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December 19, 2000
BVY 00-113

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: **Vermont Yankee Nuclear Power Station**
License No. DPR-28 (Docket No. 50-271)
Technical Specification Proposed Change No. 244
Revised P/T Curves and
Exemption Request to use Code Cases N-588 and N-640

Pursuant to 10CFR50.90, Vermont Yankee (VY) hereby proposes to amend its Facility Operating License, DPR-28, by incorporating the attached proposed change into the VY Technical Specifications (TS). The proposed change would revise the reactor vessel pressure/temperature (P/T) limit curves specified in TS 3.6.A.1, "Reactor Coolant Systems - Pressure and Temperature Limitations," as graphically represented in Figure 3.6.1, for reactor heatup, cooldown, and critical operation, as well as for inservice hydrostatic and leak tests.

In addition, VYNPC is requesting an exemption from the requirements of 10CFR50, Appendix G, to allow the use of ASME Code Cases N-588 and N-640 as the basis for the revised P/T curves. The proposed P/T curves were developed in accordance with 1995 ASME Code, Section XI, Appendix G (including the Summer 1996 Addenda); 10CFR50 Appendix G; and ASME Code Cases N-588 and N-640. The use of the Code Cases as the basis for the proposed P/T curves constitutes an alternative to the requirements of 10CFR50 Appendix G. 10CFR50.60 (b) provides that the NRC may grant an alternative to these requirements using the procedures for exemption specified in 10CFR50.12.

Application of the revised P/T limits is desired for the forthcoming refueling outage which is scheduled to commence on April 28, 2001. Since a significant reduction in critical path time can be realized by application of the revised P/T limits (due to the reduced heatup and test time associated with the reactor vessel pressure/leak test), VY respectfully requests NRC review and approval of the requested amendment and exemption by April 1, 2001.

Attachment 1 to this letter contains supporting information and the safety assessment of the proposed change. Attachment 2 contains the determination of no significant hazards consideration. Attachment 3 provides the marked-up version of the current Technical Specification and Bases pages. Attachment 4 is the retyped Technical Specification and Bases pages. Attachment 5 contains the Request for Exemption from the requirements of 10CFR50, Appendix G. Attachment 6 provides the Technical Report in support of the revised P/T limits.

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VY has reviewed the proposed Technical Specification change in accordance with 10CFR50.92 and concludes that the proposed change does not involve a significant hazards consideration.

VY has also determined that the proposed change satisfies the criteria for a categorical exclusion in accordance with 10CFR51.22(c)(9) and does not require an environmental review. Therefore, pursuant to 10CFR51.22(b), the preparation of an environmental impact statement or environmental assessment is not warranted.

If you have any questions on this transmittal, please contact Mr. Thomas B. Silko at (802) 258-4146.

Sincerely,

VERMONT YANKEE NUCLEAR POWER CORPORATION

Michael A. Balduzzi

Michael A. Balduzzi
Vice President, Operations

STATE OF VERMONT)
)ss
WINDHAM COUNTY)

Then personally appeared before me, Michael A. Balduzzi, who, being duly sworn, did state that he is Vice President, Operations of Vermont Yankee Nuclear Power Corporation, that he is duly authorized to execute and file the foregoing document in the name and on the behalf of Vermont Yankee Nuclear Power Corporation, and that the statements therein are true to the best of his knowledge and belief.

T B Silko

Thomas B. Silko, Notary Public
My Commission Expires February 10, 2003

Attachments

cc: USNRC Region 1 Administrator
USNRC Resident Inspector - VYNPS
USNRC Project Manager - VYNPS
Vermont Department of Public Service

Attachment 1

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 244

Revised P/T Curves

Supporting Information and Safety Assessment of Proposed Change

SUPPORTING INFORMATION

Purpose

The proposed change would revise the reactor vessel pressure/temperature (P/T) limit curves specified in TS 3.6.A.1, "Reactor Coolant System - Pressure and Temperature Limitations," as graphically represented in Figure 3.6.1, for reactor heatup, cooldown, and critical operation, as well as for inservice hydrostatic and leak tests. Figure 3.6.1 would be replaced with three figures:

Figure 3.6.1, "Reactor Vessel Pressure-Temperature Limitations: Hydrostatic Pressure and Leak Tests, Core Not Critical,"
Figure 3.6.2, "Reactor Vessel Pressure-Temperature Limitations: Normal Operation, Core Not Critical," and
Figure 3.6.3, "Reactor Vessel Pressure-Temperature Limitations: Normal Operation, Core Critical."

No changes to the Limiting Condition for Operation or any Surveillance Requirements are proposed.

The revised P/T limits, as proposed, would yield several benefits. A primary effect of the revised limits is to allow required reactor vessel hydrostatic and leak tests to be performed at a lower temperature. This can significantly reduce critical path time associated with such testing during refueling outages by reducing or eliminating the heatup time required to achieve required test conditions. The safety benefits that may result from this effect include a reduction in the challenges to plant operators associated with maintaining the reactor coolant system (RCS) at higher test temperatures and/or within a narrow temperature band, reduced challenges to personnel safety for inspectors due to lower ambient drywell temperatures, reduced dose to inspectors due to increased inspection effectiveness at the lower ambient drywell temperatures, and increased availability of systems connected to the RCS (including the Residual Heat Removal System) because of a reduced heatup and test duration.

Background

The current pressure-temperature limits specified in TS Figure 3.6.1 are represented by several curves on a single figure, for the various operating and/or test conditions. The current curves are to be replaced with recalculated curves on separate figures, and the associated descriptions contained on the figures are to be revised as well. Revised TS Figure 3.6.1 will have a curve for the bottom head region of the vessel and a composite RCS curve (excluding the bottom head) for hydrostatic testing and leak testing conditions for an exposure level equivalent to a gross power generation of $4.46x10^8$MWH(t) (which will bound VY power generation beyond March 21, 2012, the end of VY's current operating license (EOL) and is equivalent to 32 effective full power years (EFPY)). Figure 3.6.2 will have a curve for the bottom head region of the vessel and the composite RCS curve (excluding the bottom head) for non-critical operation for up to $4.46x10^8$MWH(t). Figure 3.6.3 will have a curve for the bottom head region of the vessel and the composite RCS curve (excluding the bottom head) for reactor critical operation for up to $4.46x10^8$MWH(t). These curves for specifying the required temperature limits will continue to ensure margin to the brittle fracture temperature, i.e., the nil ductility temperature (NDT), for the noted operations or conditions. One of the primary effects of the revised curves is to permit reactor vessel inservice hydrostatic and leak tests to be performed at a lower temperature at applicable vessel pressures.

The revised P/T limits are based, in part, on application of American Society of Mechanical Engineers (ASME) Code Cases N-588, "Alternative to Reference Flaw Orientation of Appendix G for

Circumferential Welds in the Reactor Vessels" and N-640, "Alternative to Requirement Fracture Toughness for Development of P/T Limit Curves for ASME B&PV Code Section XI, Division 1." These code cases provide alternative methods to those currently approved by the NRC and recognized per 10CFR50.60. Accordingly, the use and acceptability of these alternative methods requires an exemption from 10CFR50.60 requirements.

SAFETY ASSESSMENT

Recalculation of the P/T curves involves application of certain attributes such as Code Cases N-640 and N-588, beltline ART_{NDT}, and instrument uncertainty. Each of these items are discussed further below.

Code Case N-640

This Code Case allows use of the K_{IC} fracture toughness curve shown on ASME Code, Section XI, Appendix A, Figure A-4200-1, in lieu of the K_{IA} fracture toughness curve of ASME Code, Section XI, Appendix G, Figure G-2210-1, as the lower bound for fracture toughness in the development of the P/T limit curves. The other margins involved with the ASME Code, Section XI, Appendix G, process of determining P/T limits remain the same.

Use of Code Case N-640 is justified based upon the knowledge gained in the industry since the fracture toughness curve was created in 1974. Since that time, additional knowledge of the fracture toughness of materials and their response to applied loads has been acquired. This additional knowledge demonstrates the lower bound fracture toughness provided by the K_{IA} curve is well beyond the margin of safety required to protect against potential reactor pressure vessel (RPV) failure. The K_{IC} curve provides an adequate margin of safety as discussed below.

Use of the K_{IC} fracture toughness curve in developing P/T limits provides additional operating margin for the P/T curves, thus realizing benefits primarily for pressure tests. For example, the lower temperature requirement significantly shortens the duration of the pressure test. Additionally, a personnel safety benefit is provided while conducting inspections of the primary containment at elevated temperatures. It is reasonable to expect that, during the pressure tests, the drywell inspectors are more effective due to the decreased ambient temperatures. Additional justification for the use of Code Case N-640 is provided in Attachment 5.

Code Case N-588

This Code Case provides relief from the specific requirement of 10CFR50, Appendix G, that Article G-2120 of ASME Code, Section XI, Appendix G, be used to determine the maximum postulated defects in the RPV for the determination of P/T limits. Article G-2120 specifies that the postulated defect be in the surface of the material and normal to the direction of maximum stress. Appendix G also provides a methodology for determining the stress intensity factors for this maximum postulated defect. The purpose of the article is to ensure the prevention of nonductile fractures by providing procedures to identify the most limiting postulated fractures considered in developing the P/T limits.

Code Case N-588 revises the Article G-2120 reference flaw orientation for circumferential welds in RPVs by eliminating certain unrealistic and overly conservative assumptions. The Code Case essentially recognizes procedures and controls in the fabrication process of reactor vessels designed to minimize defects that can be introduced into the weld during the fabrication process. Also, industry experience with the repair weld indications found during preservice inspection and data taken from

nondestructive and destructive examinations were considered in developing Code Case N-588. Additional justification for the use of Code Case N-588 is provided in Attachment 5.

In the revised P/T limits, Code Case N-588 was used to calculate the thermal stress intensity factor (K_{IT}) for the beltline and lower head regions as well as pressure stress intensity (K_{IP}) for the beltline. In the Vermont Yankee vessel, weld initial RT_{NDT} and beltline weld projected ART_{NDT} are much lower than critical pressure boundary plates and forgings. Therefore the Code Case N-588 rules for flaw orientation in circumferential welds had no impact to the proposed Vermont Yankee reactor P/T limits.

Beltline ART_{NDT} through 4.46×10^8 MWH(t)

In a previous P/T limit License Amendment Request[1] the ART_{NDT} was developed using an initial RT_{NDT} of 40°F with a conservatively modified fluence factor curve from RG 1.99, Rev. 2 (current Tech Spec Figure 3.6.3). This resulted in a conservatively adjusted RT_{NDT}, (ART_{NDT}) of 89°F at the 1/4T and 73°F at the 3/4T points. In the NRC's Safety Evaluation[2] of our previous submittal, the Staff concluded that "The licensee used a more conservative safety factor than the one in RG 1.99, Rev. 2... The Staff believes that an ART of 63.1°F [at the 1/4T point] is sufficient to protect the reactor vessel from embrittlement."

Based on guidance from Branch Technical Position MTEB 5-2, Structural Integrity (Attachment 6) has performed an assessment of initial RT_{NDT} for vessel pressure boundary components. The chemistry information for beltline plate and weld material as shown in the NRC's Reactor Vessel Integrity database has been reaffirmed as appropriate and used in the Attachment 6 assessment. In Attachment 6, Structural Integrity concluded that VY was overly conservative in our previous assessment of the initial RT_{NDT} for the beltline materials. They concluded that the limiting beltline component had an initial RT_{NDT} of 30°F not 40°F. Using the guidance of RG 1.99, Rev. 2, they have determined ART_{NDT} values of 53°F (1/4T) and 45.4°F (3/4T) would satisfy 10CFR50 Appendix G requirements and provide adequate margin to account for the expected shift due to fluence.

In this proposed change, the P/T evaluation was based on the conservatively calculated ART_{NDT} previously used by VY; 89°F at the 1/4T point and 73°F at the 3/4T point. Maintaining very conservative ART_{NDT}'s provides significant additional margin for beltline region shift due to fluence and shift uncertainty. As demonstrated in Attachment 6, based on the initial RT_{NDT} values and RG 1.99, Rev. 2 criteria for calculating ART_{NDT}, the use of the conservative ART_{NDT} values equate to a minimum end-of-life surface fluence of 1.24×10^{18}n/cm^2 for critical beltline material. This is more than 5 times the peak end-of-life surface fluence calculated for Vermont Yankee by Battelle[3]. The Technical Report in Attachment 6 confirms that plate 1-14, used for the VY surveillance specimens is the critical plate from the standpoint of brittle failure up to fluence levels well beyond that expected.

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1 Reference Vermont Yankee Nuclear Power Corporation letter to the USNRC, BVY 89-113, "Proposed Change to Revise the Reactor Vessel Pressure-Temperature Curves in the Vermont Yankee Technical Specifications (Generic Letter 88-11)," dated November 10, 1989.

2 Reference USNRC Letter to Vermont Yankee Nuclear Power Corporation, NVY 90-77, "Issuance of Amendment No. 120 to Facility Operating License No. DPR-28 - Vermont Yankee Nuclear Power Station (TAC No. 75499)," dated April 17, 1990.

3 Reference Battelle Columbus Laboratories Final Report, BCL-585-84-3, "Examination, Testing, and Evaluation of Irradiated Pressure Vessel Surveillance Specimens from Vermont Yankee Nuclear Power Station," dated May 15, 1984.

In this proposed change, Technical Specification Figure 3.6.2 "Fast Neutron Fluence (E>1 MEV) as a Function of Thermal Energy and Full Power Years" and Figure 3.6.3 "Fluence Factor for use in Regulator Guide 1.99, Revision 2," have been replaced with the P/T curves for Core Not Critical and Core Critical Operation. The current Figure 3.6.2 provides beltline fluence values at the vessel surface as well at 1/4T and 3/4T as a function of full power years. However, these curves are not utilized in the development of our proposed curves. The proposed P/T limits were calculated utilizing an end of life fluence value. Accordingly, the current Figure 3.6.2 is being replaced. Figure 3.6.3 is a copy of the fluence factor curve from RG 1.99, Rev 2. It has an additional line demonstrating VY's conservatism from our previous submittal (see footnote 1). Since these curves were not utilized in the development of our current P/T limits, the figure is being removed from the specifications.

The revised bases section now states, "For these plates and welds an adjusted RT_{NDT} (ART_{NDT}) of 89°F and 73°F (¼ and ¾ thickness locations) was conservatively used in development of these curves for core region components. Based upon plate and weld chemistry, initial RT_{NDT} values, predicted peak fluence (2.3×10^{17} n/cm^2) for a gross power generation of 4.46×10^8 MWH(t) (Battelle Columbus Laboratory Report BCL 585-84-3, dated May 15, 1984) these core region ART_{NDT} values conservatively bound requirements of Regulatory Guide 1.99, Revision 2."

A peak fluence of 2.3×10^{17}n/cm^2 (E>1.0 MeV) was used in VY's previous 1989 P/T submittal (see footnote 1) to generate the conservative beltline ART_{NDT} values of 89°F and 73°F. This fluence value was from the peak EOL fluence of 2.2×10^{17}n/cm^2 (>1.0 MeV) calculated by Battelle with an additional 0.1×10^{17}n/cm^2 added to bound axial fluence variation effects. Since the 1984 Battelle evaluation was performed, core reloads have been designed as low leakage cores and the relative power of the periphery bundles and weighted power affecting fluence is less than assumed in the Battelle Report.

Assessment of Non-Beltline Areas, Instrument Uncertainty, and Anticipated Operational Transients

There were five regions of the reactor pressure vessel (RPV) that are evaluated in the development of the proposed P/T Limit curves: (1) the reactor vessel beltline region, (2) the bottom head region, (3) the feedwater nozzle, (4) the recirculation inlet nozzle, and (5) the upper vessel flange region. These regions will bound all other regions in the vessel with respect to considerations for brittle fracture.

Two lines are shown on each P/T limit figure. The dashed line is the Bottom Head Curve. The bottom head area is subject to lower temperatures than the balance of the pressure vessel. The RT_{NDT} of the lower head is lower than the ART_{NDT} used for the beltline. The lower head area is also not subject to the same high level of stress as the flange and feedwater nozzle regions. The dashed Bottom Head Curve is less restrictive than the enveloping curve used for the Upper Regions of the vessel and provides Operator's with a conservative but less restrictive P/T limit for the cooler bottom head region. The solid line is the Upper Region curve. This line conservatively bounds all regions of the vessel including the most limiting beltline and flange areas.

