Carolina Power & Light Company
PO Box 165
New Hill NC 27562

James Scarola
Vice President
Harris Nuclear Plant

SERIAL: HNP-00-065
10CFR50.90

APR 12 2000

United States Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
REQUEST FOR LICENSE AMENDMENT
TECHNICAL SPECIFICATIONS 3/4.4.9.2, 3/4.4.9.4

Dear Sir or Madam:

In accordance with the Code of Federal Regulations, Title 10, Part 50.90, Carolina Power & Light Company (CP&L) requests a revision to the Technical Specifications (TS) for the Harris Nuclear Plant (HNP). The proposed amendment revises Technical Specifications (TS) 3/4.4.9.2 "Pressure/Temperature (P-T) Limits - Reactor Coolant System", and 3/4.4.9.4 "Overpressure Protection System", and associated Bases. Specifically, HNP proposes to revise the applicable TS to incorporate results of the Reactor Vessel Surveillance Program capsule analysis. A summary report was previously submitted to the NRC (HNP-99-157, dated 11/9/99) in accordance with 10 CFR 50 Appendix H. Additionally, HNP requests an exemption to 10 CFR 50.60 (a), based on American Society of Mechanical Engineers (ASME) Code Case N-640 and WCAP-15315.

Enclosure 1 provides a description of the proposed changes and the basis for the changes. Enclosure 2 details, in accordance with 10 CFR 50.91(a), the basis for CP&L's determination that the proposed changes do not involve a significant hazards consideration. Enclosure 3 provides an environmental evaluation which demonstrates that the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental assessment is required for approval of this amendment request. Enclosure 4 provides page change instructions for incorporating the proposed revisions. Enclosure 5 provides the proposed Technical Specification pages.

In accordance with 10 CFR 50.91(b), CP&L is providing the State of North Carolina with a copy of the proposed license amendment.

CP&L requests that the proposed amendment be issued by July 5, 2000 to allow time for procedure revision and orderly incorporation into copies of the Technical Specifications prior to expiration of the current P-T limits on approximately August 5, 2000.

A001

Please refer any questions regarding this submittal to Mr. E. A. McCartney at (919) 362-2661.

Sincerely,

James Scarola

James Scarola

MSE/mse

Enclosures:
1. Basis for Change Request
2. 10 CFR 50.92 Evaluation
3. Environmental Considerations
4. Page Change Instructions
5. Technical Specification Pages

James Scarola, having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of his information, knowledge and belief, and the sources of his information are employees, contractors, and agents of Carolina Power & Light Company.

Darlene S. Yarborough

Notary (Seal)
My commission expires: 2-21-2005

DARLENE S. YARBOROUGH
NOTARY
PUBLIC
HARNETT COUNTY, NC

c: Mr. J. B. Brady, NRC Sr. Resident Inspector
Mr. Mel Fry, Director, N.C. DEHNR
Mr. R. J. Laufer, NRC Project Manager
Mr. L. A. Reyes, NRC Regional Administrator

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BASIS FOR CHANGE REQUEST

Background

The Harris Nuclear Plant (HNP) Reactor Vessel Surveillance Program requires withdrawal and analysis of surveillance capsule 'X' at 9 effective full power years (EFPY). Physical withdrawal of the capsule occurred during refueling outage 8 in November 1998. The analysis of the capsule by a vendor was completed, and a summary of the results was submitted to the NRC as required by 10 CFR 50 Appendix H (Letter HNP-99-157, dated 11/9/99).

The current technical specifications for pressure-temperature limits for heatup and cooldown, and for low temperature overpressure protection setpoints, are valid up to 11 EFPY. This will be achieved no sooner than August 5, 2000, based on anticipated plant capacity factor and refueling outage-9 schedule.

Implementation of capsule 'X' results prior to expiration of current requirements is needed to support continued plant operation beyond 11 EFPY.

Proposed Change

Harris Nuclear Plant (HNP) requests an exemption to 10 CFR 50.60 (a), based on American Society of Mechanical Engineers (ASME) Code Case N-640 and WCAP-15315.

HNP proposes to modify Technical Specifications (TS) 3/4.4.9.2 "Pressure/Temperature (P-T) Limits - Reactor Coolant System", and 3/4.4.9.4 "Overpressure Protection System", and associated Bases. Specifically, HNP proposes to revise the applicable TS to incorporate results of the Reactor Vessel Surveillance Program capsule analysis.

HNP also proposes to clarify TS 3/4.4.9.2 by eliminating the word "criticality" in the Limiting Condition for Operation.

Basis for exemption to 10 CFR 50.60 (a):

The proposed amendment to modify P-T limits and Low Temperature Overpressure Setpoints (LTOPS) rely in part on this requested exemption. These revised P-T limits, as specified in ASME Code Case N-640, use a lower stress intensity factor, K_{IC} instead K_{IR}, which results in higher allowable pressures. K_{IR} is a reference stress intensity factor and is based on the lower band values of K_{IC} and K_{IA}.

Use of the K_{IC} in determining the lower bound fracture toughness in the development of P-T operating limits curve is more technically correct than the K_{IA} curve. The K_{IC} curve appropriately implements the use of static initiation fracture toughness behavior to evaluate the integrity of the reactor vessel.

On July 29, 1999, the NRC issued a similar exemption based in part on ASME Code Case N-640 to Oconee Nuclear Station Units 1,2, and 3. The NRC stated in issuing the exemption that:

> "The licensee has determined that the use of the initial conservatism of the K_{IA} curve when the curve was codified in 1974 was justified. This initial conservatism was necessary due to the limited knowledge of reactor pressure vessel materials. Since 1974, additional knowledge has been gained about reactor pressure vessel materials, which demonstrates that the lower bound on fracture toughness provided by the K_{IA} curve is well beyond the margin of safety required to protect the public health and safety from potential reactor pressure vessel failure. In addition, P-T curves based on the K_{IC} curve will enhance overall plant safety by opening the P-T operating window with the greatest safety benefit in the region of low temperature operations. The two primary safety benefits in opening the low temperature operating window are a reduction in the challenges to RCS power operated relief valves and elimination of RCP impeller cavitation wear."

HNP has similar concerns with challenging reactor coolant system (RCS) power operated relief valves (PORV) during low temperature operations, as well as concerns with reactor coolant pump (RCP) impeller cavitation wear. In addition, the pressure undershoot due to the relief capacity of one PORV and the time delay for the valve to reclose after opening for pressure relief due to an LTOPS challenge can result in damage to the RCP seals due to inadequate differential pressure across the seal. Such an event would require an unplanned shutdown to replace the seals. By increasing the LTOPS setpoint at lower RCS temperatures, this decreases the likelihood of a challenge to the PORVs, and allows operation at a higher pressure, providing additional margin to protect the RCP seals.

The P-T limits determined using ASME Code Case N-640 are also less restrictive than the requirements of 10 CFR 50 Appendix G, Section IV.a, which limits the maximum pressure to 20% of the preservice hydrostatic test pressure of the reactor vessel.

Therefore, HNP requests an exemption to 10 CFR 50.60 (a), based on American Society of Mechanical Engineers (ASME) Code Case N-640 and WCAP-15315.

Basis for proposed License Amendment:

The summary report for Capsule 'X', BAW-2355 September 1999, was previously submitted to the NRC (HNP-99-157, dated 11/9/99). This report provided details of the examination of capsule 'X' of the reactor vessel surveillance program. The examination was conducted in accordance with the applicable regulatory requirements of 10 CFR 50 Appendix G and Appendix H, and code requirements of ASME III and XI, Appendix G, as well as Regulatory Guide 1.99 Revision 2.

The data from the analysis of capsule 'X' described in this report, and the methodology described in topical report BAW-10046A, Rev. 2, as well as ASME Code Case N-640 and WCAP-15315, provides the basis for development of the proposed pressure-temperature limits, consistent with prior limits calculated for HNP. The limits were not adjusted for instrument uncertainties, in accordance with the current Technical Specification bases. Instrument uncertainties are controlled by HNP procedure PLP-106, as described in the current Technical Specification bases.

The proposed P-T limits based on capsule 'X' are less restrictive than the current limits. This is due to a change in the methodology used to develop the limits in accordance with ASME Code Case N-640 and WCAP-15315. ASME Code Case N-640 and WCAP-15315 allows the use of a lower stress intensity factor, K_{IC} instead of K_{IR}, which results in higher allowable pressures, and justifies allowing higher pressures in the reactor vessel closure flange region at low temperatures. (K_{IR} is a reference stress intensity factor and is based on the lower band values of K_{IC} and K_{IA}.) The proposed limits are applicable up to 36 EFPY, which corresponds to the end of the current 40-year license, assuming a 90% capacity factor.

The LTOPS setpoints and heatup and cooldown rates were evaluated against the revised pressure-temperature limits, and changes to these setpoints and rates are summarized below:

RCS Temp (°F)	LTOP High Setpoint (PSIG)	LTOP Low Setpoint (PSIG)
<250	410	400
250 - 325	410 - 450 (linear)	400 - 440 (linear)
325 - 360	450 - 2400 (linear)	440 - 2400 (linear)

Heatup Rate Limit

<350°F	50°F/hr

Cooldown Rate Limit

120-350°F	50°F/hr
<120°F	30°F/hr

The licensing basis mass and heat addition transients, and the assumptions and conservatisms used to determine the acceptability of the proposed LTOPS setpoints and heatup and cooldown rate limits, are not changed, and remain as described in the current Technical Specification Bases.

The heat addition transient analysis is based on the methodology of Westinghouse Electric Corporation Pressure Mitigating Systems Transient Analysis Results, July 1977 and Supplement 1, September 1977.

