surveillance capsule is typically designed to maintain the temperature of the included specimens close to that of the coolant inlet temperature. At IP3, the temperature monitors did not melt, demonstrating that the capsule irradiation temperature did not exceed 579°F. The vessel inlet temperature for the most recent fuel cycles, 539.1°F, would approximate the temperature at the vessel cladding-base metal interface, and is consistent with the results obtained on the correlations monitor material (i.e., the measured shift for the HSST Plate 02 material was above the mean predicted shift as would be expected for an irradiation temperature less than 550°F). Therefore, there is indirect evidence that the capsule irradiation temperature and the vessel temperature were within 25°F, thus satisfying the intent of the fourth credibility criterion.

Criterion 5: The surveillance data for the correlation monitor material in the capsule should fall within the scatter band of the data base for that material.

IP3, Capsule Y contained specimens from HSST Plate 02. The measured shift was 140°F corresponding to a neutron fluence of 8.05 x 10^{18} n/cm^2. Based on a compilation of similar correlation monitor material data, the mean predicted shift for Plate 02 is 120°F with a range of 80°F to 145°F. The measured shift, therefore, falls within the scatter band of the HSST Plate 02 data base, consistent with the fifth credibility criterion. As noted previously, the measurement was greater than the mean predicted shift consistent with the lower irradiation temperature for IP3 (approximately 539.1°F) versus the nominal 550°F irradiation temperature for the overall HSST Plate 02 data base.

In conclusion, all five criteria of Regulatory Guide 1.99, Revision 2, have been addressed, and the IP3 surveillance data have been shown to satisfy those criteria, and, therefore, are credible for use in developing a plant specific relationship of RT_{NDT} shift to neutron fluence in accordance with Regulatory Position 2.1.

6.0 References

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

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

3. 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.