VY has assessed the instrumentation used to monitor the vessel P/T conditions and has included conservative uncertainty values in the revised curves. Anticipated operating transients including feedwater nozzle evaluation with cold feedwater injection, and a severe scram transient and hot sweep of the bottom head were evaluated.

For further information see Attachment 6.

Summary

VY believes that based upon consideration of the conservatism that is explicitly incorporated into the methodologies of 10CFR50, Appendix G; Appendix G of the Code; and RG 1.99, Rev. 2, that application of the Code Cases as described, would provide an adequate margin of safety against brittle failure of the RPV. This is also consistent with the determination that the Staff has reached for other licensees[4 & 5] under similar conditions based on the same considerations.

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4 Reference USNRC Letter to Commonwealth Edison Company, dated February 4, 2000, "Quad Cities – Exemption from the Requirements of 10CFR Part 50, Section 50.60(a) and Appendix G."

5 Reference USNRC Letter to Commonwealth Edison Company, dated February 4, 2000, "Quad Cities – Issuance of Amendments – Revised Pressure-Temperature Limits."

Attachment 2

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 244

Revised P/T Curves

Determination of No Significant Hazards Consideration

Determination of No Significant Hazards Consideration

Description of amendment request:

The proposed change would revise the reactor vessel pressure/temperature (P/T) limit curves specified in TS 3.6.A.1, "Reactor Coolant System - Pressure and Temperature Limitations," as graphically represented in Figure 3.6.1, for reactor heatup, cooldown, and critical operation, as well as for inservice hydrostatic and leak tests.

Basis for no significant hazards determination:

Pursuant to 10CFR50.92, Vermont Yankee (VY) has reviewed the proposed change and concludes that the change does not involve a significant hazards consideration since the proposed change satisfies the criteria in 10CFR50.92(c).

1. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The changes to the calculational methodology for the P/T limits based upon ASME Code Cases N-640 and N-588 provide adequate margin in the prevention of a brittle-type fracture of the reactor pressure vessel (RPV). The Code Cases were developed based upon the knowledge gained through years of industry experience. The experience gained in the areas of fracture toughness of materials and pre-existing undetected defects show that some of the existing assumptions used for the calculation of P/T limits are unnecessarily conservative and unrealistic. Therefore, providing the allowances of the subject Code Cases in developing the P/T limit curves will continue to provide adequate protection against nonductile-type fractures of the RPV.

   The evaluation for revising the P/T limit curves for 4.46×10^8MWH(t) (32 effective full power years) was performed using the approved methodologies of 10CFR50, Appendix G. The curves generated from these methods ensure the P/T limits will not be exceeded during any phase of reactor operation. The proposed changes will not affect any other system or equipment designed for the prevention or mitigation of previously analyzed events. Thus, the probability of occurrence and the consequences of any previously analyzed event are not significantly increased as the result of the proposed changes.

2. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed changes to the reactor pressure vessel P/T limits do not affect the assumed performance of any system, structure, or component previously evaluated. The proposed changes do not introduce any new modes of system operation or failure mechanisms. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. <u>The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.</u>

Industry experience since the inception of the P/T limits in 1974 confirms that some of the existing methodologies used to develop P/T curves is unnecessarily conservative. Accordingly, ASME Code Cases N-640 and N-588 take advantage of the acquired knowledge by establishing more enhanced methodologies for the development of P/T curves. Therefore, operational flexibility can be gained without a significant reduction in the margin of safety to RPV brittle fracture.

The revised evaluation of the P/T curves to 4.46×10^8MWH(t) was performed per the guidelines of 10CFR50, and thus, the margin of safety is not reduced as the result of the proposed changes.

Attachment 3

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 244

Revised P/T Curves

Marked-up Version of the Current Technical Specifications

3.6 LIMITING CONDITIONS FOR OPERATION

3.6 REACTOR COOLANT SYSTEM

Applicability:

Applies to the operating status of the reactor coolant system.

Objective:

To assure the integrity and safe operation of the reactor coolant system.

Specification:

A. Pressure and Temperature Limitations

1. The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.6.1, 3.6.2 AND 3.6.3, AS APPROPRIATE ~~during heatup, cooldown, criticality (except for the purposes of low-power physics testing), and inservice leak and hydrostatic testing~~.

2. The maximum heatup or cooldown rate is 100°F when averaged over any one hour period.

3. The reactor vessel head bolting shall not be tensioned unless the temperature of the vessel head flange and the head is greater than 70°F.

4. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50°F of each other.

4.6 SURVEILLANCE REQUIREMENTS

4.6 REACTOR COOLANT SYSTEM

Applicability:

Applies to the periodic examination and testing requirements for the reactor coolant system.

Objective:

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification:

A. Pressure and Temperature Limitations

1. The reactor coolant temperature and pressure shall be recorded at least once per hour during system heatup, cooldown and inservice leak and hydrostatic testing operations.

2. The reactor coolant temperature and pressure shall be recorded at the time of reactor criticality.

3. When the reactor vessel head bolting is being tightened or loosened, the reactor vessel shell temperature immediately below the vessel flange shall be permanently recorded.

4. Prior to and after startup of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be recorded.

3.6 LIMITING CONDITIONS FOR OPERATION

Reassess material properties and

B. Coolant Chemistry

1. a. During reactor power operation, the radioiodine concentration in the reactor coolant shall not exceed 1.1 microcuries of I-131 dose equivalent per gram of water, except as allowed in Specification 3.6.B.1.b.

4.6 SURVEILLANCE REQUIREMENTS

5. The reactor vessel irradiation surveillance specimens shall be removed and examined to determine changes in material properties in accordance with the following schedule:

CAPSULE	REMOVAL YEAR
1	10
2	30
3	Standby

3.6.1

The results shall be used to update Figures 3.6.2 and 3.6.3. As appropriate. The removal times shall be referenced to the refueling outage following the year specified, referenced to the date of commercial operation.

B. Coolant Chemistry

1. a. A sample of reactor coolant shall be taken at least every 96 hours and analyzed for radioactive iodines of I-131 through I-135 during power operation. In addition, when steam jet air ejector monitors indicate an increase in radioactive gaseous effluents of 25 percent or 5000 µCi/sec, whichever is greater, during steady state reactor operation a reactor coolant sample shall be taken and analyzed for radioactive iodines.

FIGURE 3.6.1

REACTOR VESSEL PRESSURE-TEMPERATURE LIMITATIONS

Valid Thru 4.46E8 MWH(t) [32 EFPY]

Adjusted Reference Temperature at 1/4T = 89'F
Adjusted Reference Temperature at 3/4T = 73'F
Adjusted Reference Temperature at Flange = 20'F

Curves are in accordance with
10CFR50 App G and RG 1.99 Rev 2

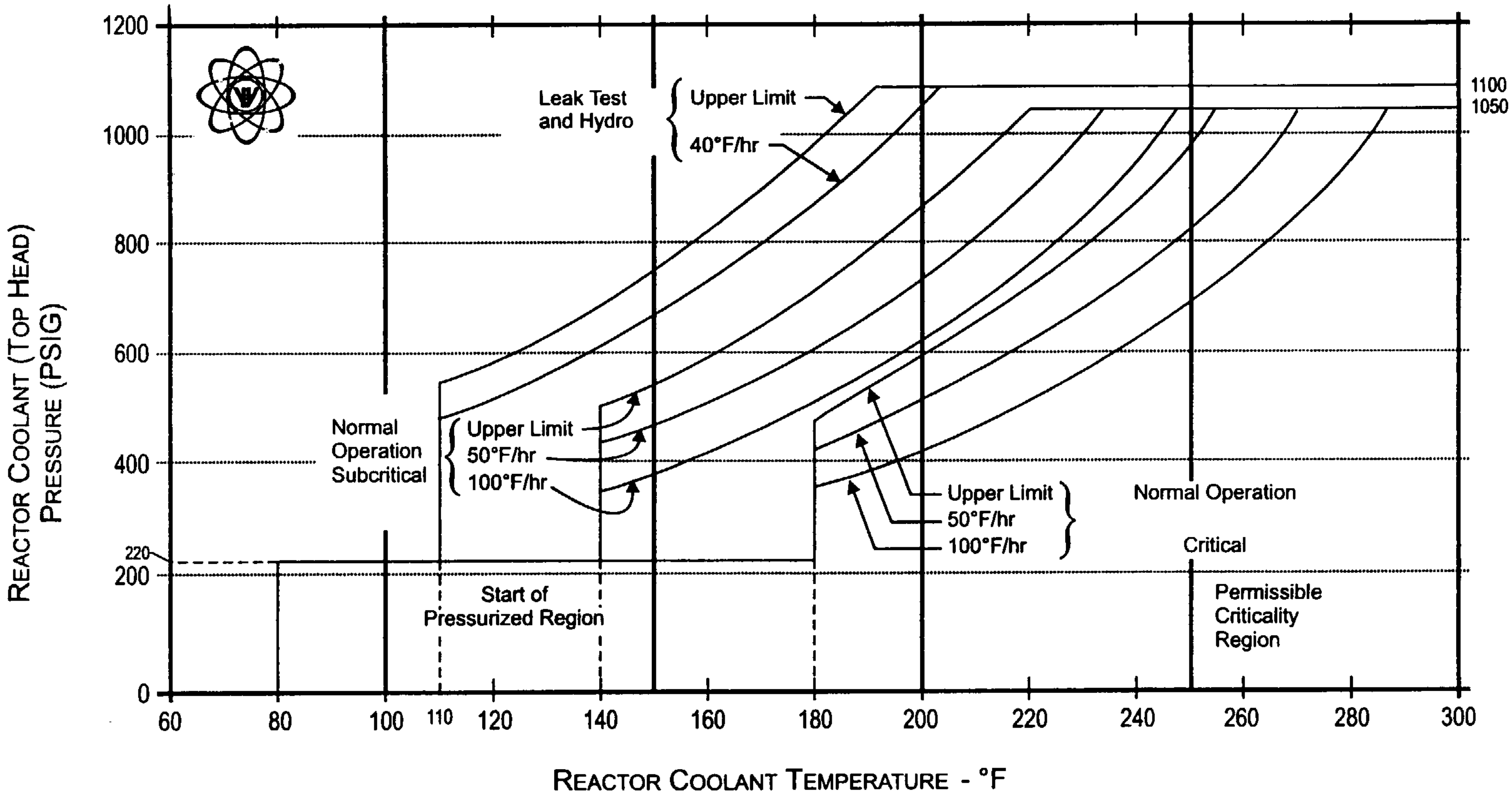

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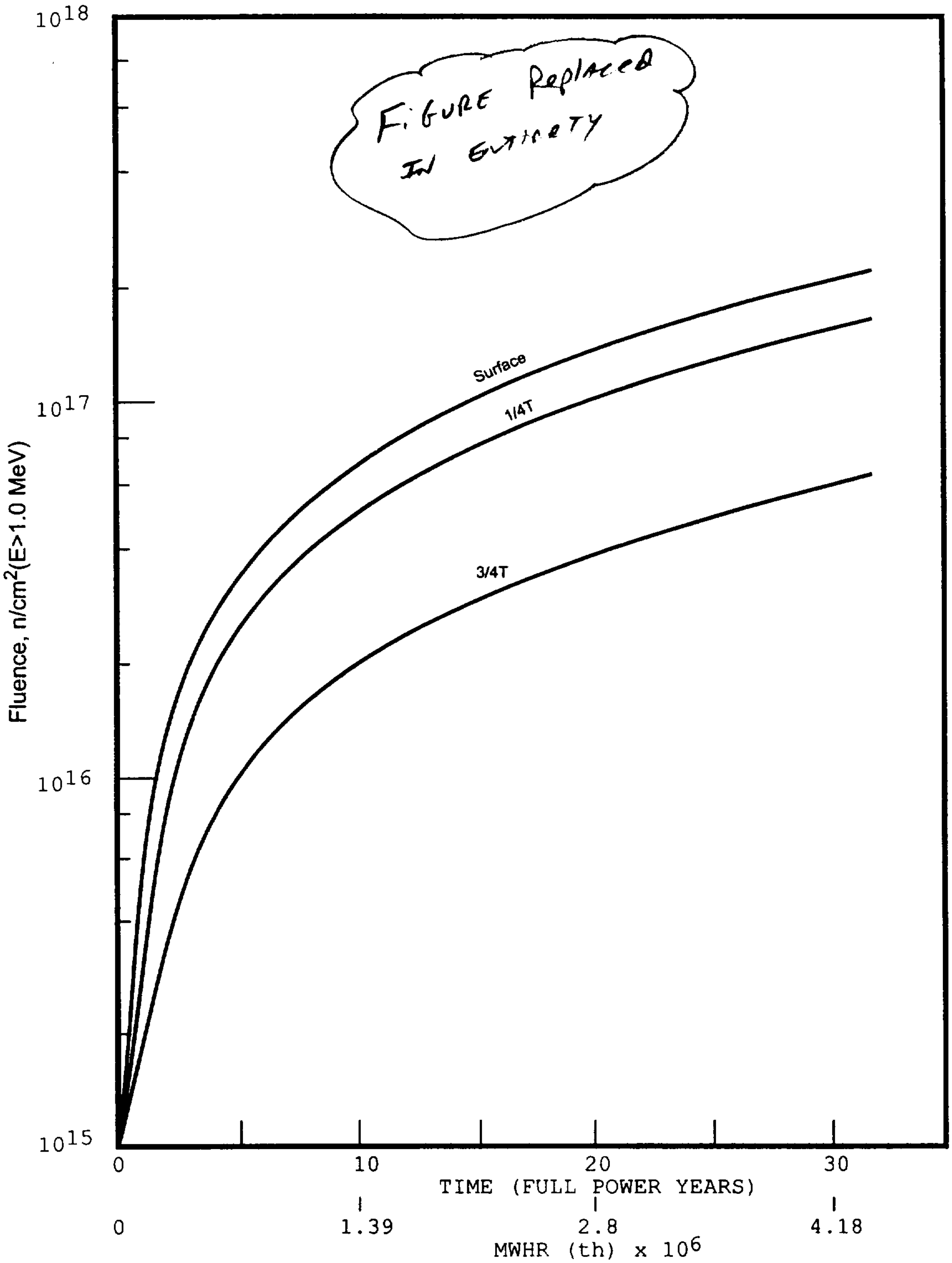
FIGURE 3.6.2

FAST NEUTRON FLUENCE (E>1 MEV) AS A FUNCTION OF THERMAL ENERGY AND FULL POWER YEARS



REFERENCE: L.M. Lowry et al. "Examination, Testing, and Evaluation of Irradiated Pressure Vessel Surveillance Specimens from Vermont Yankee Nuclear Power Station.