The mass addition transient analysis calculates the peak RCS pressure which occurs after the PORV opened at the setpoint, taking into account instrument uncertainties and response times and PORV capacity and opening times. The peak pressure is determined to be less than the allowable pressure for the limiting heatup or cooldown rate for any given temperature when LTOPS is required to be operable

The proposed changes to LTOPS setpoints occurs between 90 - 125°F, by increasing the setpoint from a low of 370/380 psig to 400/410 psig, eliminating the linear ramp from 100 - 125°F. Increasing the setpoint in this low temperature region allows a greater operating window of allowable pressures and provides additional margin between the normal operating pressures and the lower LTOPS setpoint, reducing the likelihood of a pressure transient causing an LTOPS actuation. A higher operating pressure also allows additional margin for operation of the RCPs, reducing the likelihood of damage to the RCP seals due to low differential pressure across the seals in the event of a challenge to LTOPS.

The proposed changes to the heatup and cooldown rates increase the allowable rates at lower temperatures. This allows additional operating margin during heatup and cooldown of the RCS, which reduces the burden on the operator for control of heatup and cooldown rates.
For cooldown below 125°F, the current limiting cooldown rate limit is 10°F/hr and the pressure limit at 90°F (limiting low temperature) is 538 psig with +31 psi margin. With an allowed 30°F/hr cooldown rate at 90°F, the new pressure limit is 566 psig, a 28 psi increase. Thus a 30 psi increase in the setpoint reduces the margin by 2 psi, which can be accommodated.

For heatup below 125°F, a 10°F/hr limit is currently provided, and the pressure limit at 90°F is 550 psig with +25 psi of margin. With an allowed 50°F/hr heatup rate at 90°F, the new pressure limit is 563 psig, a 13 psi increase. Thus a 30 psi increase in the setpoint reduces the margin by 17 psi, which can be accommodated.

Note that the LTOPS setpoints are only credited for overpressure protection for Technical Specifications at or above 90°F, which is unchanged from the current Technical Specification requirements.

HNP also proposes to clarify TS 3/4.4.9.2 by eliminating the word "criticality" in the Limiting Condition for Operation. HNP does not operate a critical reactor in Mode 4 through Mode 6, therefore this term is not applicable.

Conclusion:

HNP requests a revision to TS 3/4.4.9.2 and 3/4.4.9.4 and an exemption request for 10 CFR 50.60 (a), based on American Society of Mechanical Engineers (ASME) Code Case N-640 and WCAP-15315. The proposed amendment is necessary due to expiration of P-T curves that are currently in HNP TS. HNP requests that the proposed amendment be issued by July 5, 2000 to allow time for procedure revision and orderly incorporation into copies of the TS prior to expiration of the current P-T limits on approximately August 5, 2000.

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10 CFR 50.92 EVALUATION

The Commission has provided standards in 10 CFR 50.92(c) for determining whether a significant hazards consideration exists. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. Carolina Power & Light Company has reviewed this proposed license amendment request and determined that its adoption would not involve a significant hazards determination. The bases for this determination are as follows:

Proposed Change

Harris Nuclear Plant (HNP) requests an exemption to 10 CFR 50.60 (a). based on American Society of Mechanical Engineers (ASME) Code Case N-640 and WCAP-15315.

HNP proposes to modify Technical Specifications (TS) 3/4.4.9.2 "Pressure/Temperature (P-T) Limits - Reactor Coolant System", and 3/4.4.9.4 "Overpressure Protection System", and associated Bases. Specifically, HNP proposes to revise the applicable TS to incorporate results of the Reactor Vessel Surveillance Program capsule analysis.

HNP also proposes to clarify TS 3/4.4.9.2 by eliminating the word "criticality" in the Limiting Condition for Operation.

Basis

This change does not involve a significant hazards consideration for the following reasons:

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed changes affect operations of the Reactor Coolant System (RCS) components when the RCS temperature is below 350°F. The revisions to P-T limits and allowable heatup and cooldown rate limits are consistent with ASME code cases which have been authorized for other licensees by the NRC. The proposed changes modify the setpoint of the pressurizer PORVs for LTOPS. Changes to the LTOPS setpoints applicable below 350°F effectively increase the allowable operating pressure for any given temperature during shutdown. These changes do not result in conditions which are outside of the design basis for RCS Structures, Systems, and Components (SSCs). Therefore, the proposed changes do not alter the characteristics of the RCS SSCs adversely, and therefore do not impact the performance of the RCS SSCs during power operations.

The revised P-T limits and heatup and cooldown rate limits are within the design capabilities of the RCS SSCs and pressure control systems. While the proposed new P-T limits are less restrictive than the current Technical Specificaiton requirements, they assure that plant operation is within the design capacity of the reactor vessel materials. Therefore, the RCS capability as a fission product barrier is not compromised.

The changes to the LTOPS setpoints do not affect accident consequences since no credit is assumed for operation of LTOPS to mitigate accidents.

Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed changes do not involve new plant components or procedures, but only revise existing operational limits and setpoints. These changes do not place SSCs in conditions outside of their design basis, and the revised operating setpoints and conditions are within the capability of the plant control systems.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in the margin of safety.

   The proposed changes to the P-T limits and LTOPS setpoints change the calculational method from that described in the bases to one based on ASME Code Case N-640, and on WCAP-15315. The effect of this change is to allow plant operation with different limits, but still with adequate margins to assure the integrity of the reactor vessel and RCS.

   Therefore, the proposed change does not involve a significant reduction in the margin of safety.

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ENVIRONMENTAL CONSIDERATIONS

10 CFR 51.22(c)(9) provides criterion for and identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite; (3) result in a significant increase in individual or cumulative occupational radiation exposure. Carolina Power & Light Company has reviewed this request and determined that the proposed 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 needs to be prepared in connection with the issuance of the amendment. The basis for this determination follows:

Proposed Change

Harris Nuclear Plant (HNP) requests an exemption to 10 CFR 50.60 (a), based on American Society of Mechanical Engineers (ASME) Code Case N-640 and WCAP-15315.

HNP proposes to modify Technical Specifications (TS) 3/4.4.9.2 "Pressure/Temperature (P-T) Limits - Reactor Coolant System", and 3/4.4.9.4 "Overpressure Protection System", and associated Bases. Specifically, HNP proposes to revise the applicable TS to incorporate results of the Reactor Vessel Surveillance Program capsule analysis.

HNP also proposes to clarify TS 3/4.4.9.2 by eliminating the word "criticality" in the Limiting Condition for Operation.

Basis

The change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) for the following reasons:

1. As demonstrated in Enclosure 2, the proposed amendment does not involve a significant hazards consideration.

2. The proposed amendment does not result in a significant change in the types or increase in the amounts of any effluents that may be released offsite.

   The change does not introduce any new effluents or significantly increase the quantities of existing effluents. As such, the change cannot significantly affect the types or amounts of any effluents that may be released offsite.

3. The proposed amendment does not result in a significant increase in individual or cumulative occupational radiation exposure.

   The proposed change does not result in any physical plant changes or new surveillances which would require additional personnel entry into radiation controlled areas. Therefore, the amendment has no significant affect on either individual or cumulative occupational radiation exposure.

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PAGE CHANGE INSTRUCTIONS

Removed Page	Inserted Page
viii	viii
xiv	xiv
3/4 4-34	3/4 4-34
3/4 4-35	3/4 4-35
3/4 4-36	3/4 4-36
3/4 4-38	3/4 4-38
3/4 4-41	3/4 4-41
B3/4 4-6	B3/4 4-6
B3/4 4-7	B3/4 4-7
B3/4 4-9	B3/4 4-9
B3/4 4-11	B3/4 4-11
B3/4 4-12	B3/4 4-12
B3/4 4-13	B3/4 4-13

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TECHNICAL SPECIFICATION PAGES

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

Delete

LIMITING CONDITION FOR OPERATION

3.4.9.2 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with: Delete

a. A maximum heatup rate as shown on Table 4.4-6.

b. A maximum cooldown rate as shown on Table 4.4-6.

c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: MODES 4, 5, and 6 with reactor vessel head on.

ACTION:

With any of the pressure limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; if the pressure and temperature limit lines shown on Figure 3.4-2 and 3.4-3 were exceeded, perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or maintain the RCS T_{avg} and pressure at less than 200°F and 500 psig, respectively.

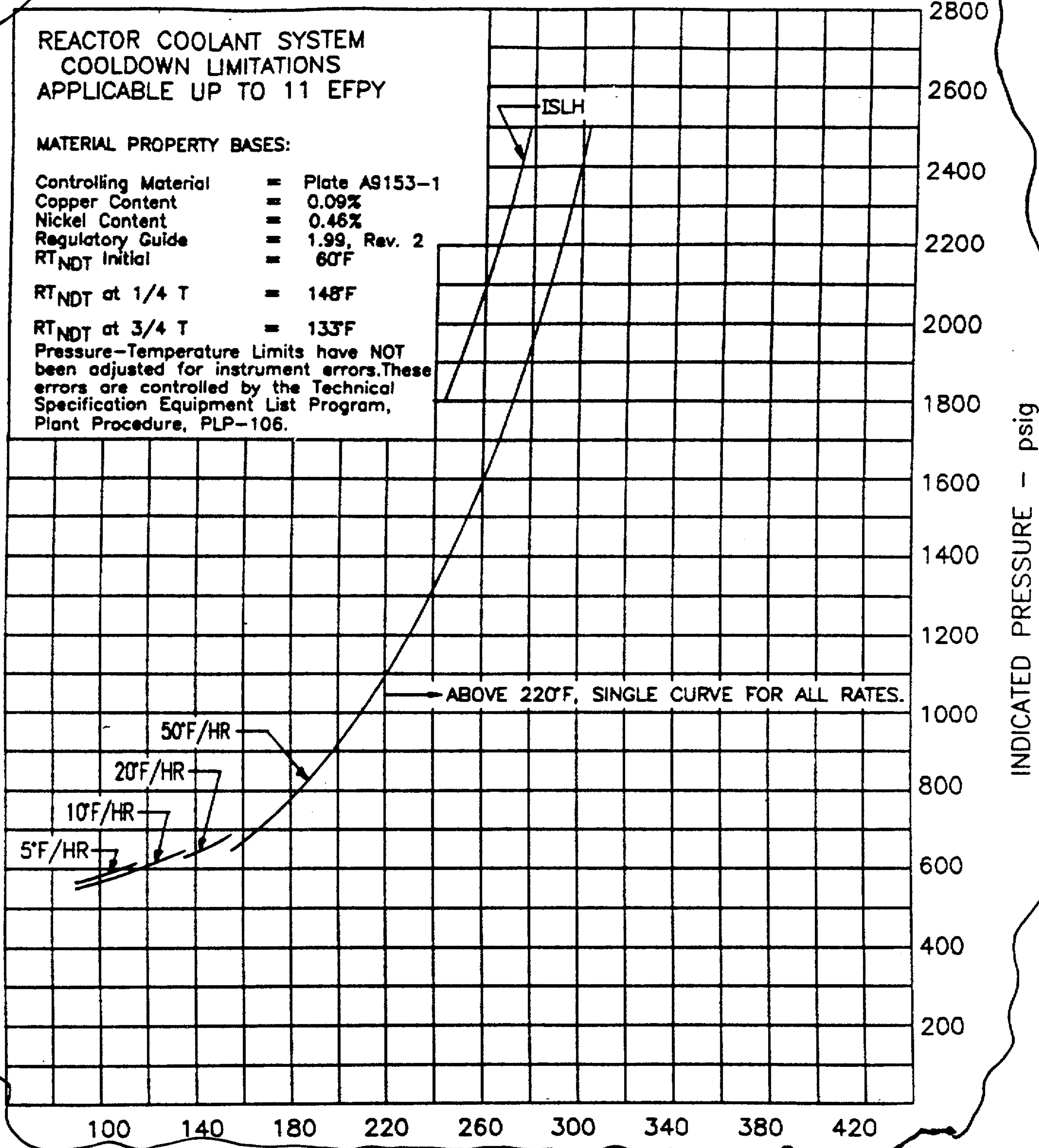
SURVEILLANCE REQUIREMENTS

4.4.9.2.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations. Delete

4.4.9.2.2 Deleted from Technical Specifications. Refer to the Technical Specification Equipment List Program, plant procedure PLP-106.

Delete

Add insert



INDICATED TEMPERATURE - DEGREES °F

FIGURE 3.4-2

REACTOR COOLANT SYSTEM

COOLDOWN LIMITATIONS - APPLICABLE UP TO 11 EFPY

36 Add

Delete

Delete

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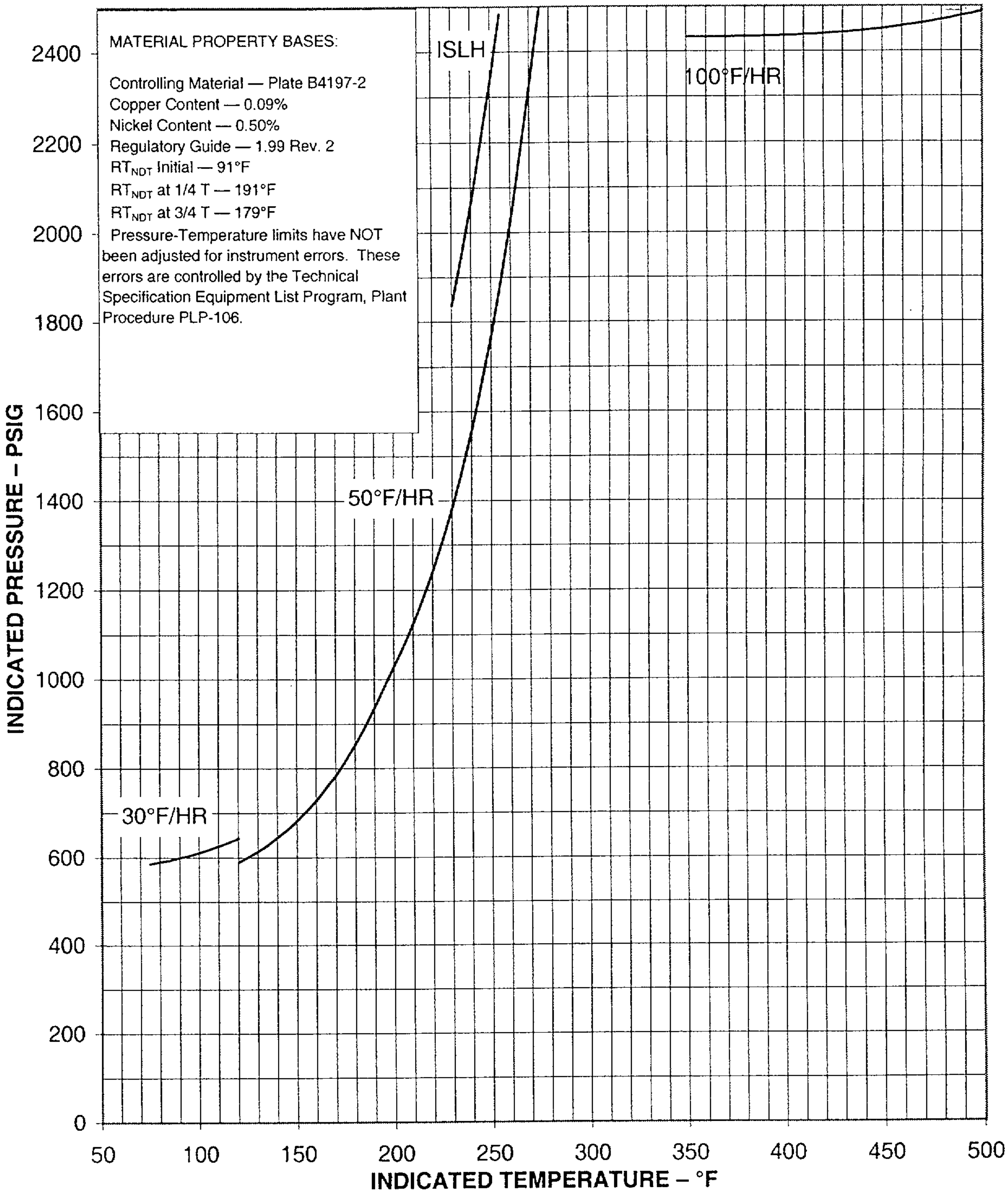