Batelle Columbus Laboratories Report #BCL-585-84-3, May 15, 1984

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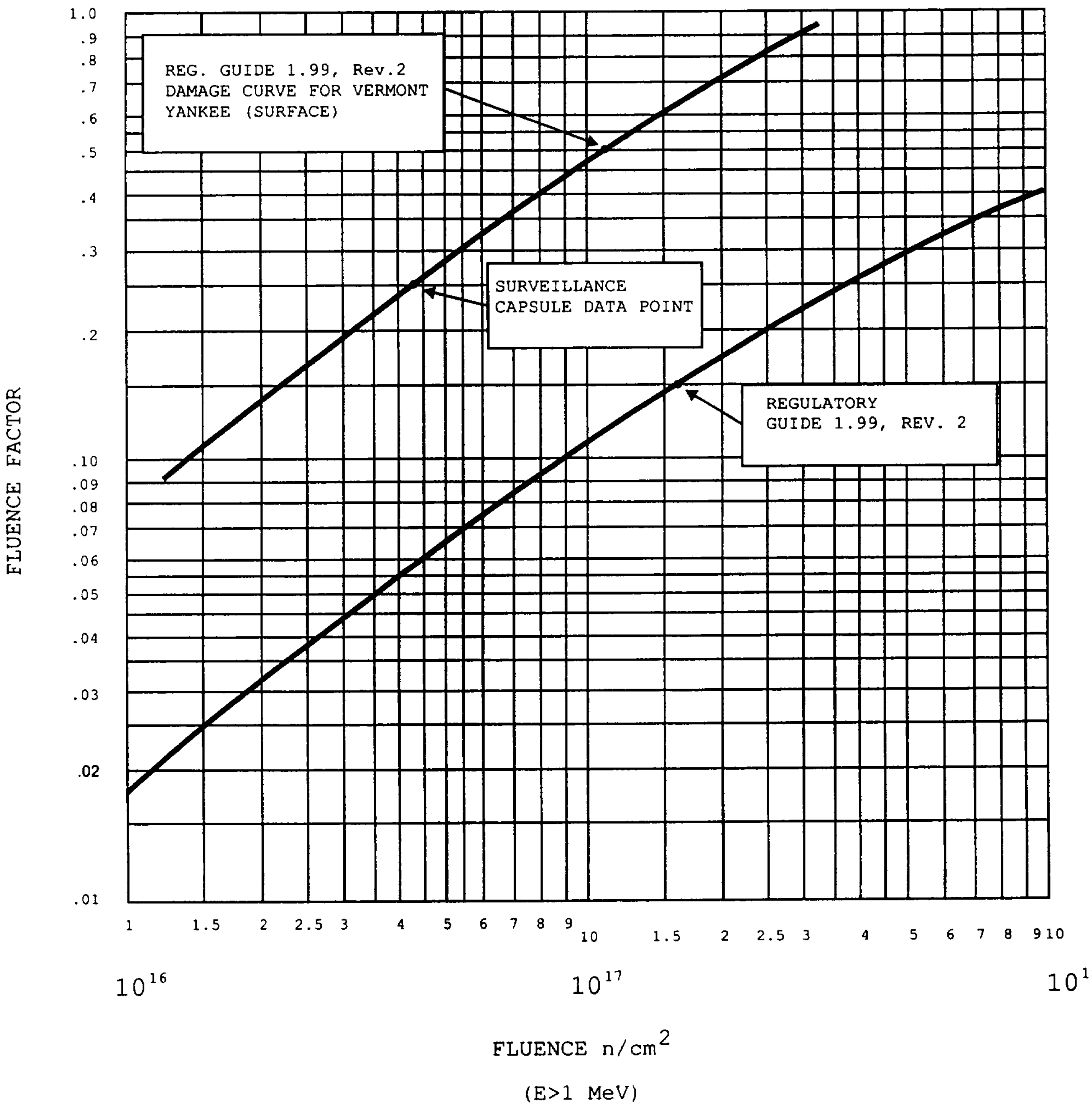
FIGURE 3.6.3

FLUENCE FACTOR FOR USE IN REGULATORY GUIDE 1.99

REVISION 2

Figure Replaced In Entirety



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BASES:

3.6 and 4.6 REACTOR COOLANT SYSTEM

A. Pressure and Temperature Limitations

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4.2 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

INSERT 1 →

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by their internal pressure. Therefore, a pressure-temperature curve based on steady-state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing locations.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures ~~should be~~ will be maintained within 50°F of each other prior to startup of an idle loop.

move to location A

The reactor vessel materials have been tested to determine their initial reference temperature nil-ductility transition temperature (RT_{NDT}) of 40°F maximum. Reactor operation and resultant fast neutron (E > 1 Mev) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature can be predicted using current industry practices and Vermont Yankee Surveillance Program data. (Regulatory Guide 1.99, Revision 2, and Battelle Columbus Laboratory Report BCL 585-84-3, dated May 15, 1984. The pressure/temperature limit curve, Figure 3.6.1, includes predicted adjustments for this shift in RT_{NDT} for operation through 4.46×10^8 MWH(t), as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The reference temperature of the closure flange material was determined by material testing and Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements for Older Plants". The closure flange is located in a low neutron fluence area and therefore no measurable RT_{NDT} shift is expected over the life of the plant.

BASES: 3.6 and 4.6 (Cont'd)

CRITICAL PLATE AND WELD

IN THE CORE REGION

The actual shift in RT_{NDT} of the ~~vessel~~ material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185 reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. Battelle Columbus Laboratory Report BCL-585-84-3, dated May 15, 1984, provides this information for the ten-year surveillance capsule. ~~In order to estimate the material properties at the 1/4 and 3/4 T positions in the vessel plate, the shift in RT_{NDT} is determined in accordance with Regulatory Guide 1.99, Revision 2. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.~~

LOCATION A

The pressure-temperature limit lines, shown on Figure 3.6.1~~, for reactor criticality and~~ for inservice leak and hydrostatic testing have been provided to assure ~~compliance with~~ the minimum temperature requirements of ~~Appendix~~ G to 10CFR50 for reactor ~~criticality~~ and for ~~inservice leak~~ and hydrostatic testing.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided to assure compliance with the requirements of Appendix H to 10CFR Part 50.

WHEN DATA FROM THE NEXT SURVEILLANCE CAPSULE IS AVAILABLE, THE PREDICTED BELTLINE ΔRT_{NDT} WILL BE REASSESSED AND THE P/T CURVES REVISED AS APPROPRIATE.

B. Coolant Chemistry

A steady-state radioiodine concentration limit of 1.1 µCi of I-131 dose equivalent per gram of water in the Reactor Coolant System can be reached if the gross radioactivity in the gaseous effluents is near the limit, as set forth in the Offsite Dose Calculation Manual, or if there is a failure or prolonged shutdown of the cleanup demineralizer. In the event of a steam line rupture outside the drywell, the NRC staff calculations show the resultant radiological dose at the site boundary to be less than 30 Rem to the thyroid. This dose was calculated on the basis of the radioiodine concentration limit of 1.1 µCi of I-131 dose equivalent per gram of water, atmospheric diffusion from an equivalent elevated release of 10 meters at the nearest site boundary (190 m) for a $X/Q = 3.9 \times 10^{-3}$ sec/m^3 (Pasquill D and 0.33 m/sec equivalent), and a steam line isolation valve closure time of five seconds with a steam/water mass release of 30,000 pounds.

The iodine spike limit of four (4) microcuries of I-131 dose equivalent per gram of water provides an iodine peak or spike limit for the reactor coolant concentration to assure that the radiological consequences of a postulated LOCA are within 10CFR Part 100 dose guidelines.

The reactor coolant sample will be used to assure that the limit of Specification 3.6.B.1 is not exceeded. The radioiodine concentration would not be expected to change rapidly during steady-state operation over a period of 96 hours. In addition, the trend of the radioactive gaseous effluents, which is continuously monitored, is a good indicator of the trend of the radioiodine concentration in the reactor coolant. When a significant increase in radioactive gaseous effluents is indicated, as specified, an additional reactor coolant sample shall be taken and analyzed for radioactive iodine.

Insert 1

The Pressure / Temperature (P/T) curves included as Figures 3.6.1, 3.6.2, and 3.6.3 were developed using 10CFR50 Appendix G, 1995 ASME Code, Section XI, Appendix G (including the Summer 1996 Addenda), and ASME Code Cases N-588 and N-640. These three curves provide P/T limit requirements for Pressure Test, Core Not Critical, and Core Critical. The P/T curves are not derived from Design Basis Accident analysis. They are prescribed to avoid encountering pressure, temperature or temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the reactor pressure boundary, a condition that is unanalyzed.

During heating events, the thermal gradients in the reactor vessel wall produce thermal stresses that vary from compressive at the inner wall to tensile at the outer wall. During cooling events the thermal stresses vary from tensile at the inner wall to compressive at the outer wall. The thermally induced tensile stresses are additive to the pressure induced tensile stresses. In the flange region, bolt preload has a significant affect on stress in the flange and adjacent plates. Therefore heating/cooling events and bolt preload are used in the determination of the pressure-temperature limitations for the vessel.

The guidance of Branch Technical Position - MTEB 5-2, material drop weight, and Charpy impact test results were used to determine a reference nil-ductility temperature (RT_{NDT}) for all pressure boundary components. For the plates and welds adjacent to the core, fast neutron (E > 1 Mev) irradiation will cause an increase in the RT_{NDT}. For these plates and welds an adjusted RT_{NDT} (ART_{NDT}) of 89°F and 73°F (¼ and ¾ thickness locations) was conservatively used in development of these curves for core region components. Based upon plate and weld chemistry, initial RT_{NDT} values, predicted peak fluence (2.3×10^{17} n/cm^2) for a gross power generation of 4.46×10^8 MWH(t) (Battelle Columbus Laboratory Report BCL 585-84-3, dated May 15, 1984) these core region ART_{NDT} values conservatively bound the guidance of Regulatory Guide 1.99, Revision 2.

There were five regions of the reactor pressure vessel (RPV) that were evaluated in the development of the P/T Limit curves: (1) the reactor vessel beltline region, (2) the bottom head region, (3) the feedwater nozzle, (4) the recirculation inlet nozzle, and (5) the upper vessel flange region. These regions will bound all other regions in the vessel with respect to considerations for brittle fracture.

Two lines are shown on each P/T limit figure. The dashed line is the Bottom Head Curve. This is applicable to the bottom head area only and includes the bottom head knuckle plates and dollar plates. Based on bottom head fluid temperature and bottom head surface temperature, the reactor pressure shall be maintained below the dashed line at all times.

Due to convection cooling, stratification, and cool CRD flow, the bottom head area is subject to lower temperatures than the balance of the pressure vessel. The RT_{NDT} of the lower head is lower than the ART_{NDT} used for the beltline. The lower head area is also not subject to the same high level of stress as the flange and feedwater nozzle regions. The dashed Bottom Head Curve is less restrictive than the enveloping curve used for the upper regions of the vessel and provides Operator's with a conservative, but less restrictive P/T limit for the cooler bottom head region.

The solid line is the Upper Region Curve. This line conservatively bounds all regions of the vessel including the most limiting beltline and flange areas. At temperatures below the 10CFR50 Appendix G minimum temperature requirement (vertical line) based on the downcomer

temperature and flange temperature, the reactor pressure shall be maintained below the solid line. At temperatures in excess of the 10CFR50 Appendix G minimum temperature requirement, the allowable pressure based on the flange is much higher than the beltline limit. Therefore, when the flange temperature exceeds the 10CFR50 Appendix G minimum temperature requirement, the reactor pressure shall be maintained below the solid line based on downcomer temperature.

The Pressure Test curve (3.6.1) is applicable for heatup/cooldown rates up to 40°F/hr. The Core Not Critical curve (3.6.2) and the Core Critical curve (3.6.3) are applicable for heatup/cooldown rates up to 100°F/hr. In addition to heatup and cooldown events, the more limiting anticipated operational occurrences (AOOs) were evaluated (Structural Integrity Report, SIR-00-155, Rev 0). For the feedwater nozzles, a sudden injection of 50°F cold water into the nozzle was postulated in the development of all three curves. The bottom head region was independently evaluated for AOOs in addition to 40°F/hr and 100°F/hr heatup/cooldown rates. This evaluation demonstrated that P/T requirements of the bottom head would be maintained for transients that would bound rapid cooling as well as step increases in temperature. The rapid cooling event would bound scrams and other upset condition (level B) cold water injection events. The bottom head was also evaluated for a series of step heatup transients. This would depict hot sweep transients typically associated with reinitiation of recirculation flow with stratified conditions in the lower plenum. This demonstrated that there was significant margin to P/T limits with GE SIL 251 recommendations for reinitiating recirculation flow in stratified conditions.

Adjustments for temperature and pressure instrument uncertainty have been included in the curves. The minimum temperature requirements were all increased by 10°F to compensate for temperature loop uncertainty error. The maximum pressure values were all decreased by 30psi to account for pressure loop uncertainty error. In addition, the maximum pressure was reduced further to account for static elevation head assuming the level was at the top of the reactor and at 70°F.

Attachment 4

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 244

Revised P/T Curves

Retyped Technical Specification Pages

Listing of Affected Technical Specifications Pages

Replace the Vermont Yankee Nuclear Power Station Technical Specifications pages listed below with the revised pages. The revised pages contain vertical lines in the margin indicating the areas of change.

Remove	Insert
115	115
116	116
135	135
136	136
137	137
138	138
139	139
140	140

3.6 LIMITING CONDITIONS FOR OPERATION

3.6 REACTOR COOLANT SYSTEM

Applicability:

Applies to the operating status of the reactor coolant system.

Objective:

To assure the integrity and safe operation of the reactor coolant system.

Specification:

A. Pressure and Temperature Limitations

1. The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.6.1, 3.6.2 and 3.6.3, as appropriate.

2. The maximum heatup or cooldown rate is 100°F when averaged over any one hour period.

3. The reactor vessel head bolting shall not be tensioned unless the temperature of the vessel head flange and the head is greater than 70°F.

4. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50°F of each other.

4.6 SURVEILLANCE REQUIREMENTS

4.6 REACTOR COOLANT SYSTEM

Applicability:

Applies to the periodic examination and testing requirements for the reactor coolant system.

Objective:

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification:

A. Pressure and Temperature Limitations

1. The reactor coolant temperature and pressure shall be recorded at least once per hour during system heatup, cooldown and inservice leak and hydrostatic testing operations.

2. The reactor coolant temperature and pressure shall be recorded at the time of reactor criticality.

3. When the reactor vessel head bolting is being tightened or loosened, the reactor vessel shell temperature immediately below the vessel flange shall be permanently recorded.

4. Prior to and after startup of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be recorded.

3.6 LIMITING CONDITIONS FOR OPERATION

4.6 SURVEILLANCE REQUIREMENTS

5. The reactor vessel irradiation surveillance specimens shall be removed and examined to determine changes in material properties in accordance with the following schedule:

CAPSULE	REMOVAL YEAR
1	10
2	30
3	Standby

The results shall be used to reassess material properties and update Figures 3.6.1, 3.6.2 and 3.6.3, as appropriate. The removal times shall be referenced to the refueling outage following the year specified, referenced to the date of commercial operation.

B. Coolant Chemistry

1. a. During reactor power operation, the radioiodine concentration in the reactor coolant shall not exceed 1.1 microcuries of I-131 dose equivalent per gram of water, except as allowed in Specification 3.6.B.1.b.

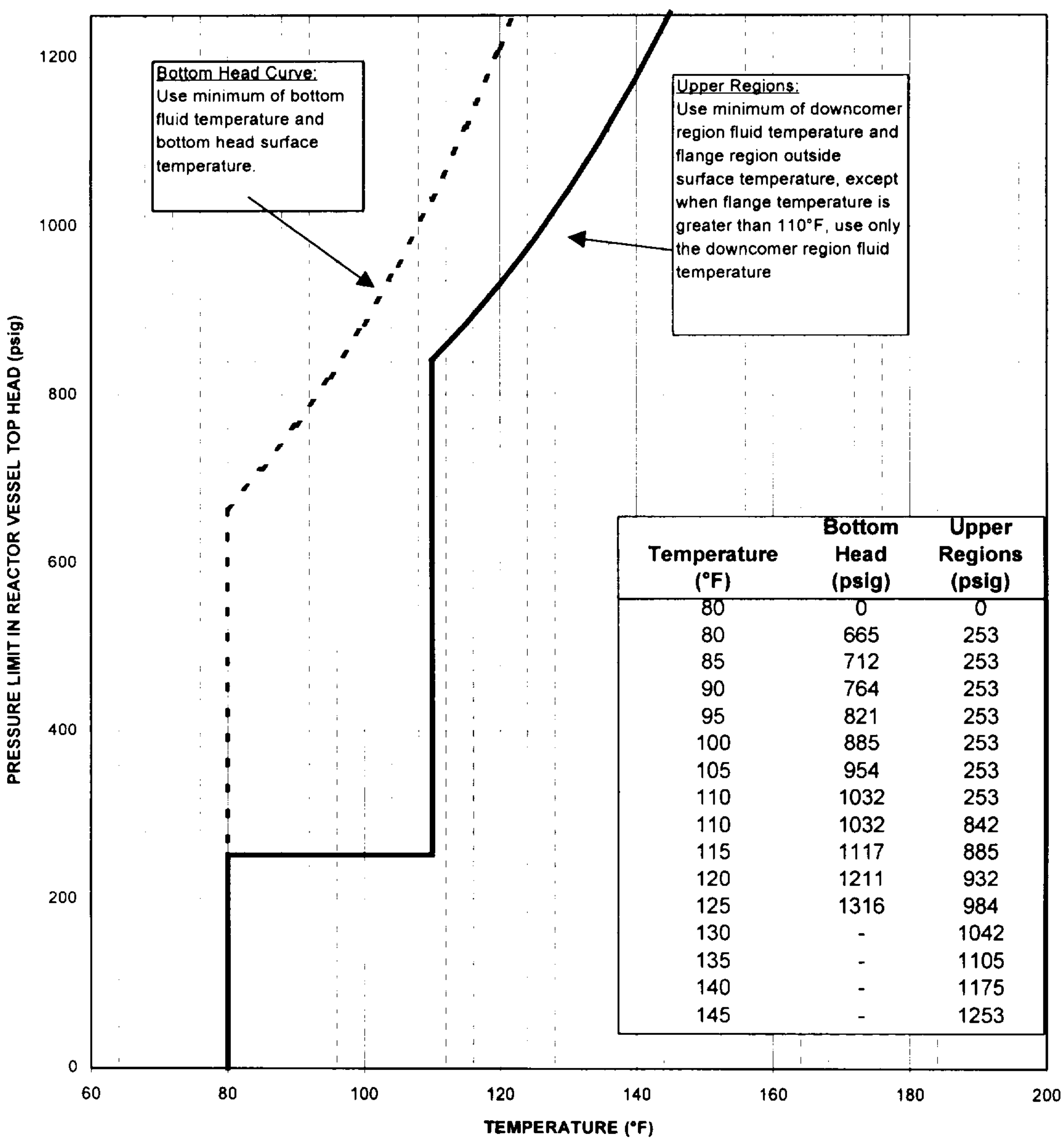
B. Coolant Chemistry

1. a. A sample of reactor coolant shall be taken at least every 96 hours and analyzed for radioactive iodines of I-131 through I-135 during power operation. In addition, when steam jet air ejector monitors indicate an increase in radioactive gaseous effluents of 25 percent or 5000 μCi/sec, whichever is greater, during steady state reactor operation a reactor coolant sample shall be taken and analyzed for radioactive iodines.