FIGURE 3.4–2
REACTOR COOLANT SYSTEM
COOLDOWN LIMITATIONS—APPLICABLE UP TO 36 EFPY

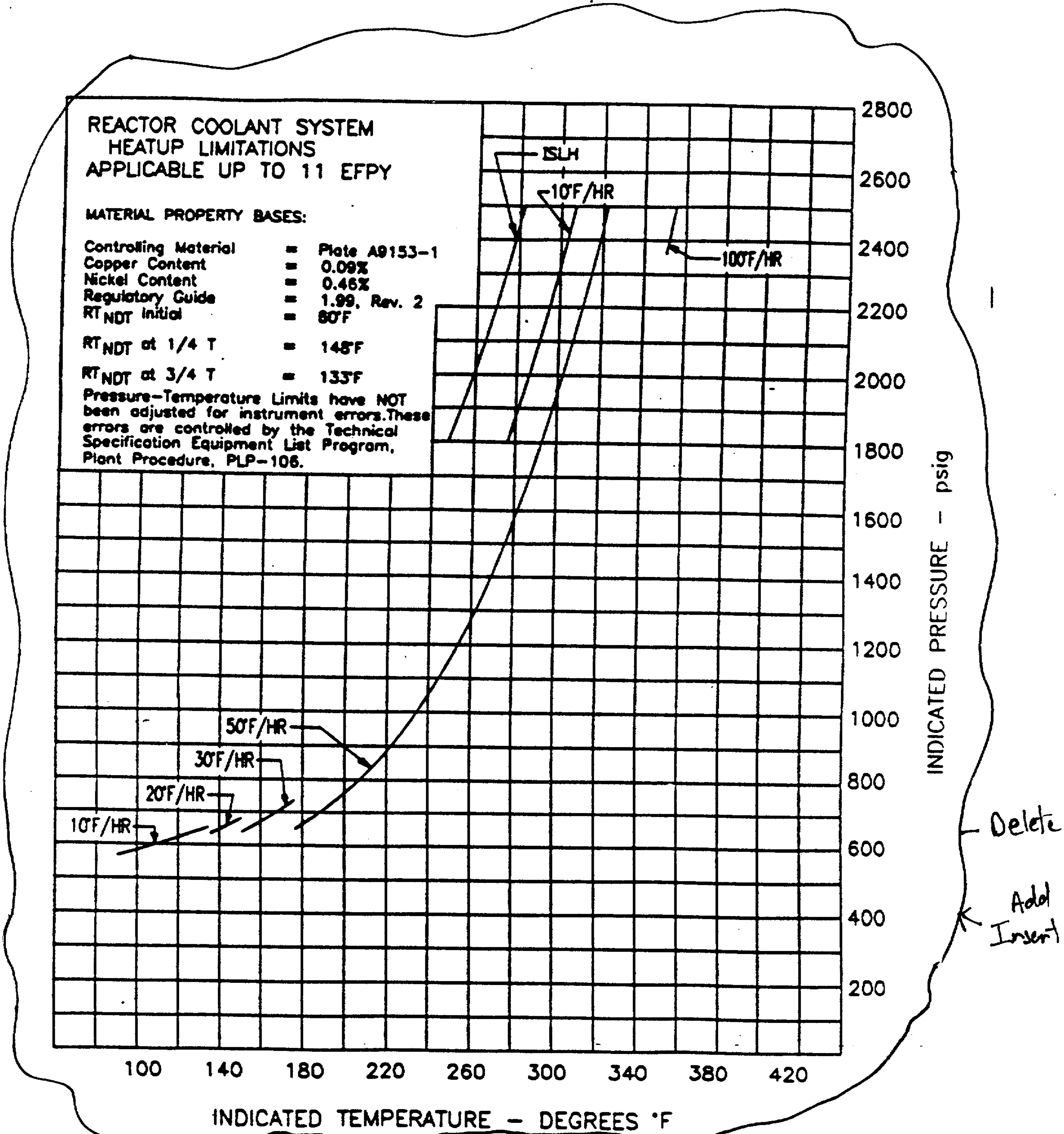

FIGURE 3.4-3

REACTOR COOLANT SYSTEM 26 Add

HEATUP LIMITATIONS - APPLICABLE UP TO 11 EFPY

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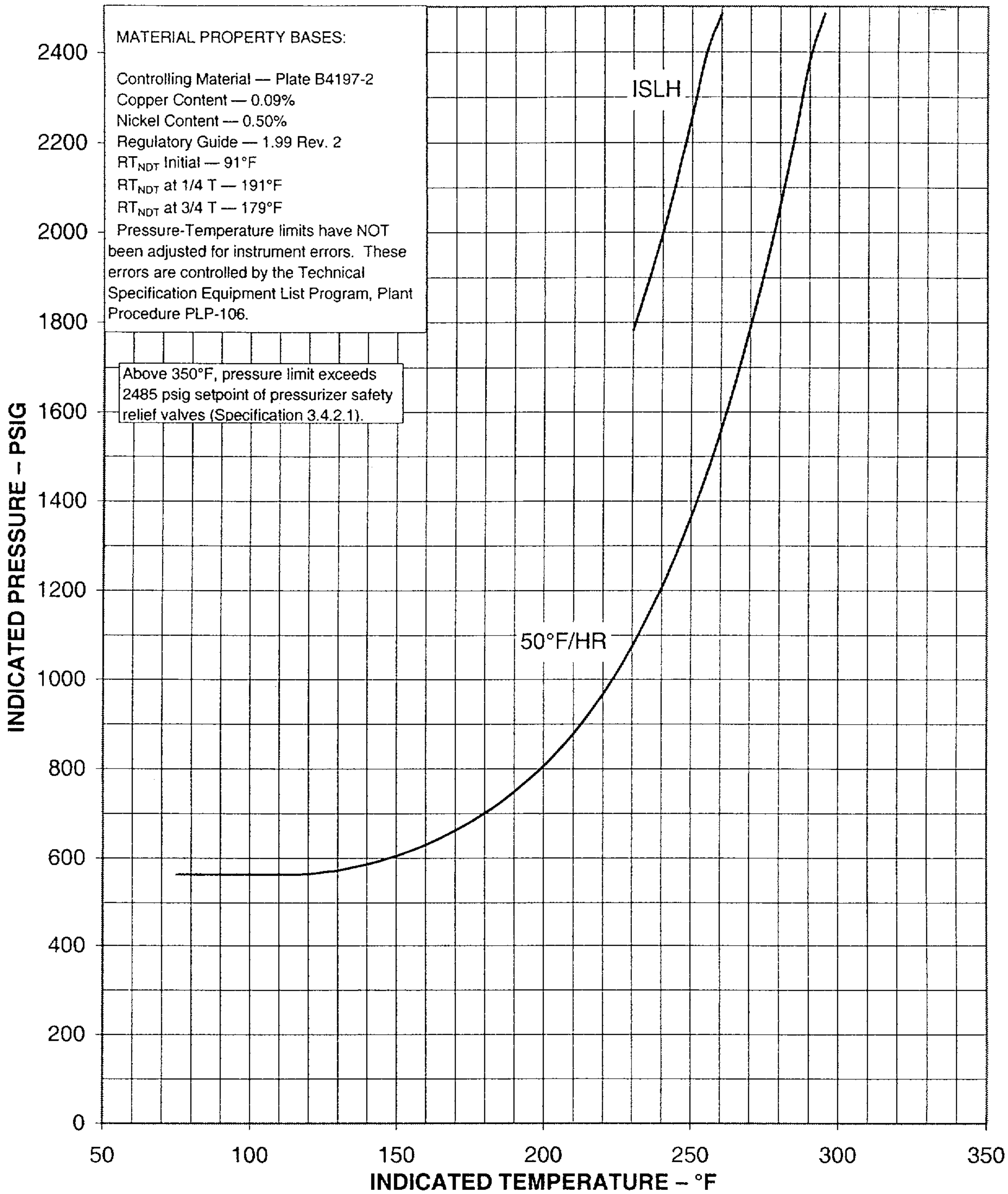

FIGURE 3.4–3
REACTOR COOLANT SYSTEM
HEATUP LIMITATIONS—APPLICABLE UP TO 36 EFPY

TABLE 4.4-6

MAXIMUM COOLDOWN AND HEATUP RATES
FOR MODES 4, 5, AND 6 (WITH REACTOR VESSEL HEAD ON)

COOLDOWN RATES

Delete — 120 — Add

TEMPERATURE*	COOLDOWN IN ANY 1 HOUR PERIOD*
350-155°F	50°F
155-135°F	20°F
135-115°F	10°F
<115°F	5°F/10°F**

Delete

Delete

Delete

Delete

120 — Add

Add — 30

HEATUP RATES

Delete — 350 — Add

Delete — 50 — Add

TEMPERATURE*	HEATUP IN ANY 1 HOUR PERIOD*
<135°F	10°F
135-150°F	20°F
150-175°F	30°F
175-350°F	50°F

Delete

Delete

*Temperature used should be based on lowest RCS cold leg value except when no RCP is in operation; then use an operating RHR heat exchanger outlet temperature.

**10°F/HR cooldown rate may be used if less than three RCPs are operating.

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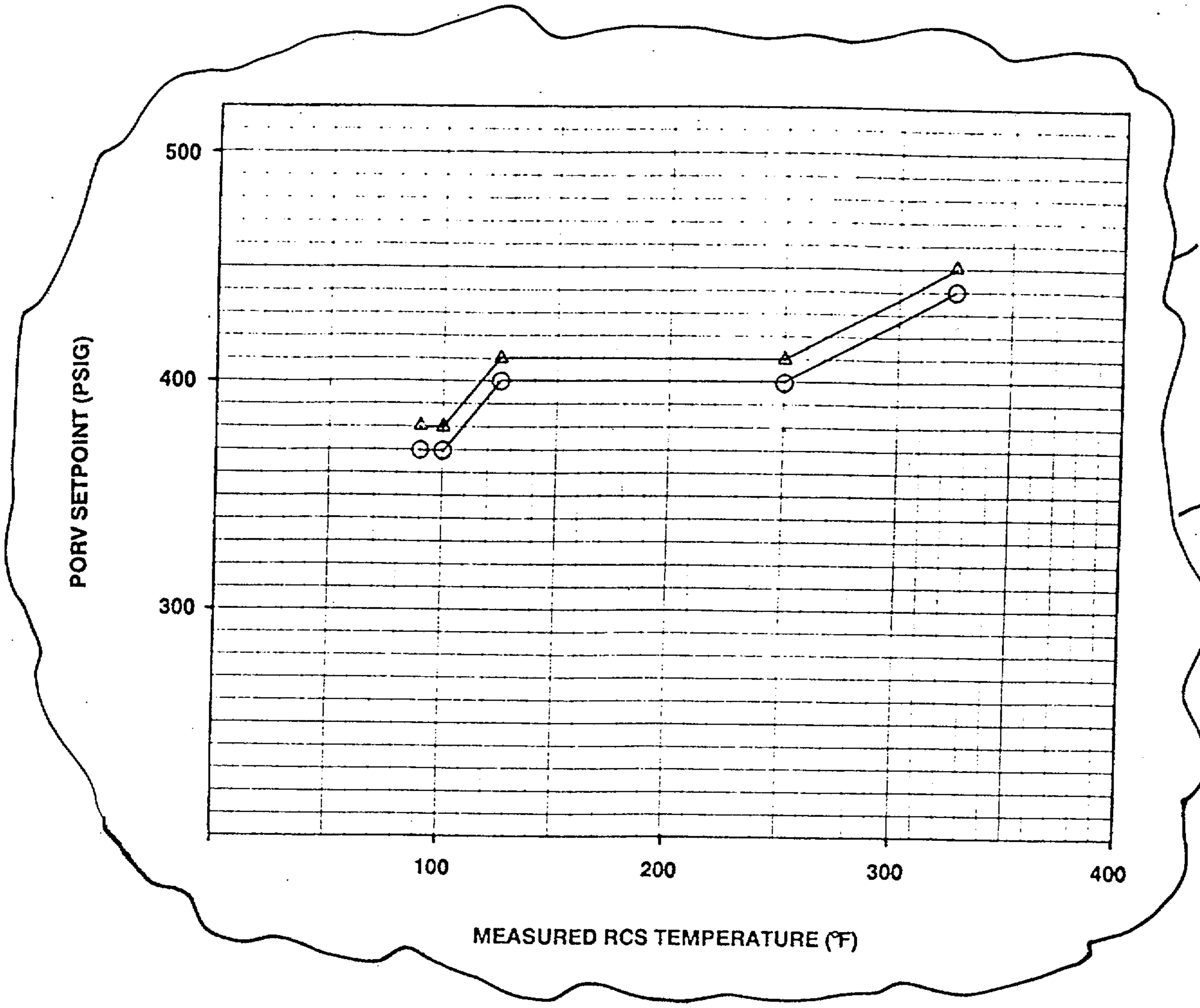

Delete

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RCS TEMP OF	LOW PORV * PSIG ○	HIGH PORV * PSIG Δ
90	Delete – 370 400 - Add	380 410 - Add
100	370	380
125	400	410 - Delete
250	400	410
300	427	437 - Delete
325	440	450

* VALUES BASED ON 11 (Delete) 36 (Add) EFPY REACTOR VESSEL DATA.

Add 1 Delete

INSTRUMENT ERRORS ARE CONTROLLED BY THE TECHNICAL SPECIFICATION EQUIPMENT LIST PROGRAM, PLANT PROCEDURE PLP-106.

FIGURE 3.4-4

MAXIMUM ALLOWED PORV SETPOINT FOR THE LOW TEMPERATURE OVERPRESSURE SYSTEM

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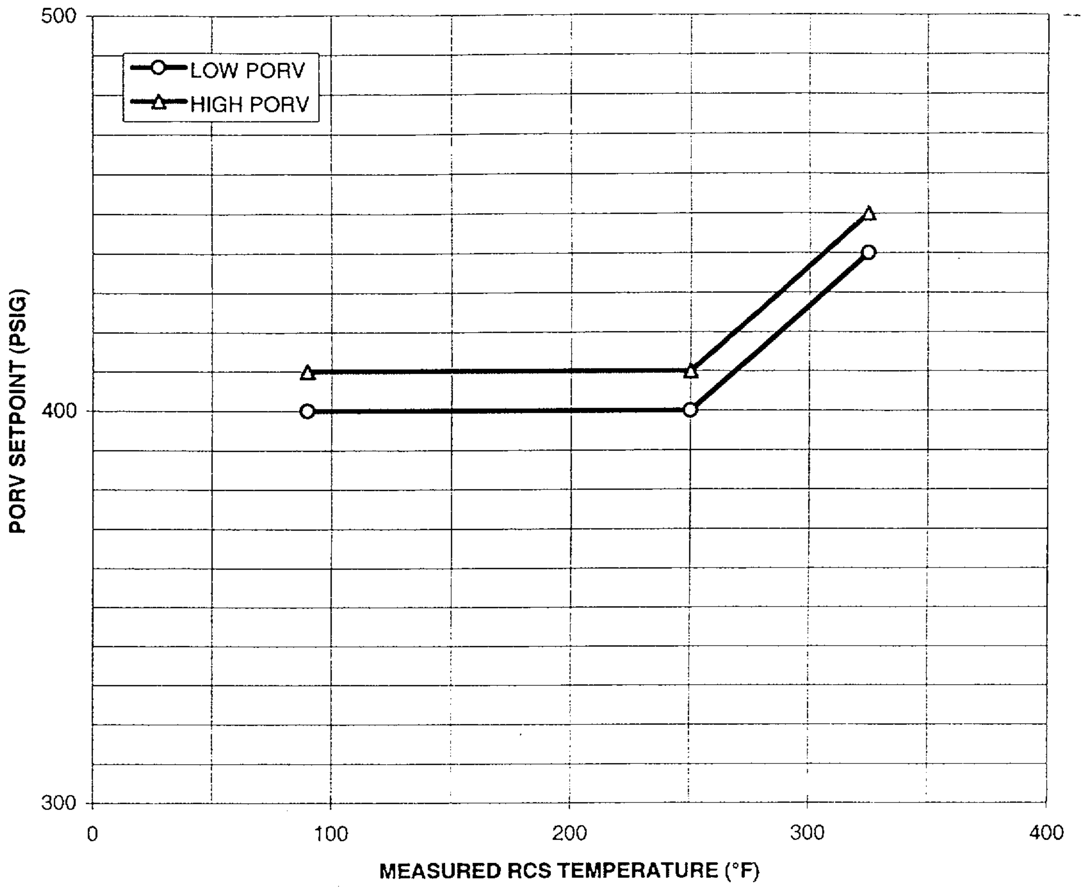

LOW PORV
HIGH PORV
500
400
300
PORV SETPOINT (PSIG)
0
100
200
300
400
MEASURED RCS TEMPERATURE (°F)

SPECIFIC ACTIVITY (Continued)

distinction between the radionuclides above and below a half-life of 15 minutes. For these reasons the radionuclides that are excluded from consideration are expected to decay to very low levels before they could be transported from the reactor coolant to the SITE BOUNDARY under any accident condition.

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. After 90 minutes, the gross count should be made in a reproducible geometry of sample and counter having reproducible beta or gamma self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides. The radiochemical determination of nuclides should be based on multiple counting of the sample within typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture occur, since the saturation pressure of the reactor 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 reactor coolant will be detected in sufficient time to take corrective action. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

Delete

Insert A

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, and 10 CFR 50 Appendix G and H. 10 CFR 50, Appendix G also addresses the metal temperature of the closure head flange and vessel flange regions. The minimum metal temperature of the closure flange region should be at least 120°F higher than the limiting RT NDT for these regions when the pressure exceeds 20% (621 psig for Westinghouse plants) of the preservice hydrostatic test pressure. For Shearon Harris Unit 1, the minimum temperature of the closure flange and vessel flange regions is 120°F because the limiting RT NDT is 0°F (see Table B 3/4 4-1). The Shearon Harris Unit 1 cooldown and heatup limitations shown in Figures 3.4-2 and 3.4-3 and Table 4.4-6 are not impacted by the 120°F limit.

1. The reactor coolant temperature and pressure and system cooldown and heatup rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 and Table 4.4-6 for the service period specified thereon:

   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; and

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INSERT A

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, ASME Code Case N-640, and 10 CFR 50 Appendices G and H.

Appendix G to 10 CFR 50 provides criteria for the metal temperature of the closure head flange and vessel flange regions. ASME Code Case N-640 has prompted a proposed change to 10CFR50 Appendix G to eliminate the closure head flange and vessel flange regions from consideration in developing pressure-temperature limits. Westinghouse Electric Company Report WCAP-15315, Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants, October 1999, demonstrates that the KIC reference fracture toughness curve, permitted by ASME Code Case N-640, provides adequate safety margin against failure of the closure head/vessel flange region, and provided the basis for elimination of this temperature requirement.

PRESSURE/TEMPERATURE LIMITS (Continued)

b. Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4. The pressurizer heatup and cooldown rates shall not exceed 100°F/h and 200°F/h, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 625°F, and

5. System preservice hydrotests and inservice leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness testing of the ferritic materials in the reactor vessel was performed in accordance with the 1971 Winter Addenda to Section III of the ASME Boiler and Pressure Vessel Code. These properties are then evaluated in accordance with the NRC Standard Review Plan.

Add 36 Delete

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 11 effective full power years (EFPY) of service life. The service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements. Delete

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, copper content, and nickel content of the material in question, can be predicted using Figure B 3/4.4-1 and the value of ΔRT_{NDT}, including margin, computed by Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." Delete 1

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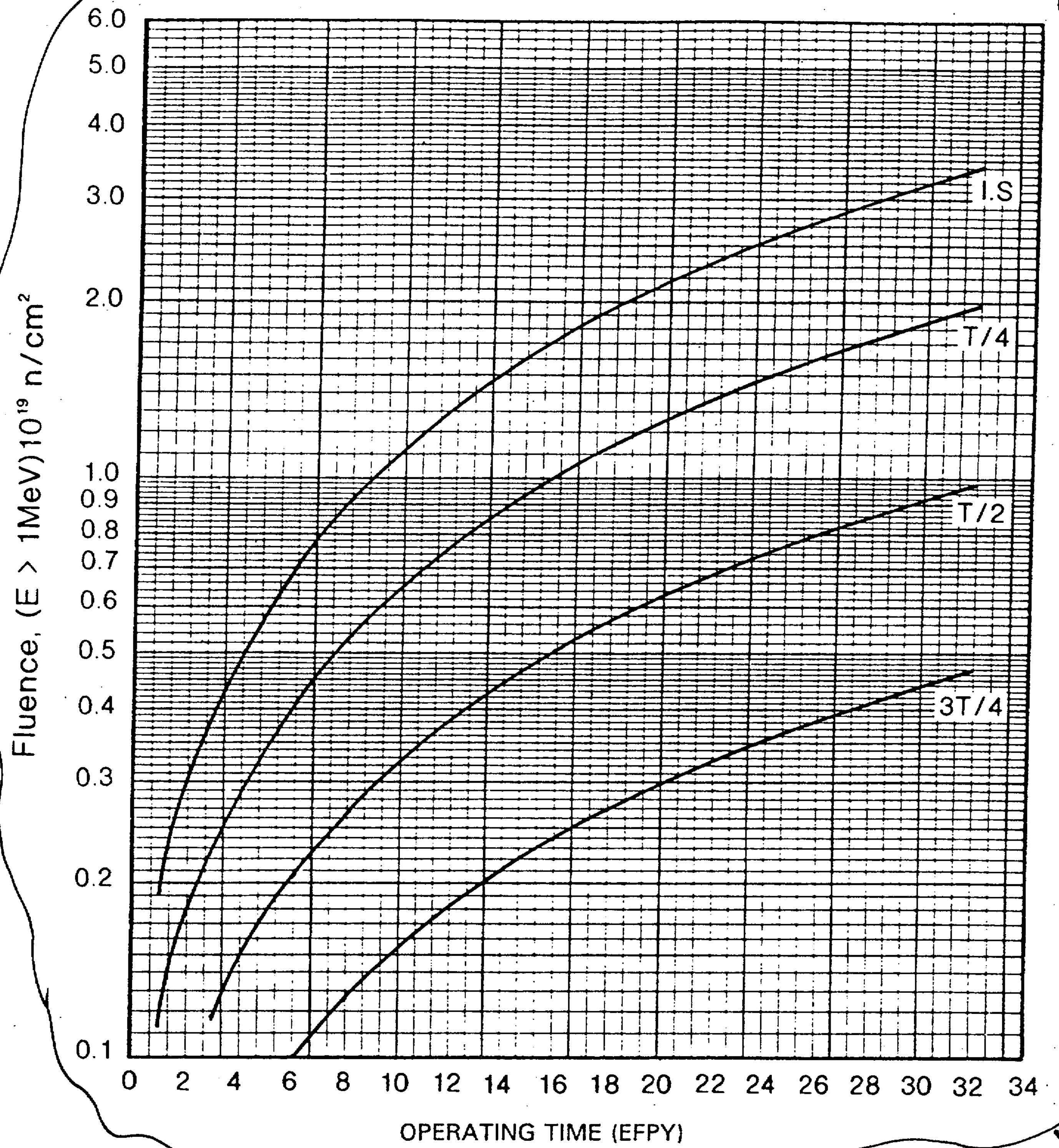