FIGURE 3.6.1

Reactor Vessel Pressure-Temperature Limitations
Hydrostatic Pressure and Leak Tests, Core Not Critical

40°F/hr Heatup/Cooldown Limit
Valid Through 4.46E8 MWH(t)

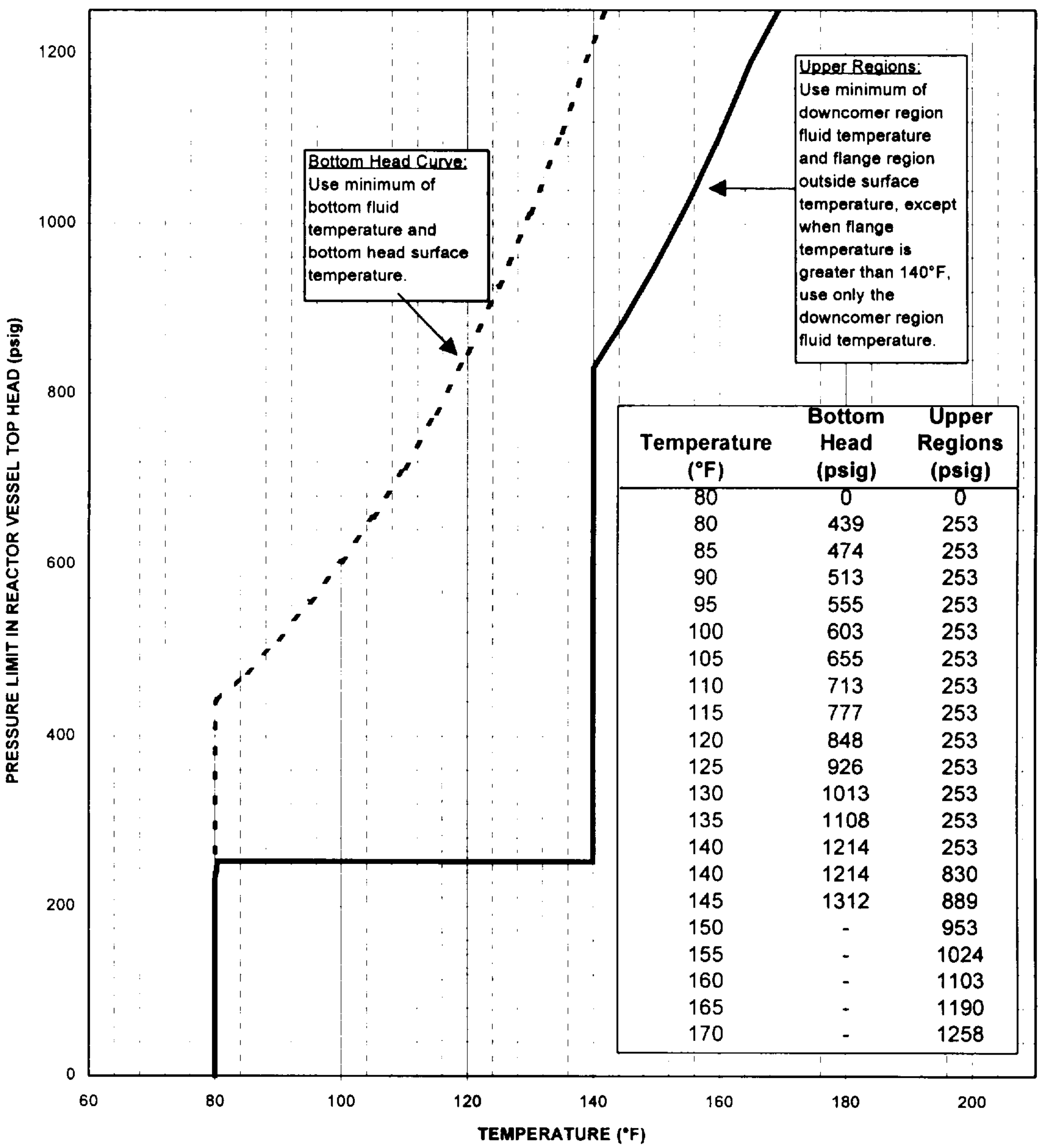


Temperature (°F)	Bottom Head (psig)	Upper Regions (psig)
80	0	0
80	665	253
85	712	253
90	764	253
95	821	253
100	885	253
105	954	253
110	1032	253
110	1032	842
115	1117	885
120	1211	932
125	1316	984
130	-	1042
135	-	1105
140	-	1175
145	-	1253

FIGURE 3.6.2

Reactor Vessel Pressure-Temperature Limitiations
Normal Operation, Core Not Critical

100°F/hr Heatup/Cooldown Limit
Valid Through 4.46E8 MWH(t)

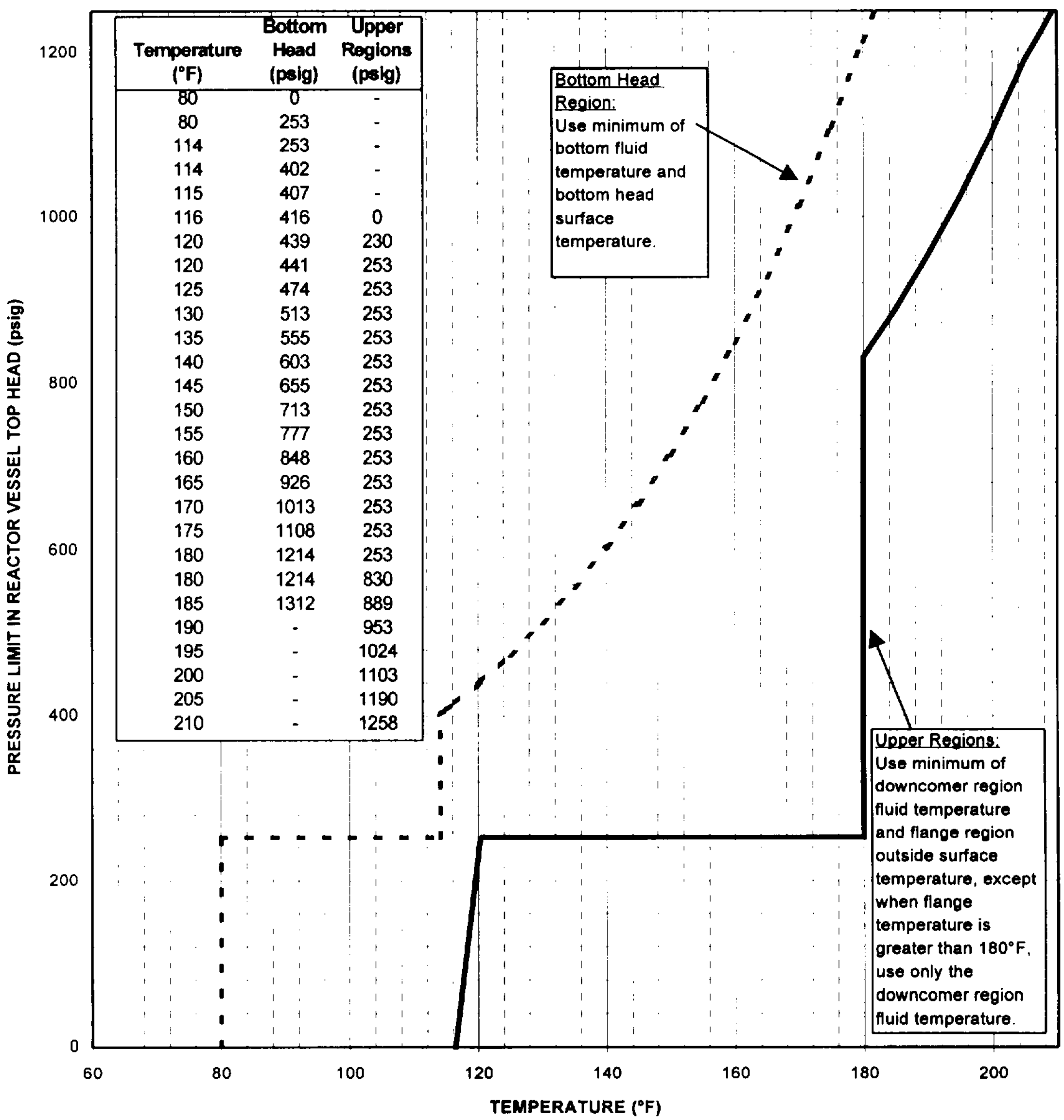


Temperature (°F)	Bottom Head (psig)	Upper Regions (psig)
80	0	0
80	439	253
85	474	253
90	513	253
95	555	253
100	603	253
105	655	253
110	713	253
115	777	253
120	848	253
125	926	253
130	1013	253
135	1108	253
140	1214	253
140	1214	830
145	1312	889
150	-	953
155	-	1024
160	-	1103
165	-	1190
170	-	1258

FIGURE 3.6.3

Reactor Vessel Pressure-Temperature Limitations
Normal Operation, Core Critical

100°F/hr Heatup/Cooldown Limit
If Pressure < 253 psig, Water Level must be within
Normal Range for Power Operation
Valid Through 4.46E8 MWH(t)



Temperature (°F)	Bottom Head (psig)	Upper Regions (psig)
80	0	-
80	253	-
114	253	-
114	402	-
115	407	-
116	416	0
120	439	230
120	441	253
125	474	253
130	513	253
135	555	253
140	603	253
145	655	253
150	713	253
155	777	253
160	848	253
165	926	253
170	1013	253
175	1108	253
180	1214	253
180	1214	830
185	1312	889
190	-	953
195	-	1024
200	-	1103
205	-	1190
210	-	1258

BASES:

3.6 and 4.6 REACTOR COOLANT SYSTEM

A. Pressure and Temperature Limitations

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4.2 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

The Pressure/Temperature (P/T) curves included as Figures 3.6.1, 3.6.2, and 3.6.3 were developed using 10CFR50 Appendix G, 1995 ASME Code, Section XI, Appendix G (including the Summer 1996 Addenda), and ASME Code Cases N-588 and N-640. These three curves provide P/T limit requirements for Pressure Test, Core Not Critical, and Core Critical. The P/T curves are not derived from Design Basis Accident analysis. They are prescribed to avoid encountering pressure, temperature or temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the reactor pressure boundary, a condition that is unanalyzed.

During heating events, the thermal gradients in the reactor vessel wall produce thermal stresses that vary from compressive at the inner wall to tensile at the outer wall. During cooling events the thermal stresses vary from tensile at the inner wall to compressive at the outer wall. The thermally induced tensile stresses are additive to the pressure induced tensile stresses. In the flange region, bolt preload has a significant affect on stress in the flange and adjacent plates. Therefore heating/cooling events and bolt preload are used in the determination of the pressure-temperature limitations for the vessel.

The guidance of Branch Technical Position - MTEB 5-2, material drop weight, and Charpy impact test results were used to determine a reference nil-ductility temperature (RT_{NDT}) for all pressure boundary components. For the plates and welds adjacent to the core, fast neutron (E > 1 Mev) irradiation will cause an increase in the RT_{NDT}. For these plates and welds an adjusted RT_{NDT} (ART_{NDT}) of 89°F and 73°F (¼ and ¾ thickness locations) was conservatively used in development of these curves for core region components. Based upon plate and weld chemistry, initial RT_{NDT} values, predicted peak fluence (2.3×10^{17} n/cm^2) for a gross power generation of 4.46×10^8 MWH(t) (Battelle Columbus Laboratory Report BCL 585-84-3, dated May 15, 1984) these core region ART_{NDT} values conservatively bound the guidance of Regulatory Guide 1.99, Revision 2.

There were five regions of the reactor pressure vessel (RPV) that were evaluated in the development of the P/T Limit curves: (1) the reactor vessel beltline region, (2) the bottom head region, (3) the feedwater nozzle, (4) the recirculation inlet nozzle, and (5) the upper vessel flange region. These regions will bound all other regions in the vessel with respect to considerations for brittle fracture.

BASES: 3.6 and 4.6 (Cont'd)

Two lines are shown on each P/T limit figure. The dashed line is the Bottom Head Curve. This is applicable to the bottom head area only and includes the bottom head knuckle plates and dollar plates. Based on bottom head fluid temperature and bottom head surface temperature, the reactor pressure shall be maintained below the dashed line at all times.

Due to convection cooling, stratification, and cool CRD flow, the bottom head area is subject to lower temperatures than the balance of the pressure vessel. The RT_{NDT} of the lower head is lower than the ART_{NDT} used for the beltline. The lower head area is also not subject to the same high level of stress as the flange and feedwater nozzle regions. The dashed Bottom Head Curve is less restrictive than the enveloping curve used for the upper regions of the vessel and provides Operator's with a conservative, but less restrictive P/T limit for the cooler bottom head region.

The solid line is the Upper Region Curve. This line conservatively bounds all regions of the vessel including the most limiting beltline and flange areas. At temperatures below the 10CFR50 Appendix G minimum temperature requirement (vertical line) based on the downcomer temperature and flange temperature, the reactor pressure shall be maintained below the solid line. At temperatures in excess of the 10CFR50 Appendix G minimum temperature requirement, the allowable pressure based on the flange is much higher than the beltline limit. Therefore, when the flange temperature exceeds the 10CFR50 Appendix G minimum temperature requirement, the reactor pressure shall be maintained below the solid line based on downcomer temperature.

The Pressure Test curve (3.6.1) is applicable for heatup/cooldown rates up to 40°F/hr. The Core Not Critical curve (3.6.2) and the Core Critical curve (3.6.3) are applicable for heatup/cooldown rates up to 100°F/hr. In addition to heatup and cooldown events, the more limiting anticipated operational occurrences (AOOs) were evaluated (Structural Integrity Report, SIR-00-155, Rev 0). For the feedwater nozzles, a sudden injection of 50°F cold water into the nozzle was postulated in the development of all three curves. The bottom head region was independently evaluated for AOOs in addition to 40°F/hr and 100°F/hr heatup/cooldown rates. This evaluation demonstrated that P/T requirements of the bottom head would be maintained for transients that would bound rapid cooling as well as step increases in temperature. The rapid cooling event would bound scrams and other upset condition (level B) cold water injection events. The bottom head was also evaluated for a series of step heatup transients. This would depict hot sweep transients typically associated with reinitiation of recirculation flow with stratified conditions in the lower plenum. This demonstrated that there was significant margin to P/T limits with GE SIL 251 recommendations for reinitiating recirculation flow in stratified conditions.

Adjustments for temperature and pressure instrument uncertainty have been included in the curves. The minimum temperature requirements were all increased by 10°F to compensate for temperature loop uncertainty error. The maximum pressure values were all decreased by 30psi to account for pressure loop uncertainty error. In addition, the maximum pressure was reduced further to account for static elevation head assuming the level was at the top of the reactor and at 70°F.

BASES: 3.6 and 4.6 (Cont'd)

The actual shift in RT_{NDT} of the critical plate and weld material in the core region will be established periodically during operation by removing and evaluating, in accordance with ASTM E185, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. Battelle Columbus Laboratory Report BCL-585-84-3, dated May 15, 1984, provides this information for the ten-year surveillance capsule. When data from the next surveillance capsule is available, the predicted beltline ART_{NDT} will be re-assessed and the P/T curves revised as appropriate.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures will be maintained within 50°F of each other prior to startup of an idle loop.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided to assure compliance with the requirements of Appendix H to 10CFR Part 50.