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FIGURE B 3/4.4-1

FAST NEUTRON FLUENCE (E>1MeV) AS A FUNCTION OF FULL POWER SERVICE LIFE

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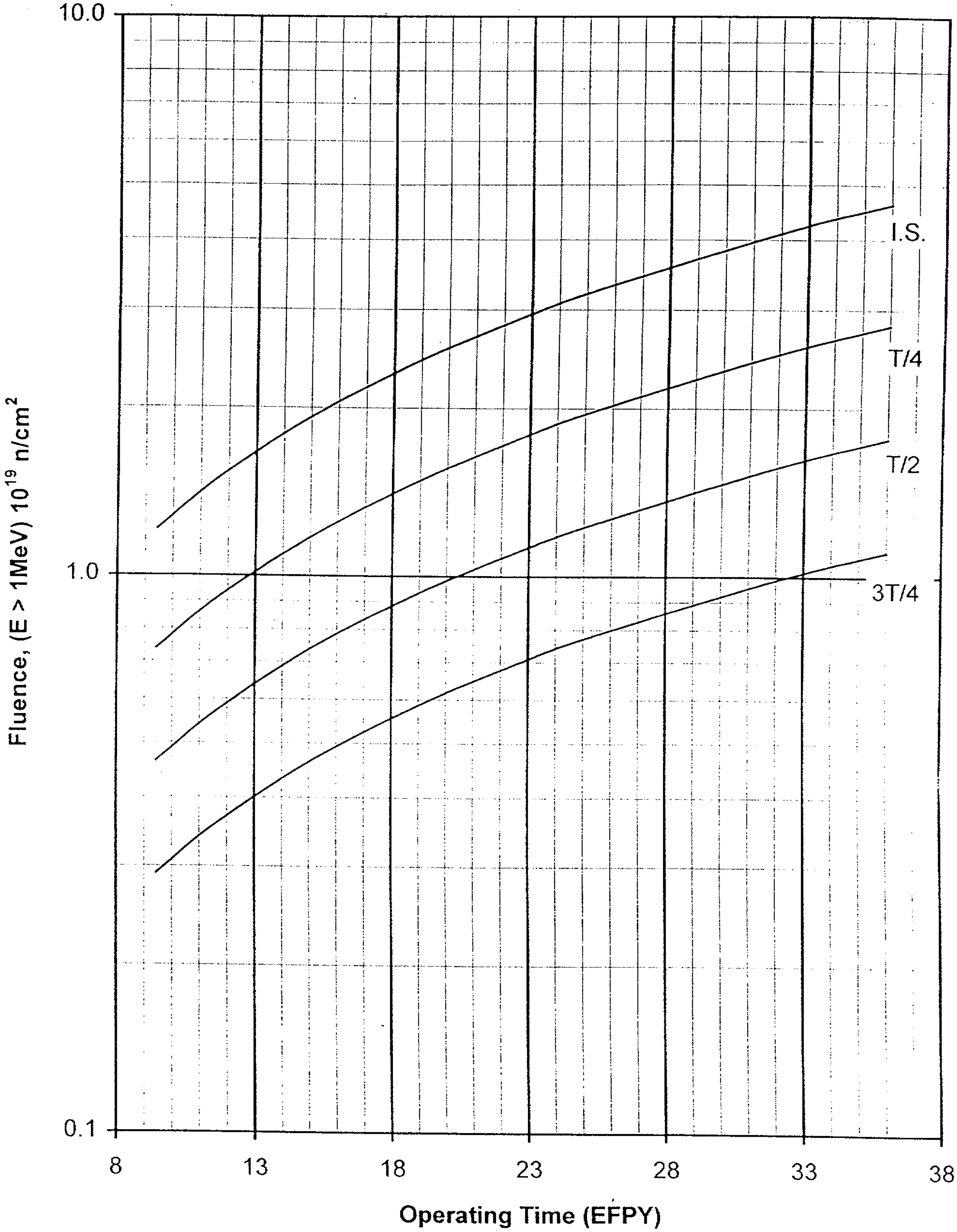

FIGURE B 3/4.4-1
FAST NEUTRON FLUENCE (E>1MeV) AS A FUNCTION OF FULL POWER SERVICE LIFE

PRESSURE/TEMPERATURE LIMITS (Continued)

The cooldown and heatup limits of Figures 3.4-2 and 3.4-3 are based upon an adjusted RT_{NDT} (initial RT_{NDT} plus predicted adjustments for this shift in RT_{NDT} plus margin).

In accordance with Regulatory Guide 1.99, Revision 2, the results from the material surveillance program, evaluated according to ASTM E185, may be used to determine ΔRT_{NDT} when two or more sets of credible surveillance data are available. Capsules will be removed and evaluated in accordance with the requirements of ASTM E185-82 and 10 CFR Part 50, Appendix H. The results obtained from the surveillance specimens can be used to predict future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. The cooldown and heatup curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Delete 1

Allowable pressure-temperature relationships for various cooldown and heatup rates are calculated using methods derived from Appendix G in Section XI of ~~the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50~~. Add and ASME Code Case N-640 for the reactor vessel controlling material.
Delete

The general method for calculating heatup ~~and cooldown limit curves~~ for the beltline shell region Add is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures, a semielliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is ~~assumed to exist~~ at the ~~inside of the vessel wall as well as at the outside of the vessel wall~~. The dimensions ~~of this~~ postulated cracks referred to in Appendix G of ASME Section XI as ~~the~~ reference flaws amply exceed the current capabilities of inservice inspection ~~techniques~~. Therefore, the reactor operation limit curves developed for ~~this~~ these reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which cooldown and heatup curves are generated.

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The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the

Add

A semielliptical inside corner flaw is assumed for the nozzle regions with a depth of one-quarter of the nozzle belt wall thickness. The inlet nozzle is used in the calculation procedures since the inner radius of this tapered nozzle is larger at the corner than the inner radius of the more tapered outlet nozzle.

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PRESSURE/TEMPERATURE LIMITS (Continued) Insert B

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metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \quad (1)$$

Where: K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil-ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\ K_{IM} + K_{It} \leq K_{IR} \quad (2)$$

Where: K_{IM} = the stress intensity factor caused by membrane (pressure) stress,

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

K_{IR} = constant provided by the Code as a function of temperature relative to the RT_{NDT} of the material,

Delete (1)

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice leak and hydrostatic (ISLH) test operations.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses Delete resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{IT}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

Delete Delete

COOLDOWN Add and the inlet nozzle corner Delete surface Add from equation 2 Add

For the calculation of the allowable pressure versus coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

Add The composite limit curves are developed considering the controlling reactor vessel component, either the beltline shell or the inlet nozzle.

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K_{IR} is obtained from reference fracture toughness curves defined in the ASME Code. Pressure-temperature limits are developed for the vessel using the K_{IR} curve defined in Appendix A to the ASME Code, as permitted by ASME Code Case N-640. For the remaining components of the primary pressure boundary, pressure-temperature limits are based on the K_{IR} curve defined in Appendix G to the ASME Code. The K_{IR} curves are given by the equations:

Vessel regions:

$$K_{IR} = K_{Ic} = 33.2 + 2.806 \exp[0.02(T - RT_{NDT} + 100°F)] \quad (1a)$$

Remaining regions:

$$K_{IR} = K_{Ia} = 26.8 + 1.233 \exp[0.0145(T - RT_{NDT} + 160°F)] \quad (1b)$$

PRESSURE/TEMPERATURE LIMITS (Continued)

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IR} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

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inside surface

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The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

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HEATUP

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4T crack during heatup is lower than the K_{IR} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and different K_{IR}'s for steady-state and finite heatup rates do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of

Add

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

BASES

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

LIMITING CONDITION FOR OPERATION

3.4.9.2 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, and inservice leak and hydrostatic testing with:

a. A maximum heatup rate as shown on Table 4.4-6.

b. A maximum cooldown rate as shown on Table 4.4-6.

c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: MODES 4, 5, and 6 with reactor vessel head on.

ACTION:

With any of the pressure limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; if the pressure and temperature limit lines shown on Figure 3.4-2 and 3.4-3 were exceeded, perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or maintain the RCS T_{avg} and pressure at less than 200°F and 500 psig, respectively.

SURVEILLANCE REQUIREMENTS

4.4.9.2.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.2.2 Deleted from Technical Specifications. Refer to the Technical Specification Equipment List Program, plant procedure PLP-106.