B. Coolant Chemistry

A steady-state radioiodine concentration limit of 1.1 µCi of I-131 dose equivalent per gram of water in the Reactor Coolant System can be reached if the gross radioactivity in the gaseous effluents is near the limit, as set forth in the Offsite Dose Calculation Manual, or if there is a failure or prolonged shutdown of the cleanup demineralizer. In the event of a steam line rupture outside the drywell, the NRC staff calculations show the resultant radiological dose at the site boundary to be less than 30 Rem to the thyroid. This dose was calculated on the basis of the radioiodine concentration limit of 1.1 µCi of I-131 dose equivalent per gram of water, atmospheric diffusion from an equivalent elevated release of 10 meters at the nearest site boundary (190 m) for a $X/Q = 3.9 \times 10^{-3}$ sec/m^3 (Pasquill D and 0.33 m/sec equivalent), and a steam line isolation valve closure time of five seconds with a steam/water mass release of 30,000 pounds.

The iodine spike limit of four (4) microcuries of I-131 dose equivalent per gram of water provides an iodine peak or spike limit for the reactor coolant concentration to assure that the radiological consequences of a postulated LOCA are within 10CFR Part 100 dose guidelines.

The reactor coolant sample will be used to assure that the limit of Specification 3.6.B.1 is not exceeded. The radioiodine concentration would not be expected to change rapidly during steady-state operation over a period of 96 hours. In addition, the trend of the radioactive gaseous effluents, which is continuously monitored, is a good indicator of the trend of the radioiodine concentration in the reactor coolant. When a significant increase in radioactive gaseous effluents is indicated, as specified, an additional reactor coolant sample shall be taken and analyzed for radioactive iodine.

Docket No. 50-271
BVY 00-113

Attachment 5

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 244

Revised P/T Curves

Request for Exemption from the requirements of 10CFR50, Appendix G

REQUEST FOR EXEMPTION FROM THE REQUIREMENTS OF 10CFR50.60(a) AND 10CFR50 APPENDIX G

In accordance with 10CFR50.12, Vermont Yankee Nuclear Power Corporation (VYNPC) is requesting an exemption from the requirements of 10CFR50.60(a) for the Vermont Yankee Nuclear Power Plant. The exemption permits the use of ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves, Section XI, Division 1," and ASME Code Case N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels, Section XI, Division 1," in lieu of 10CFR50, Appendix G, Paragraph IV.A.2.b.

Justification for Use of Code Case N-640

The requested exemption to allow use of ASME Code Case N-640 in conjunction with ASME Code, Section XI, Appendix G, to determine the pressure and temperature (P/T) limits meets the criteria of 10CFR50.12 as discussed below.

10CFR 50.12 states that the Commission may grant an exemption from requirements contained in 10CFR50 provided:

The requested exemption is authorized by law: No law precludes the activities covered by this exemption request. 10CFR50.60(b) allows the use of alternatives to 10CFR50, Appendices G and H when the NRC grants an exemption under 10CFR50.12.

The requested exemption does not present an undue risk to the public health and safety: The proposed revision to the P/T limits relies, in part, on the requested exemption. The revised P/T limits were developed using the K_{IC} fracture toughness curve shown on ASME Code, Section XI, Appendix A, Figure A-4200-1, in lieu of the K_{IA} fracture toughness curve of ASME Code, Section XI, Appendix G, Figure G-2210-1, as the lower bound for fracture toughness. The other margins involved with the ASME Code, Section XI, Appendix G process of determining P/T limit curves remain unchanged.

Use of the K_{IC} curve 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 models the slow heatup and cooldown process of a reactor vessel. Use of this approach is justified by the initial conservatism of the K_{IA} curve when the curve was codified in 1974. This initial conservatism was necessary due to limited knowledge of reactor pressure vessel (RPV) material fracture toughness.

Since 1974, additional knowledge about the fracture toughness of vessel materials and their fracture response to applied loads has been gained. The additional knowledge demonstrates the lower bound fracture toughness provided by the K_{IA} curve is well beyond the margin of safety required to protect against potential RPV failure. The lower bound K_{IC} fracture toughness provides an adequate margin of safety to protect against potential RPV failure and does not present an undue risk to public health and safety.

P/T curves based upon the K_{IC} toughness limits will enhance overall plant safety by opening the P/T operating window, especially in the region of low-temperature operations. The two primary benefits occurring during the pressure test are a reduction in the duration of the pressure test and personnel safety while conducting inspections in primary containment at elevated temperatures with no decrease to the margin of safety.

The requested exemption will not endanger the common defense and security: The common defense and security are not endangered by this exemption request.

Special circumstances are present which necessitate the request for an exemption to the regulations of 10CFR50.60: In accordance with 10CFR50.12(a)(2), the NRC will consider granting an exemption to the regulations if special circumstances are present. This exemption meets the special circumstances of the following paragraphs:

- 10CFR50.12(a)(2)(ii) - Demonstrates the underlying purpose of the regulation will continue to be achieved.

- 10CFR50.12(a)(2)(iii)- Will result in undue hardship or other cost that are significant if the regulation is enforced.

- 10CFR50.12(a)(2)(v) - Will provide only temporary relief from the applicable regulation and the licensee has made good-faith efforts to comply with the regulations.

10CFR50.12(a)(2)(ii): ASME Code, Section XI, Appendix G, provides procedures for determining allowable loading on the RPV and is approved for that purpose by 10CFR50, Appendix G. Application of these procedures in the determination of P/T operating and test curves satisfies the underlying requirement that:

1) The reactor coolant pressure boundary be operated in a regime having sufficient margin to ensure, when stressed, the vessel boundary behaves in a non-brittle manner and the probability of a rapidly propagating fracture is minimized; and

2) P/T operating and test limit curves provide adequate margin in consideration of uncertainties in determining the effects of irradiation on material properties.

The ASME Code, Section XI, Appendix G, procedure was conservatively developed based upon the level of knowledge existing in 1974 concerning RPV materials and the estimated effects of operation. Since 1974, the level of knowledge concerning these topics has greatly expanded. This increased knowledge permits relaxation of the ASME Code, Section XI, Appendix G, requirements via application of ASME Code Case N-640, while maintaining the underlying purpose of the ASME Code and NRC regulations to ensure an acceptable margin of safety.

10CFR50.12(a)(2)(iii): The reactor coolant system (RCS) P/T operating window is defined by the P/T operating and test limit curves developed in accordance with the ASME Code, Section XI, Appendix G procedure. Continued operation of Vermont Yankee with these P/T curves without the relief provided by ASME Code Case N-640 would unnecessarily restrict the P/T operating window. This restriction requires the Operations staff to maintain a high temperature during pressure tests and also subjects inspection personnel to increased safety hazards while conducting inspections of systems with the potential for steam leaks in a primary containment at elevated temperatures.

This constitutes an unnecessary burden that can be alleviated by the application of ASME Code Case N-640 in the development of the proposed P/T curves. Implementation of the proposed P/T curves, as allowed by ASME Code Case N-640, does not significantly reduce the margin of safety.

10CFR50.12(a)(2)(v): The requested exemption provides only temporary relief, since VY anticipates that the provisions of Code Case N-640 will be incorporated into (or reconciled with) the requirements of 10CFR50 Appendix G, based on ongoing industry efforts to do so.

NRC approval of the Code Case is pending, but additional action may be required to allow use of the Code Case without requiring an exemption to 10CFR50 Appendix G. The estimated time for such actions to be completed is unknown, and therefore, the effective period of time that the exemption would be effective is indefinite.

Code Case N-640, Conclusion for Exemption Acceptability

Compliance with the specified requirement of 10CFR50.60(a) will result in hardship and unusual difficulty without a compensating increase in the level of quality and safety. ASME Code Case N-640 allows a reduction in the lower bound fracture toughness used by ASME Code, Section XI, Appendix G, in the determination of RPV P/T limits. This proposed alternative is acceptable, because the ASME Code Case maintains the relative margin of safety commensurate with the margin of safety that existed at the time ASME Code, Section XI, Appendix G, was approved in 1974. Therefore, application of ASME Code Case N-640 for Vermont Yankee ensures an acceptable margin of safety and does not present an undue risk to the public health and safety.

Justification for the Use of Code Case N-588

The requested exemption to allow use of ASME Code Case N-588 to determine stress intensity factors for postulated flaws and postulated flaw orientation for circumferential welds meets the criteria of 10CFR50.12 as discussed below. 10CFR50.12 states that the Commission may grant an exemption from requirements contained in 10CFR50 provided that:

The requested exemption is authorized by law: No law precludes the activities covered by this exemption request. 10CFR50.60(b) allows the use of alternatives to 10CFR50, Appendices G and H, when the NRC grants an exemption under 10CFR50.12.

The requested exemption does not present an undue risk to the public health and safety: 10CFR50, Appendix G, requires that Article G-2120 of ASME Code, Section XI, Appendix G, be used to determine the maximum postulated defects in RPVs for the vessel P/T limits. These limits are determined for normal operation and pressure/leak test conditions. Article G-2120 specifies, in part, that the postulated defect be in the surface of the vessel material and normal (perpendicular in the plane of the material) to the direction of maximum stress. ASME Code, Section XI, Appendix G, also provides methodology for determining the stress intensity factors for a maximum postulated defect normal to the maximum stress. The purpose of this article is, in part, to ensure the prevention of nonductile fractures by providing procedures to identify the most limiting postulated fractures to be considered in the development of pressure-temperature limits.

Code Case N-588 provides benefits in terms of calculating P/T limits by revising the Article G-2120 reference flaw orientation for circumferential welds in reactor vessels. The reference flaw is a postulated flaw that accounts for the possibility of a prior existing defect that may have gone undetected during the fabrication process. Thus, the intended application of a reference flaw is to account for defects that could physically exist within the geometry of the weldment. The current ASME Code, Section XI, Appendix G approach mandates the consideration of an axial reference flaw in circumferential welds for purposes of calculating the P/T limits. Postulating the Appendix G reference flaw in a circumferential weld is physically unrealistic and overly conservative, because the length of the flaw is 1.5 times the vessel wall thickness, which is much longer than the width of circumferential welds. The possibility that an axial flaw may extend from a circumferential weld into a plate/forging or axial weld is already adequately covered by the requirement that defects be postulated in plates/forgings and axial welds. The fabrication of RPVs for nuclear power plant

operation involved precise welding procedures and controls designed to optimize the resulting weld microstructure and provide the required material properties.

These controls are also designed to minimize defects that could be introduced into the weld during the fabrication process. Industry experience with the repair of weld indications found during preservice inspection, inservice nondestructive examinations, and data taken from destructive examination of actual vessel welds confirms that any remaining defects are small, laminar in nature, and do not cross transverse to the weld bead. Therefore, any postulated defects introduced during the fabrication process and not detected during subsequent nondestructive examinations would only be expected to be oriented in the direction of weld fabrication. For circumferential welds, this indicates a postulated defect with a circumferential orientation.

ASME Code Case N-588 addresses this issue by allowing consideration of maximum postulated defects oriented circumferentially in circumferential welds. ASME Code Case N-588 also provides appropriate procedures for determining the stress intensity factors for use in developing RPV P/T limits per ASME Code, Section XI, Appendix G procedures. The procedures allowed by ASME Code Case N-588 are conservative and provide a margin of safety in the development of RPV P/T operating and pressure test limits that will prevent nonductile fracture of the vessel.

The proposed P/T limits include restrictions on allowable operating conditions and equipment operability requirements to ensure operating conditions are consistent with the assumptions of the accident analysis. Specifically, RCS pressure and temperature must be maintained within the heatup and cooldown P/T limits specified in Technical Specification Figures 3.6.1, 3.6.2 and 3.6.3. Therefore, this exemption does not present an undue risk to the public health and safety.

The requested exemption will not endanger the common defense and security: The common defense and security are not endangered by this exemption request.

Special circumstances are present which necessitate the request for an exemption to the regulations of 10CFR50.60: In accordance with 10CFR50.12(a)(2), the NRC will consider granting an exemption to the regulations if special circumstances are present. This exemption meets the special circumstances of the following paragraphs:

- 10CFR50.12(a)(2)(ii) - Demonstrates that the underlying purpose of the regulation will continue to be achieved;

- 10CFR50.12(a)(2)(iii)- Will result in undue hardship or other costs that are significant if the regulation is enforced and;

- 10CFR50.12(a)(2)(v) - Will provide only temporary relief from the applicable regulation and the licensee has made good faith efforts to comply with the regulations.

10CFR50.12(a)(2)(ii): The underlying purpose of 10CFR50, Appendix G and ASME Code, Section XI, Appendix G, is to satisfy the underlying requirement that:

1) The reactor coolant pressure boundary be operated in a regime having sufficient margin to ensure that when stressed the vessel boundary behaves in a non-brittle manner and the probability of a rapidly propagating fracture is minimized; and

2) P/T operating and test curves provide margin in consideration of uncertainties in determining the effects of irradiation on material properties.

Application of ASME Code Case N-588 when determining P/T operating and test limit curves per ASME Code, Section XI, Appendix G, provides appropriate procedures for determining limiting maximum postulated defects and considering those defects in the P/T limits. This application of the Code Case maintains the margin of safety originally contemplated when ASME Code, Section XI, Appendix G was developed. Therefore, use of ASME Code Case N-588, as described above, satisfies the underlying purpose of the ASME Code and the NRC regulations to ensure an acceptable level of safety.

10CFR50.12(a)(2)(iii): The RCS P/T operating window is defined by the P/T operating and test curves developed in accordance with the ASME Code Section XI, Appendix G procedure. Continued operation without the relief provided by ASME Code Case N-588 will unnecessarily restrict the P/T operating window for Vermont Yankee. This restriction requires the Operations staff to maintain a high temperature during pressure tests and also subjects inspection personnel to increased safety hazards while conducting inspections of systems with the potential for steam leaks in a primary containment at elevated temperatures.

This constitutes an unnecessary burden that can be alleviated by the application of ASME Code Case N-588 in the development the proposed P/T curves. Implementation of the proposed P/T curves as allowed by ASME Code Case N-588 does not reduce the margin of safety originally contemplated by either the NRC or ASME.

10CFR50.12(a)(2)(v): The requested exemption provides only temporary relief, since VY anticipates that the provisions of Code Case N-588 will be incorporated into (or reconciled with) the requirements of 10CFR50 Appendix G, based on ongoing industry efforts to do so. NRC approval of the Code Case is pending, but additional action may be required to allow use of the Code Case without requiring an exemption to 10CFR50 Appendix G. The estimated time for such actions to be completed is unknown, and therefore, the effective period of time that the exemption would be effective is indefinite.

ASME Code Case N-588, Conclusion for Exemption Acceptability

Compliance with the specified requirements of 10CFR50.60 will result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. ASME Code Case N-588 allows postulation of a circumferential defect in circumferential welds to be considered in lieu of requiring the defect to be oriented across the weld from one plate or forging to the adjoining plate or forging. This circumstance was not considered at the time ASME Code, Section XI, Appendix G was developed and imposes restrictions on P/T operating limits beyond those originally contemplated.

This proposed alternative is acceptable, because the Code Case N-588 maintains the relative margin of safety commensurate with the margin of safety that existed at the time ASME Code, Section XI, Appendix G, was approved in 1974. Therefore, application of ASME Code Case N-588 will ensure an acceptable margin of safety. The approach is justified by consideration of the overpressurization design basis events and the resulting margin to RPV failure.

Restrictions on allowable operating conditions and equipment operability requirements are established to ensure operating conditions are consistent with the assumptions of the accident analysis. Specifically, RCS pressure and temperature must be maintained within the heatup and cooldown rate-dependent P/T limits specified in Technical Specification Figures 3.6.1, 3.6.2 and 3.6.3. Therefore, this exemption does not present an undue risk to the public health and safety.

Attachment 6

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 244

Revised P/T Curves

Technical Report "Revised Pressure-Temperature Curves for Vermont Yankee Nuclear Plant"

Structural Integrity Associates, Inc.

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December 8, 2000
SIR-00-155, Rev. 0
AFD-00-092

Mr. Enrico Betti
Vermont Yankee Nuclear Power Corp.
P. O. Box 250, PSB
546 Governor Hunt Road
Vernon, VT 05354

Subject: Revised Pressure-Temperature Curves for Vermont Yankee Nuclear Plant

Reference: Purchase Order No. 99-59048-00, Rev. 2, C.O. No. 6

Dear Rico:

Attachment 1 to this letter documents the revised set of pressure-temperature (P-T) curves developed for the Vermont Yankee Nuclear Plant (VY), in accordance with SI's Quality Assurance Program. This work was performed in accordance with the referenced contract, and includes a full set of updated P-T curves (i.e., pressure test, core not critical, and core critical conditions) for VY. The curves were developed in accordance with 1995 ASME Code, Section XI, Appendix G (including the Summer 1996 Addenda), U.S. 10CFR 50 Appendix G, and ASME Code Case N-588 and N-640.

The inputs, methodology, and results for this effort are summarized in Attachment 1. Please don't hesitate to call me if you have any questions.

Very truly yours,

AFDeardorff

A. F. Deardorff, P.E.
Associate

jj
Attachment
cc: VY-04Q-402

SIR-00-155, Rev. 0

ATTACHMENT 1
REVISED PRESSURE-TEMPERATURE CURVES FOR VERMONT YANKEE NUCLEAR POWER STATION

Prepared by: Arthur F. Deardorff, P.E.
Associate

Reviewed by: Keith R. Evon
Engineering Specialist

Timothy D. Gilman
Principal Engineer

Approved by: Arthur F. Deardorff, P.E. 12/8/2000
Associate

REVISED P-T CURVES FOR VERMONT YANKEE NUCLEAR POWER STATION

1.0 Introduction

This attachment documents the revised set of pressure-temperature (P-T) curves developed for the Vermont Yankee Nuclear Power Station (VY). This work includes a full set of updated P-T curves (i.e., pressure and leak test, core not critical, and core critical conditions) applicable for a gross power generation of 4.46×10^8 MWHR(th). (which will bound VY power generation beyond March 12, 2012, the end of VY's current operating license (EOL).)

The curves were developed using the methodology specified in ASME Code Cases N-588 [1] and N-640 [2], the 1995 ASME Code, Section XI, Appendix G (including the Summer 1996 Addenda) [3], and 10CFR50 Appendix G [4].

2.0 Material Properties

An assessment of the fracture toughness properties of all material used in the VY reactor vessel plate, weld and forgings was conducted by SI. Estimation of the initial value of the nil-ductility reference temperature (RT_{NDT}) was based on the methods described in Branch Technical Position MTEB 5-2 [5]. Charpy impact and drop weight test data from original construction Certified Materials Test Reports (CMTRs) and as-fabricated material testing [6,7], supplemented by more recent data from Battelle for one beltline plate [8], were used. The resulting initial RT_{NDT}'s are listed in Table 1.

For all material adjacent to the reactor vessel flange region, the GE vessel purchase contract required that a nil-ductility transition temperature (NDTT) of 10°F be met. Review of the CMTR data shows that the minimum Charpy energy (longitudinal specimens) was 69 ft-lb at 10°F, with 52 mils lateral expansion reported. Two "no-break" drop weight tests at 20°F were also reported. Based on MTEB 5-2, this justifies an RT_{NDT} = 10°F.

For the limiting material adjacent to the core region, the previous submittal by VY [10] stated that the initial RT_{NDT} of plate 1-14 was 40°F. Further evaluation justifies that the RT_{NDT} can be conservatively taken as 30°F.

- Evaluation of the CMTR data shows that the minimum Charpy energy (from longitudinal specimens) was 42 ft-lb at a test temperature of 10°F. Lateral expansion was not reported. Two no-break drop weight tests at 40°F were reported, justifying the NDTT of ≤ 30°F. Based on MTEB 5-2, this justifies an initial RT_{NDT} = 30°F.
- Evaluation of the "as-fabricated" test data shows that the minimum Charpy energy (from longitudinal specimens) was 65 ft-lb at 40°F. The minimum lateral expansion was 54 mils. Two no-break drop weight tests at 20°F were reported, justifying an NDTT of ≤ 10°F. Based on MTEB 5-2, this justifies an initial RT_{NDT} ≤ 10°F.
- Additional testing by Battelle exhibited relatively low Charpy energy (longitudinal specimens) [8]. At 40°F, 80°F and 120°F, the Charpy energy was 46.5 ft-lb, 57.5 ft-lb and 87.5 ft-lb, respectively with lateral expansion greater than 35 mils in all cases. From this data, it is estimated that the 50 ft-lb Charpy energy could have been achieved at ≤ 70°F. Using the criteria from MTEB 5-2, this also justifies an RT_{NDT} of 30°F.

Similar evaluations were conducted by SI in establishing the initial RT_{NDT}'s for all other materials.

Table 2 shows an evaluation of the expected irradiation shift for the beltline plates. The peak fluence of 2.3×10^{17} m/cm^2 (E>1.0 MeV) used in this table was used in VY's previous 1989 PT submittal [10]. The fluence value was from the peak fluence of 2.2×10^{17} n/cm^2 (>1.0 MeV) calculated by Battelle [9] with an additional $0.1 \times E^{17}$ n/cm^2 added for axial fluence variation effects.

For purposes of determining the P-T curves for the vessel core region materials, the evaluation has been based on the more conservatively shifted ART_{NDT}'s previously used by VY: 89°F at the 1/4T point and 73°F at the 3/4T point. Based on NRC's safety evaluation of the VY submittal, lower values of ART_{NDT} could have been used [11].

The conservatism of employing these ART_{NDT} values is expressed in terms of equivalent fluence in Table 3. Based on the initial RT_{NDT} values and chemistry factors from Table 2, and Regulatory Guide 1.99, Rev. 2 [12] criteria for calculating ART_{NDT}, the use of the conservative ART_{NDT} values equate to a minimum end-of-life surface fluence of 1.24×10^{18} n/cm^2 for the four core region plates. This is more than 5 times the peak end-of-life surface fluence calculated for Vermont Yankee by Battle [9]. This also confirms that plate 1-14, used for the VY surveillance specimens [9], is the critical plate from the standpoint of brittle failure up to fluence levels well beyond that expected at VY.

3.0 P-T Curve Methodology

The P-T curve methodology is based on the requirements of References [1] through [4]. There are five regions of the reactor pressure vessel (RPV) that were evaluated by SI: (1) the reactor vessel beltline region, (2) the bottom head region, (3) the feedwater nozzle, (4) the recirculation inlet nozzle, and (5) the upper vessel flange region. These regions will bound all other regions in the vessel with respect to considerations for brittle fracture. For the feedwater nozzle, the limiting conditions of sudden injection of 50°F cold water into the nozzle were considered. For the remainder of the locations, 100°F/hr heatup and cooldown were considered for Service Level A/B curves and 40°F/hr heatup and cooldown were conservatively assumed for pressure and leak test conditions. The bottom head region was independently evaluated for anticipated operational occurrences including rapid cooling following a plant scram and hot sweep transients typically associated with re-initiation of recirculation flow into a relatively colder lower head region following a reactor scram and recirculation pump trip.

3.1 General Approach for Analytical P-T Limit Curves

The general approach for development of the P-T curves was as follows:

a. A temperature at the crack tip, $T_{1/4t}$ (i.e., 1/4t into the inside or outside vessel wall surface) is either determined using ASME Section XI, Appendix G methods or is assumed. The method for each location addressed in discussed in subsequent sections.

b. Calculate the allowable stress intensity factor, K_{IC}, based on $T_{1/4t}$ using the relationship specified by Code Case N-640 [2], as follows:

$$K_{IC} = 20.734\, e^{[0.02(T_{1/4t} - ART_{NDT})]} + 33.2$$

where:
- $T_{1/4t}$ = metal temperature at assumed flaw tip (°F)
- ART_{NDT} = adjusted reference temperature for location under consideration and desired EFPY (°F)
- K_{IC} = allowable stress intensity factor (ksi$\sqrt{\text{inch}}$)

c. Calculate the thermal stress intensity factor, K_{IT}. This is calculated based on Code Case N-588 [1] for the beltline and lower head regions, from alternate analysis for the feedwater nozzle or recirculation inlet nozzle/upper vessel regions, or using membrane and bending stresses from the reactor vessel stress report for the upper flange region.

d. Calculate the allowable pressure stress intensity factor, K_{IP}, using the following relationship:

$$K_{IP} = (K_{IC} - K_{IT})/SF$$

where: K_{IP} = allowable pressure stress intensity factor (ksi$\sqrt{\text{inch}}$)

SF = safety factor

= 1.5 for pressure test conditions

= 2.0 for normal operation heatup/cooldown conditions (Level A/B)

For the upper flange region, the expression also includes an additional term that subtracts the preload stress intensity factor (multiplied by SF) from the numerator of the equation.

e. Compute the allowable pressure, P, from the allowable pressure stress intensity factor, K_{IP}, using either the Code Case N-588 [1] formula (for the beltline) or alternate analytical values for other locations.

f. Make adjustments for temperature and/or pressure uncertainties and hydrostatic head to $T_{1/4t}$ and P, respectively.

g. Repeat steps (a) through (f) for other temperatures to generate a series of P-T points.

3.2 Adjustments to the Curves

The following additional requirements were used to define the P-T curves. These limits are established in Reference [4]:

For Pressure Test Conditions (Curve A):

- If the pressure is greater than 20% of the pre-service hydrotest pressure, the temperature must be greater than RT_{NDT} of the limiting flange material + 90°F.
- If the pressure is less than or equal to 20% of the pre-service hydrotest pressure, the minimum temperature is conservatively taken as greater than or equal to the RT_{NDT} of the limiting flange material + 60°F. This limit has been a standard GE recommendation for the BWR industry for non-ductile failure protection.

For Core Not Critical Conditions (Curve B):

- If the pressure is greater than 20% of the pre-service hydrotest pressure, the temperature must be greater than RT_{NDT} of the limiting flange material + 120°F.
- If the pressure is less than or equal to 20% of the pre-service hydrotest pressure, the minimum temperature is conservatively taken as greater than or equal to the RT_{NDT} of the limiting flange material + 60°F. This limit has been a standard GE recommendation for the BWR industry for non-ductile failure protection.

For Core Critical Conditions (Curve C):

- The core critical P-T limits must be 40°F above any Pressure Test or Core Not Critical curve limits. Core Not Critical conditions are more limiting than Pressure Test conditions, so Core Critical conditions are equal to Core Not Critical conditions plus 40°F. In addition, when pressure is less than or equal to 20% of the pre-service hydro test pressure and water level is in the normal range for power operation, the minimum temperature must be greater than or equal to the RT_{NDT} of the limiting flange material + 60°F.

- At pressures above 20% of the pre-service hydro test pressure, the minimum Core Critical curve temperature must be at least that required for the in-service pressure test (taken as 1,100 psig), or 160°F above the highest RT_{NDT} of the vessel flange region. As a result of these requirements, the Core Critical curve must have a step at a pressure equal to 20% of the pre-service hydro pressure to

the temperature required by the Pressure Test curve at 1,100 psig, or Curve B + 40°F, whichever is greater.

The resulting pressure and temperature points constitute the P-T curves. These curves relate the minimum required monitored temperature to the allowable reactor pressure. Applicable temperature and pressure adjustments (described below) are also included in Curves A, B, and C.

The lower head area of a BWR, due to convection cooling, stratification, and cool CRD flow is subject to lower temperatures than the balance of the pressure vessel. In addition, the RT_{NDT} of the lower head is much lower than the assumed ART_{NDT} being used for the beltline. The lower head is also not subject same high level of stress as the flange and feedwater nozzle regions. Therefore, separate curves were provided for the lower head. These curves are less restrictive than the enveloping curve used for the beltline and the balance of the vessel. This will provide Operator's with a more accurate data for assessment of PT limits for this cooler region.

3.3 Instrument Uncertainty and Hydrostatic Head

A conservative evaluation of instrument uncertainty by VY derived the following bounding error due to instruments:

Temperature: ±10F
Pressure: ± 30 psig

Thus, the derived P-T curves were shifted to the right by 10°F. When adjusted for the maximum effects of hydrostatic head (from the top head), the resulting pressure margins are shown in Table 4, where the conservatively adjusted margins are used in the P-T curves.

3.4 Beltline Evaluation

For the beltline evaluation, the equations in Code Case N-588 are used to predict the stress intensity factors and temperature shifts for inside and outside 1/4T flaws. For the cooldown, K_{IC} was conservatively based on reactor temperature; for heatup, the ASME Section XI, Appendix G methods for estimation of temperature at the 3/4T point in the wall were used. Tables 5-8 provide detailed results for the calculations.

3.5 Flange Region

For the flange evaluation, membrane and bending stresses were extracted from the original vessel stress report for pressure, preload and thermal expansion (heatup/cooldown) loadings. The critical location was determined to be weld region between the upper head and the head flange [13]. Stress intensity factors were calculated based on the equations similar to Code Case N-588 for membrane and bending stress except that actual stresses were substituted for the pressure stresses in the Code Case. For this region, notes have been added to the P-T curves requiring that the minimum of the fluid or the measured vessel flange skin temperatures be used; thus this temperature may conservatively be used to compute K_{IC}. At temperatures in excess of the 10CFR50 Appendix B limits, the P-T limits based on the flange are much higher than those resulting from the beltline. Tables 9 and 10 provide detailed results for the critical cases (without the margins discussed in Section 3.2).

3.5 N4 Feedwater Nozzle

For the feedwater nozzle, the assessment did not consider heatup and cooldown, but considered the effects of injection of 50°F feedwater into the nozzle at various reactor temperatures, this being the minimum realistic temperature for establishing flow into the feedwater nozzles. The stress intensities for pressure and for the feedwater injection were taken from previous analysis in support of VY's NUREG-0619 feedwater nozzle inspection interval evaluation. For this evaluation, a 1/8T flaw at the feedwater nozzle blend radius

region (1.0 inches base metal, 1.1875 inches including the cladding) was evaluated. This is considerably larger than the 0.823 maximum allowable flaw size (including cladding) that determines the blend radius inspection interval at VY and has been accepted by the NRC [14]. K_{IC} for the thermal shock transient was conservatively based on the mean of the injected feedwater and the reactor temperature, whereas the initial temperature is steady state at reactor temperature. The deepest point of the postulated blend radius would actually be slightly more affected by reactor temperature due to the larger exposed area for heat transfer. The results are shown in Table 11.

3.6 N2 Recirculation Nozzle

This nozzle was evaluated because of the relatively high RT_{NDT} of one of the nozzles. An evaluation, based on the similar FW nozzle analysis discussed above, was conducted to determine a conservative stress intensity factor for a 1/4T nozzle corner crack. Cooldown was the only condition evaluated since the postulated flaw is at the inside surface in the nozzle blend radius. No credit was taken for the difference between the fluid temperature and the crack-tip temperature in computing K_{IC}. The results are shown in Table 10 and show that significant margin exists.

3.7 Bottom Head

The bottom head evaluation was conducted with methods similar to that for the beltline region. Since the bottom head has the control rod drive penetrations, the stresses and stress intensity factors were modified. An evaluation of the effects of the effects of the penetrations showed that the membrane stresses in the bottom head could be bounded by using a factor of 2.75 times the nominal stress computed for the spherical bottom head. Then, the stress intensity factors were multiplied by a factor of 1.28 based on assuming a flaw aspect ratio (a/L) of zero instead of a 1/6 aspect ratio flaw traditionally utilized for ASME Appendix G evaluations. This approach conservatively accounted for the fact that elliptical cracks could potentially interact

with the CRD penetrations in the bottom head region. For the bottom head, the P-T curves were based on the minimum of the bottom head fluid or the measured outside surface temperatures, such that K_{IC} is based on a minimum temperature.

Alternate evaluations were conducted to show that anticipated operating occurrences would not control for the bottom head region. Of significance to a BWR is a reactor scram with recirculation trip. For this transient, the lower head region can cool relatively quickly from normal reactor temperature. Then, if recirculation pumps are restarted, the relatively colder water in the bottom head can be swept out by hot water from the bottom head region.

- For the cooldown transients, a transient was synthesized that bounded data taken from a reactor scram transient at VY and another BWR plant. It included cooldown for 527°F to 375°F in 10 minutes, then a 200°F/hr cooldown to 175°F, followed by a 100°F/hr cooldown. This transient showed that the limiting high pressure was 1050 psig (with margins) at the end of the initial rapid cooldown period, and that the low temperature portion of the cooldown was essentially the same as that based on the normal P-T cooldown evaluations. The resulting allowable pressure versus bottom head fluid temperature for an inside 1/4T flaw is shown in Figure 1. This evaluation is conservative since 1) there is normally a slight depressurization following a reactor scram, and 2) the initial assumed cooldown was significantly more severe than experienced at VY.
- For the recirculation pump restart transient, the maximum possible pressure and temperature conditions of the water sweeping the bottom head region are at saturated conditions, coming from the upper vessel region. Analysis was conducted to evaluate a transient temperature and stress intensity factor for an outside 1/4T flaw due to a step-change transient in the bottom head. Then, using these results, a limiting step change from any initial bottom head temperature to saturated steam conditions could be iteratively determined such that the K_{IC} would not be exceeded at the assumed flaw. The results are shown in Figure 2. Additional pressure margin would be available

above 350°F, since the maximum possible value of the step-change temperature difference starts to decrease as a result of BWR operating pressure and temperatures conditions. Also shown on the curve is the expected pressure based on a maximum recommended top-to-bottom temperature difference of 145°F between the top and bottom head region temperatures for recirculation pump start, as recommended in GE Service Information Letter (SIL) 251 [15]. This shows that there is significant margin between the fracture limiting pressure and the pressures expected when using the SIL as a guideline for when the recirculation pumps may be restarted.