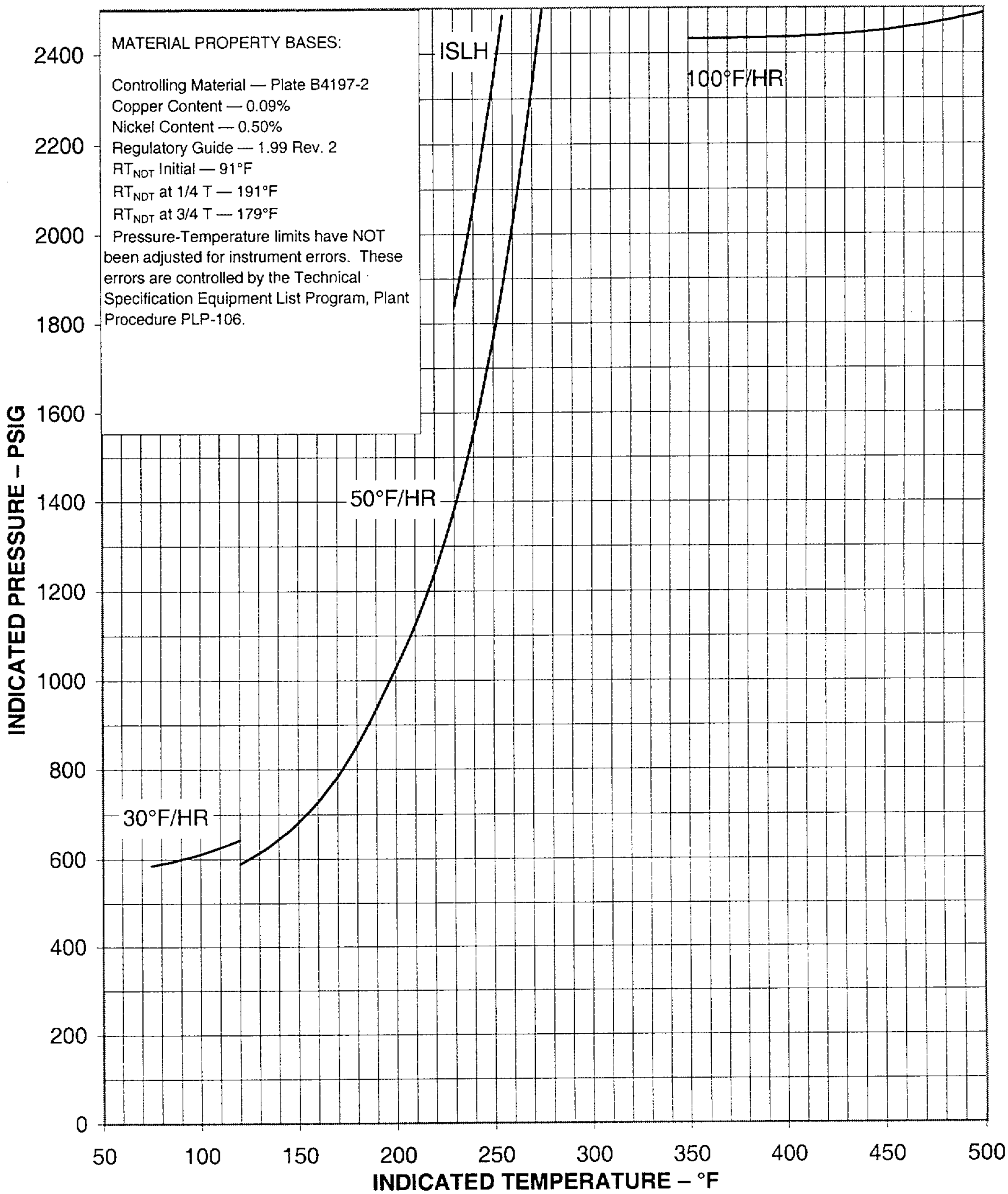


FIGURE 3.4–2
REACTOR COOLANT SYSTEM
COOLDOWN LIMITATIONS—APPLICABLE UP TO 36 EFPY

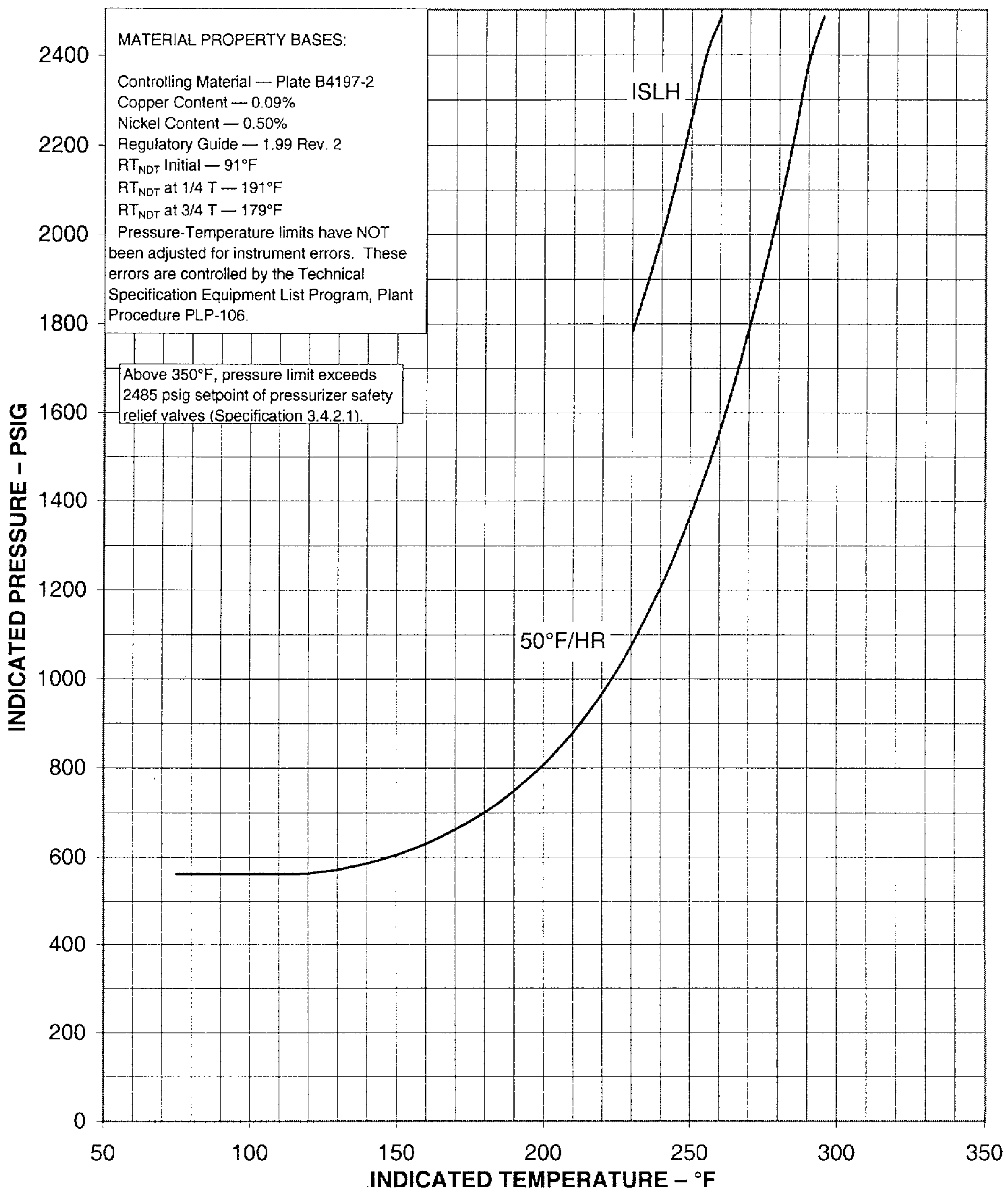


FIGURE 3.4–3
REACTOR COOLANT SYSTEM
HEATUP LIMITATIONS—APPLICABLE UP TO 36 EFPY

TABLE 4.4-6

MAXIMUM COOLDOWN AND HEATUP RATES FOR MODES 4, 5, AND 6 (WITH REACTOR VESSEL HEAD ON)

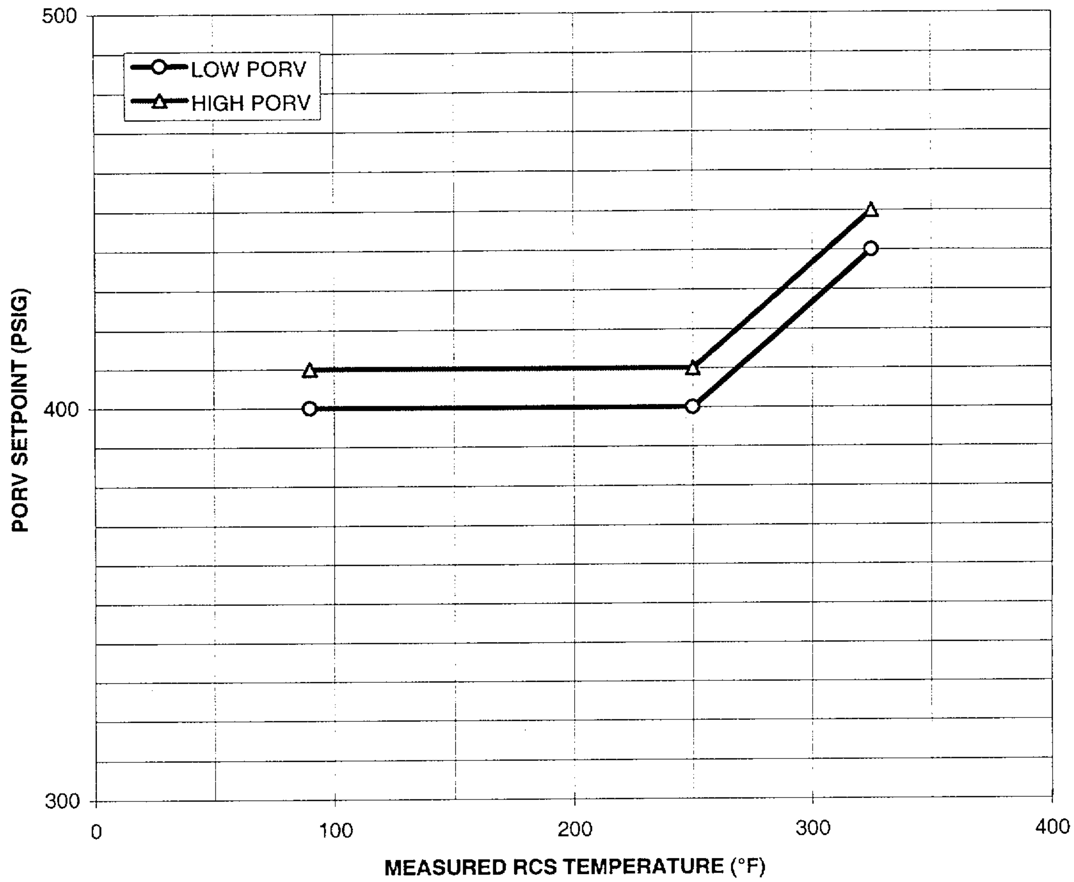
COOLDOWN RATES

TEMPERATURE*	COOLDOWN IN ANY 1 HOUR PERIOD*
350-120°F	50°F
< 120°F	30°F

HEATUP RATES

TEMPERATURE*	HEATUP IN ANY 1 HOUR PERIOD*
<350°F	50°F

*Temperature used should be based on lowest RCS cold leg value except when no RCP is in operation; then use an operating RHR heat exchanger outlet temperature.



RCS TEMP (°F)	LOW PORV* (psig)	HIGH PORV* (psig)
90	400	410
250	400	410
325	440	450

* VALUES BASED ON 36 EFPY REACTOR VESSEL DATA

INSTRUMENT ERRORS ARE CONTROLLED BY THE TECHNICAL SPECIFICATION EQUIPMENT LIST PROGRAM, PLANT PROCEDURE PLP-106.

FIGURE 3.4–4

MAXIMUM ALLOWED PORV SETPOINT FOR THE LOW TEMPERATURE OVERPRESSURE PROTECTION SYSTEM

SPECIFIC ACTIVITY (Continued)

distinction between the radionuclides above and below a half-life of 15 minutes. For these reasons the radionuclides that are excluded from consideration are expected to decay to very low levels before they could be transported from the reactor coolant to the SITE BOUNDARY under any accident condition.

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. After 90 minutes, the gross count should be made in a reproducible geometry of sample and counter having reproducible beta or gamma self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides. The radiochemical determination of nuclides should be based on multiple counting of the sample within typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture occur, since the saturation pressure of the reactor 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 reactor coolant will be detected in sufficient time to take corrective action. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, ASME Code Case N-640, and 10 CRF 50 Appendices G and H.

Appendix G to 10 CFR 50 provides criteria for the metal temperature of the closure head flange and vessel flange regions. ASME Code Case N-640 has prompted a proposed change to 10CFR 50 Appendix G to eliminate the closure head flange and vessel flange regions from consideration in devleoping pressure-temperature limits. Westinghouse Electric Company Report WCAP-15315, Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants, October 1999, demonstrates that the K_{IC} reference fracture toughness curve, permitted by ASME Code Case N-640, provides adequate safety margin against failure of the closure head/vessel flange region, and provided the basis for elimination of this temperature requirement.

1. The reactor coolant temperature and pressure and system cooldown and heatup rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 and Table 4.4-6 for the service period specified thereon:

   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; and

PRESSURE/TEMPERATURE LIMITS (Continued)

b. Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4. The pressurizer heatup and cooldown rates shall not exceed 100°F/h and 200°F/h, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 625°F, and

5. System preservice hydrotests and inservice leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness testing of the ferritic materials in the reactor vessel was performed in accordance with the 1971 Winter Addenda to Section III of the ASME Boiler and Pressure Vessel Code. These properties are then evaluated in accordance with the NRC Standard Review Plan.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 36 effective full power years (EFPY) of service life.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, copper content, and nickel content of the material in question, can be predicted using Figure B 3/4.4-1 and the value of ΔRT_{NDT}, including margin, computed by Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials."

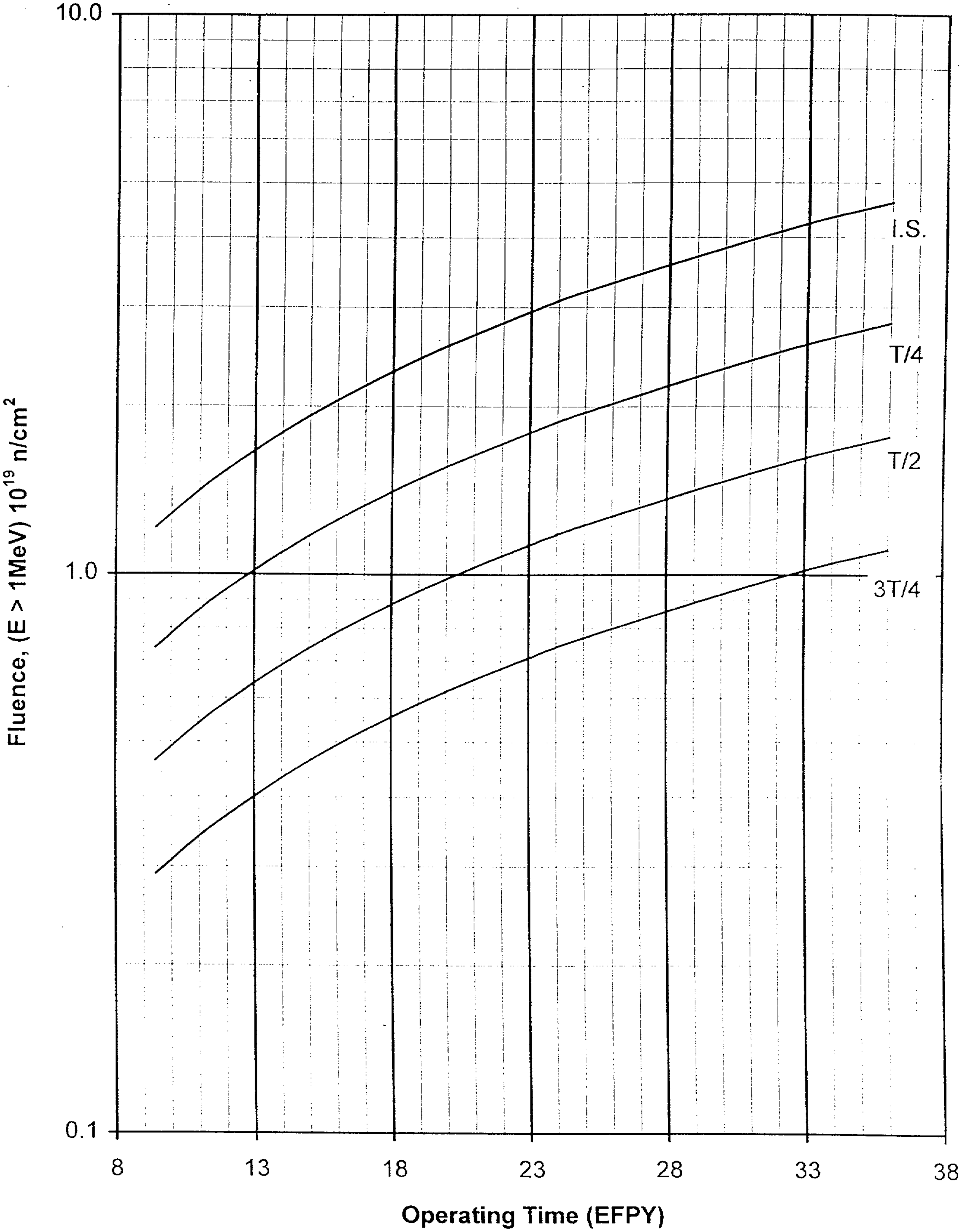


FIGURE B 3/4.4-1
FAST NEUTRON FLUENCE (E>1MeV) AS A FUNCTION OF FULL POWER SERVICE LIFE

PRESSURE/TEMPERATURE LIMITS (Continued)

The cooldown and heatup limits of Figures 3.4-2 and 3.4-3 are based upon an adjusted RT_{NDT} (initial RT_{NDT} plus predicted adjustments for this shift in RT_{NDT} plus margin).

In accordance with Regulatory Guide 1.99, Revision 2, the results from the material surveillance program, evaluated according to ASTM E185, may be used to determine ΔRT_{NDT} when two or more sets of credible surveillance data are available. Capsules will be removed and evaluated in accordance with the requirements of ASTM E185-82 and 10 CFR Part 50, Appendix H. The results obtained from the surveillance specimens can be used to predict future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. The cooldown and heatup curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various cooldown and heatup rates are calculated using methods derived from Appendix G in Section XI of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50 and ASME Code Case N-640 for the reactor vessel controlling material.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures for the beltline shell region a semielliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. A semielliptical inside corner flaw is assumed for the nozzle regions with a depth of one-quarter of the nozzle belt wall thickness. The inlet nozzle is used in the calculation procdures since the inner radius of this tapered nozzle is larger at the corner than the inner radius of the more tapered outlet nozzle. The dimensions of these postulated cracks, referred to in Appendix G of ASME Section XI as reference flaws, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which cooldown and heatup curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the

PRESSURE/TEMPERATURE LIMITS (Continued)

metal temperature at that time. K_{IR} is obtained from reference fracture toughness curves defined in the ASME Code. Pressure-temperature limits are developed for the vessel using the K_{IR} curve defined in Appendix A to the ASME Code, as permitted by ASME Code Case N-640. For the remaining components of the primary pressure boundary, pressure-temperature limits are based on the K_{IR} curve defined in Appendix G to the ASME Code. The K_{IR} curves are given by the equations:

Vessel regions:

$$K_{IR} = K_{Ic} = 33.2 + 2.806 \exp [0.02(T-RT_{NDT} + 100°F)] \qquad (1a)$$

Remaining regions:

$$K_{IR} = K_{Ia} = 26.8 + 1.233 \exp [0.0145(T-RT_{NDT} + 160°F)] \qquad (1b)$$

Where: K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil-ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\ K_{IM} + K_{It} \leq K_{IR} \qquad (2)$$

Where: K_{IM} = the stress intensity factor caused by membrane (pressure) stress,

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

K_{IR} = constant provided by the Code as a function of temperature relative to the RT_{NDT} of the material,

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice leak and hydrostatic (ISLH) test operations.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the wall are calculated and then the corresponding thermal stress intensity factor, K_{IT}, for the reference flaw is computed. The pressure stress intensity factors are obtained and allowable pressures are calculated from equation 2.

COOLDOWN

For the calculation of the allowable pressure versus coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall and the inlet nozzle corner. During cooldown, the controlling location of the flaw is always at the inside surface because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest. The composite limit curves are developed considering the controlling reactor vessel component, either the beltline shell or the inlet nozzle.

PRESSURE/TEMPERATURE LIMITS (Continued)

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T inside surface location is at a higher temperature than the fluid adjacent to the inside surface. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IR} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

HEATUP

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside surface. The thermal gradients during heatup produce compressive stresses at the inside surface that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4T crack during heatup is lower than the K_{IR} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and different K_{IR}'s for steady-state and finite heatup rates do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of