4.0 P-T Curves

The resulting P-T curves, including the 10CRF 50 margins discussed in Section 3.2 are shown in Figures 3 through 5.

5.0 References

1. ASME Boiler and Pressure Vessel Code, Code Case N-588, "Attenuation to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels," Section XI, Division 1, Approved December 12, 1997.

2. ASME Boiler and Pressure Vessel Code, Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves," Section XI, Division 1, Approved February 26, 1999.

3. ASME Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, Nonmandatory Appendix G, "Fracture Toughness Criteria for Protection Against Failure," 1995 Edition, Summer 1996 Addenda.

4. U. S. Code of Federal Regulations, Title 10, Part 50, Appendix G, "Fracture Toughness Requirements," December 1995.

5. Branch Technical Position - MTEB 5-2, "Fracture Toughness Requirements", July 1981, Rev. 1.

6. Pressure Vessel Record Exhibit E "As Fabricated Test Reports," CB&I Contract 9-6201.

7. Pressure Vessel Record Exhibit D "Certified Test Reports," CB&I Contract 9-6201.

8. Battelle Columbus Report BCL-585-84-1, "Testing of Unirradiated Pressure Vessel Surveillance Baseline Specimens for the Vermont Yankee Nuclear Generating Plant," 3/21/84.

9. Battelle Columbus Report BCL-585-84-3, "Examination, Testing and Evaluation of Irradiated Pressure Vessel Surveillance Specimens from the Vermont Yankee Nuclear Power Station," 8/15/84.

10. Letter from Vermont Yankee Nuclear Power Corporation BVY 89-113, to U.S. NRC, "Proposed Change to Revise the Reactor Vessel Pressure-Temperature Curves in the Vermont Yankee Technical Specifications (Generic Letter 88-11)," 11/10/89.

11. Letter from Nuclear Regulatory Commission, NVY 90-077 to Vermont Yankee Nuclear Power Corporation, "Issuance of Amendment No. 120 To Facility Operating License No. DPR-28 – Vermont Yankee Nuclear Power Station (Tac No. 75499).

12. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.99, Revision 2, May 1988.

13. Chicago Bridge & Iron Company Stress Report # 9-6201-I, Volume 3, Vermont Yankee Reactor Vessel, Revision 6, 1/06/71, S.I. File No. VY-04Q-205.

14. Letter from U.S. Nuclear Regulatory Commission NVY 95-02, "Evaluation of the Request for Relief From NUREG-0619 for Vermont Yankee Nuclear Power Station (TAC No. M88803)," 2/6/95.

15. GE Service Information Letter (SIL) No. 251, "Control of RPV Bottom Head Temperature," 10/31/77.

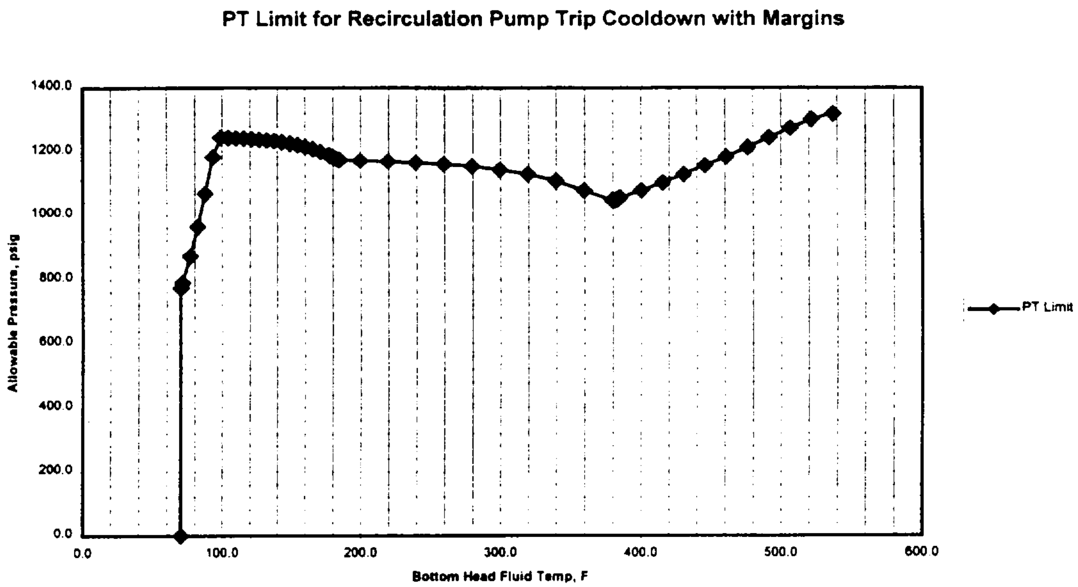


Figure 1: Bottom Head Recirculation Pump Trip Pressure/Temperature Limit Curve

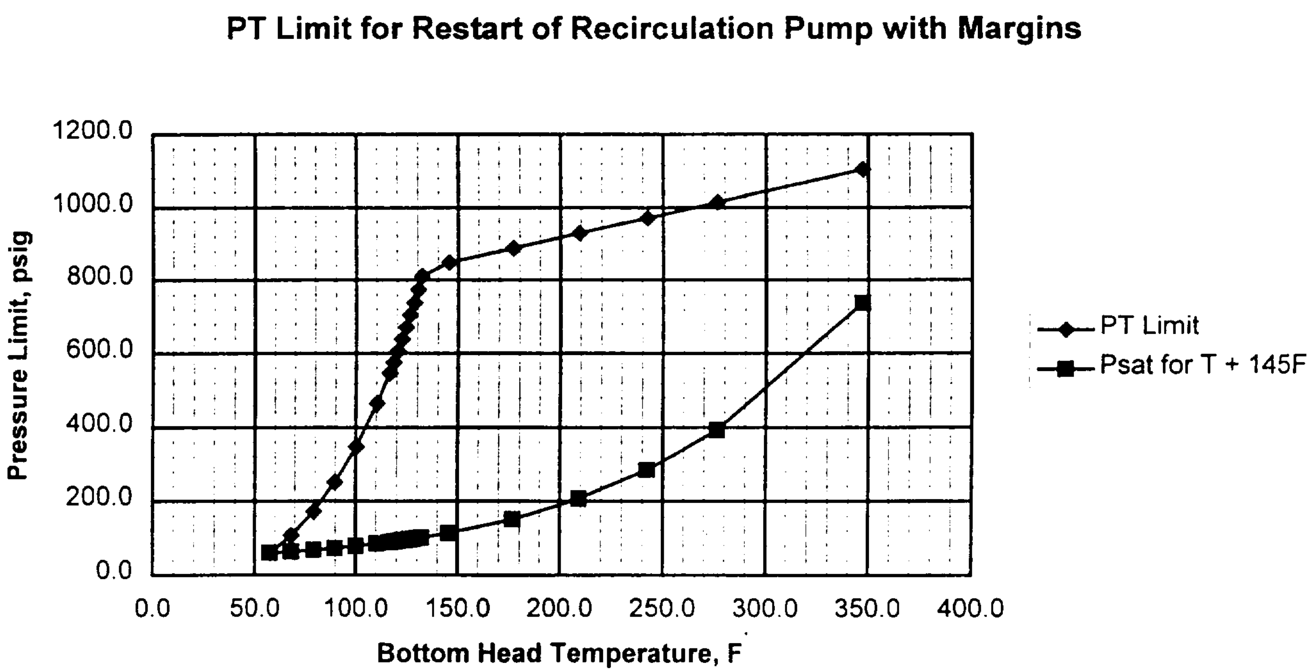
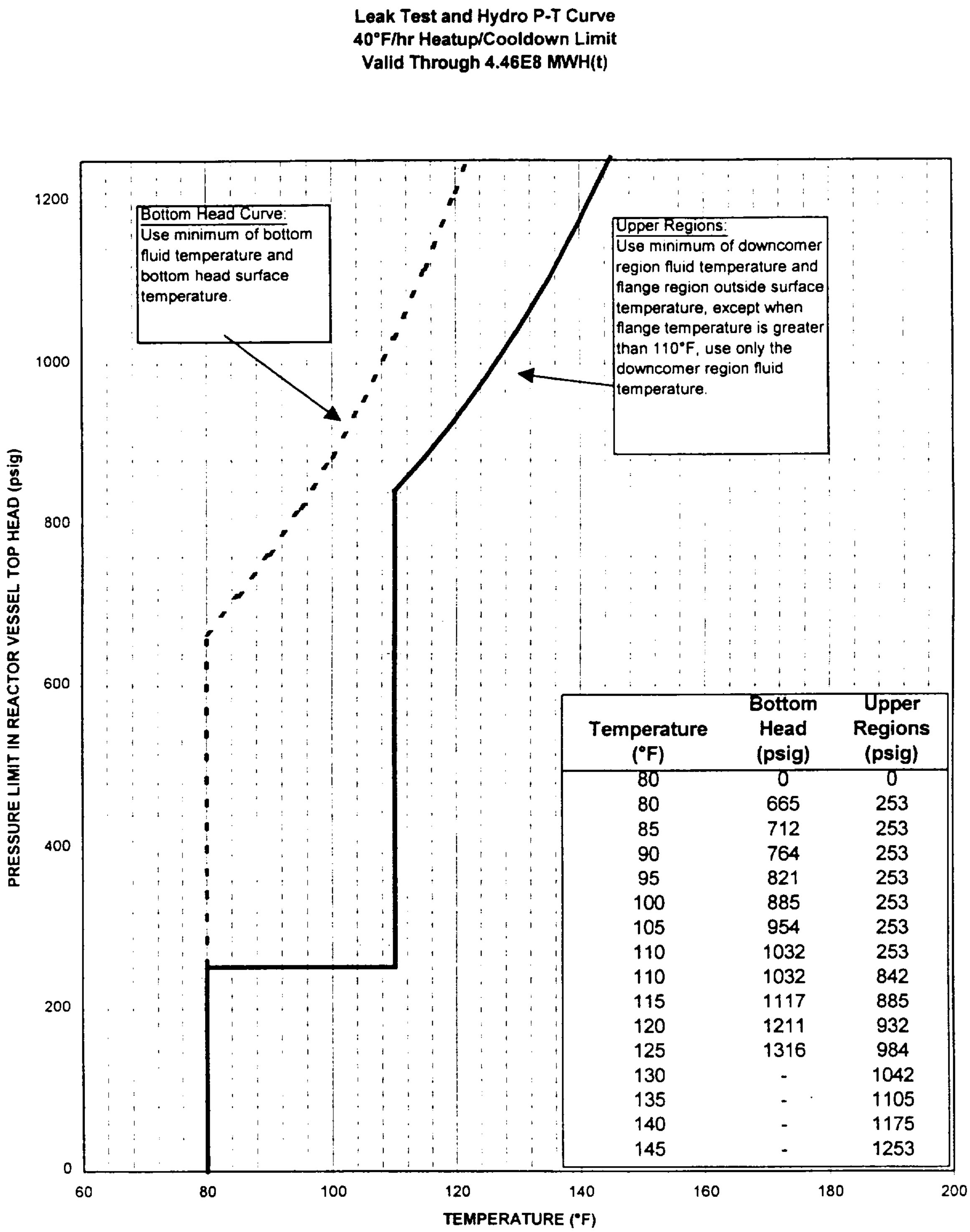
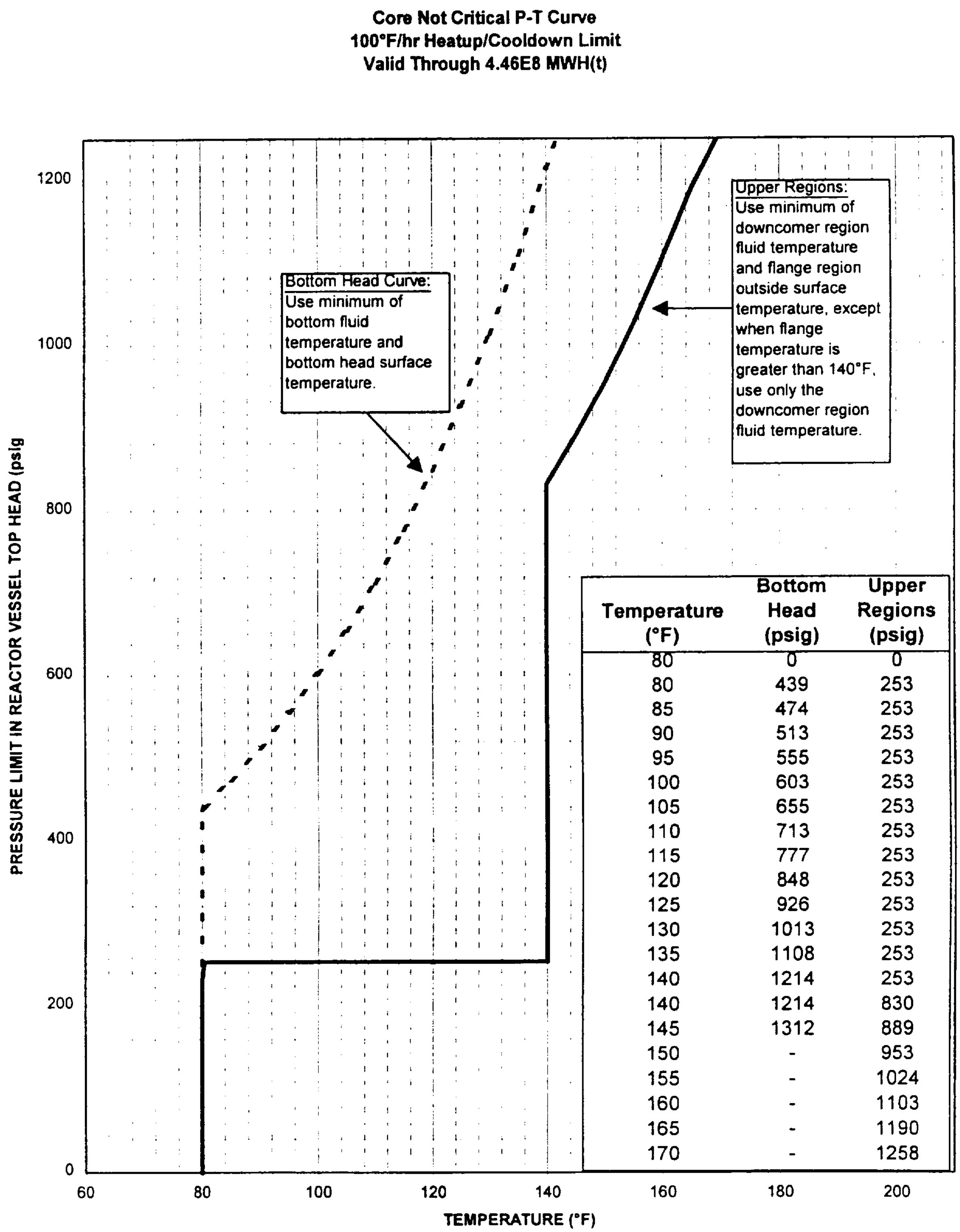


Figure 2. Pressure/Temperature Limit Curve for Recirculation Pump Start



Temperature (°F)	Bottom Head (psig)	Upper Regions (psig)
80	0	0
80	665	253
85	712	253
90	764	253
95	821	253
100	885	253
105	954	253
110	1032	253
110	1032	842
115	1117	885
120	1211	932
125	1316	984
130	-	1042
135	-	1105
140	-	1175
145	-	1253

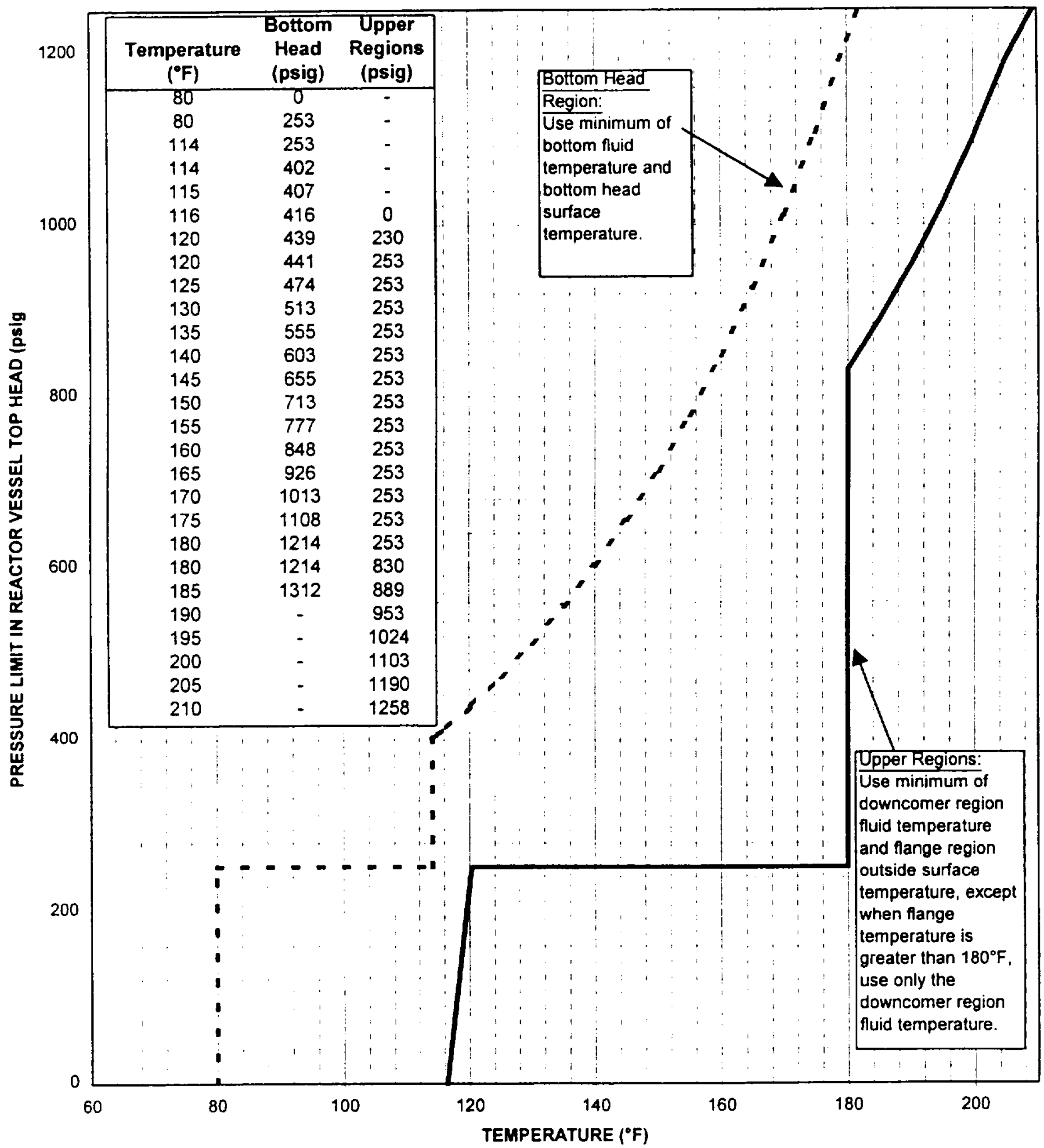
Figure 3: Pressure Test P-T Curve (Curve A)



Temperature (°F)	Bottom Head (psig)	Upper Regions (psig)
80	0	0
80	439	253
85	474	253
90	513	253
95	555	253
100	603	253
105	655	253
110	713	253
115	777	253
120	848	253
125	926	253
130	1013	253
135	1108	253
140	1214	253
140	1214	830
145	1312	889
150	-	953
155	-	1024
160	-	1103
165	-	1190
170	-	1258

Figure 4: Core Not Critical P-T Curve (Curve B)

Core Critical P-T Curve
100°F/hr Heatup/Cooldown Limit
If Pressure < 253 psig, Water Level must be within
Normal Range for Power Operation
Valid Through 4.46E8 MWH(t)



Temperature (°F)	Bottom Head (psig)	Upper Regions (psig)
80	0	-
80	253	-
114	253	-
114	402	-
115	407	-
116	416	0
120	439	230
120	441	253
125	474	253
130	513	253
135	555	253
140	603	253
145	655	253
150	713	253
155	777	253
160	848	253
165	926	253
170	1013	253
175	1108	253
180	1214	253
180	1214	830
185	1312	889
190	-	953
195	-	1024
200	-	1103
205	-	1190
210	-	1258

Figure 5: Core Critical P-T Curve (Curve C)

Table 1 – Initial RT_{NDT} for Materials in Vermont Yankee Reactor Vessel

Region	Material Location	Initial RT_{NDT}, °F
Top Head	Top Head Dollar 1-1	0
Flange Region	Top Head Knuckle 1-5/7	0
	Top Head Knuckle 1-2/4	0
	Top Head Flange	10
	Vessel Shell Flange	10
	Upper (#4) Shell 1-10	0
	Upper(#4) Shell 1-11	0
Intermediate Shell Region	Upper Int. (#3) Shell 1-12	10
	Upper Int. (#3) Shell 1-13	60
Irradiated Shell Region Adjacent to Core	Lower Int. (#2) Shell 1-14	30[1]
	Lower Int. (#2) Shell 1-15	-10
	Lower (#1) Shell 1-16	0
	Lower (#1) Shell 1-17	0
Bottom Head Region	Skirt Knuckle 17-1	40
	Bottom Head Knuckle 1-18/21	30
	Bottom Head Knuckle 1-22/25	0
	Bottom Head Dollar 1-26	30[2]
	Bottom Head Dollar 1-27	0[2]
	Bottom Head Dollar 1-28	30[2]
Nozzles	Recirculation Nozzle N2B	60
	Nozzles (All Others, Incl. Feedwater)	40
All Areas	Welds	0

1. Limiting beltline plate used in initial surveillance capsule evaluation [9]
2. Bottom head dollar plate includes all bottom head control rod drive penetrations

Table 2 - Evaluation of Shift in RT_{NDT} for Core Region Plates

Beltline Plate	I-14	I-15	I-16	I-17	Weld
Initial RT_{NDT},°F	30	-10	0	0	0
Cu w/%	0.11	0.14	0.13	0.12	0.04
Ni w/%	0.63	0.66	0.59	0.61	1.00
Chemistry Factor	74	102	91	83	54
ΔRT_{NDT},°F (1/4T)	11.5	15.8	14.1	12.9	8.4
ΔRT_{NDT}, °F (3/4T)	7.7	10.6	9.5	8.6	5.6
σ_{Δ}° F (1/4T	5.7	7.9	7.1	6.4	4.2
σ_{Δ}° F (3/4T)	3.8	5.3	4.7	4.3	2.8
σ_i°F	0.0	0.0	0.0	0.0	0.0
ART_{NDT},°F (1/4T)	53.0	21.6	28.2	25.8	16.8
ART_{NDT}, °F (3/4T)	45.4	11.2	18.9	17.2	11.2

Based on ID Fluence = 2.3×10^{17} n/cm^2
1/4T Fluence = 1.7×10^{17} n/cm^2
3/4T Fluence = 9.2×10^{16} n/cm^2

Table3. Calculation of Equivalent Peak Beltline Fluence Values

Parameters	Units	Regulatory Guide 1.99 fluence that matches ART_{NDT} used by VY		
Plate		1-14	1-15	1-16
Equivalent Factor on Fluence, k	-	5.37	14.5	11.5
Effective Operating Duration	EFPY	32 EFPY	32 EFPY	32 EFPY
Effective Inside Surface Fluence Value=k*2.3x10^{17}	n/cm^2	1.24E+18	3.34E+18	2.65E+18
Vessel Thickness	Inches	5.06	5.06	5.06
Fluence at 1/4 thickness	n/cm^2	9.12E+17	2.46E+18	1.95E+18
Fluence at 3/4 thickness	n/cm^2	4.97E+17	1.34E+18	1.06E+18
Initial RT_{NDT}	°F	30	-10	0
Chemistry Factor, CF	-	74	102	91
Delta RT_{NDT} @ 1/4 T	°F	29.5	63.3	51.3
Delta RT_{NDT} @ 3/4 T	°F	21.6	48.8	39.1
σ_i, Standard Deviation of Initial RT_{NDT}	°F	0.0	0.0	0.0
Margin@ 1/4T=2*SQRT($\sigma_\Delta^2+\sigma_i^2$)	°F	29.5	34.0	34.0
σ_Δ, Standard Deviation of ΔRT_{NDT} @ 1/4T	°F	14.7	17.0	17.0
Margin@ 3/4T=2*SQRT($\sigma_\Delta^2+\sigma_i^2$)	°F	21.6	34.0	34.0
σ_Δ, Standard Deviation of ΔRT_{NDT} @ 3/4T	°F	10.8	17.0	17.0
Adjusted RT_{NDT} @ 1/4T	°F	89.0	87.3	85.3
Adjusted RT_{NDT} @ 3/4T	°F	73	73	73

NOTE: σ_Δ lesser value of 17°F or ½ ΔRT_{NDT}

Table 4. Pressure Margins at Locations of Interest

Location	Instrument Uncertainty, psi	Static Head Pressure, psi	Total Margin Calculated, psi	Total Margin Used, psi
Closure Head Flange	30	3.72	33.72	35.0
N4 FW Nozzle	30	10.54	10.54	45.0
Bottom of Core Region	30	19.87	19.87	50.0
N2 Recirculation Nozzle	30	20.65	20.65	55.0
Bottom Head	30	27.36	27.36	60.0

Table 5 – P-T Evaluation - Beltline Hydrostatic Test (Heatup)

Pressure-Temperature Curve Calculation

(Pressure Test w/ Heatup = Curve A)

Inputs:

Plant =	**Yankee**	
Component =	**Beltline**	
Vessel thickness, t =	**5.0600**	inches, so $\sqrt{t}$ = 2.249 $\sqrt{inch}$
Vessel Radius, R =	**103.1875**	inches
ART_{NDT} =	**73.0**	°F
Heatup Rate, HU =	**40**	°F/hr
K_{IT} =	**1.73**	ksi*inch$^{1/2}$ (From N-588 for cooldown rate above)
M_T =	**0.26**	(From App G, Fig. G-2214-1)
$\Delta T_{1/4t}$ =	**6.1**	°F = (K_{IT}/M_T) * 0.92 using Figs G-2214-1 & G-2214-2
Safety Factor =	**1.50**	(for hydrotest)
M_m =	**2.009**	(From N-588, for inside surface axial flaw)
Temperature Adjustment =	**10.0**	°F
Pressure Adjustment =	**50.0**	psig (hydrostatic pressure + Uncertainty)

Fluid Temperature T (°F)	1/4t Temperature (°F)	K_{IC} (ksi*inch$^{1/2}$)	K_{IP} (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
50.0	43.9	44.78	28.69	700	60.0	650
55.0	48.9	45.99	29.51	720	65.0	670
60.0	53.9	47.34	30.40	742	70.0	692
65.0	58.9	48.83	31.39	766	75.0	716
70.0	63.9	50.47	32.49	793	80.0	743
75.0	68.9	52.29	33.70	823	85.0	773
80.0	73.9	54.29	35.04	855	90.0	805
85.0	78.9	56.51	36.52	891	95.0	841
90.0	83.9	58.96	38.15	931	100.0	881
95.0	88.9	61.67	39.96	975	105.0	925
100.0	93.9	64.67	41.96	1024	110.0	974
105.0	98.9	67.98	44.16	1078	115.0	1,028
110.0	103.9	71.64	46.60	1138	120.0	1,088
115.0	108.9	75.68	49.30	1203	125.0	1,153
120.0	113.9	80.15	52.27	1276	130.0	1,226
125.0	118.9	85.08	55.57	1356	135.0	1,306

Table 6 – P-T Evaluation - Beltline Hydrostatic Test (Cooldown)

Pressure-Temperature Curve Calculation

(Pressure Test w/ Cooldown = Curve A)

Inputs:

Plant =	**Yankee**	
Component =	**Beltline**	
Vessel thickness, t =	**5.0600**	inches, so $\sqrt{t}$ = 2.249 $\sqrt{inch}$
Vessel Radius, R =	**103.1875**	inches
ART_{NDT} =	**89.0**	°F
Cooldown Rate, CR =	**40**	°F/hr
K_{IT} =	**2.20**	ksi*inch$^{1/2}$ (From N-588, for cooldown rate above)
M_T =	**0.26**	(From App G, Fig. G-2214-1)
$\Delta T_{1/4t}$ =	**3.7**	°F = (K_{IT}/M_T) * 0.44 using Figs. G-2214-1 & G-2214-2
Safety Factor =	**1.50**	(for hydrotest)
M_m =	**2.083**	(From N-588, for inside surface axial flaw)
Temperature Adjustment =	**10.0**	°F
Pressure Adjustment =	**50.0**	psig (hydrostatic pressure + Uncertainty)

Fluid Temperature T (°F)	1/4t Temperature (°F)	K_{IC} (ksi*inch$^{1/2}$)	K_{IP} (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
50.0	50.0	42.70	27.01	636	60.0	586
55.0	55.0	43.70	27.67	651	65.0	601
60.0	60.0	44.81	28.41	669	70.0	619
65.0	65.0	46.03	29.22	688	75.0	638
70.0	70.0	47.38	30.12	709	80.0	659
75.0	75.0	48.87	31.12	733	85.0	683
80.0	80.0	50.52	32.22	758	90.0	708
85.0	85.0	52.34	33.43	787	95.0	737
90.0	90.0	54.35	34.77	819	100.0	769
95.0	95.0	56.58	36.25	853	105.0	803
100.0	100.0	59.04	37.89	892	110.0	842
105.0	105.0	61.75	39.71	935	115.0	885
110.0	110.0	64.76	41.71	982	120.0	932
115.0	115.0	68.08	43.92	1034	125.0	984
120.0	120.0	71.74	46.37	1092	130.0	1,042
125.0	125.0	75.80	49.07	1155	135.0	1,105
130.0	130.0	80.28	52.05	1225	140.0	1,175
135.0	135.0	85.23	55.35	1303	145.0	1,253

Table 7 – P-T Evaluation - Beltline Level A/B (Heatup)

Pressure-Temperature Curve Calculation

(Core Not Critical/ Heatup = Curve B)

Inputs:

Plant = **Yankee**
Component = **Beltline**
Vessel thickness, t = **5.0600** inches, so $\sqrt{t}$ = 2.249 $\sqrt{\text{inch}}$
Vessel Radius, R = **103.1875** inches
ART_{NDT} = **73.0** °F
Heatup Rate, HU = **100** °F/hr
K_{IT} = **4.34** ksi*inch$^{1/2}$ (From N-588 for heatup rate above)
M_T = **0.26** (From App G, Fig. G-2214-1)
$\Delta T_{1/4t}$ = **15.3** °F = (K_{IT}/M_T) * 0.92 using Figs G-2214-1 & G-2214-2
Safety Factor = **2.00** (for level A/B)
M_m = **2.009** (From N-588, for outside surface axial flaw)
Temperature Adjustment = **10.0** °F
Pressure Adjustment = **50.0** psig (hydrostatic pressure + uncertainty)

Fluid Temperature T (°F)	1/4t Temperature (°F)	K_{IC} (ksi*inch$^{1/2}$)	K_{IP} (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
50.0	34.7	42.83	19.25	470	60.0	420
55.0	39.7	43.84	19.75	482	65.0	432
60.0	44.7	44.96	20.31	496	70.0	446
65.0	49.7	46.20	20.93	511	75.0	461
70.0	54.7	47.57	21.61	528	80.0	478
75.0	59.7	49.08	22.37	546	85.0	496
80.0	64.7	50.75	23.20	566	90.0	516
85.0	69.7	52.59	24.13	589	95.0	539
90.0	74.7	54.63	25.15	614	100.0	564
95.0	79.7	56.89	26.27	641	105.0	591
100.0	84.7	59.38	27.52	672	110.0	622
105.0	89.7	62.13	28.90	705	115.0	655
110.0	94.7	65.17	30.42	743	120.0	693
115.0	99.7	68.53	32.10	784	125.0	734
120.0	104.7	72.25	33.96	829	130.0	779
125.0	109.7	76.36	36.01	879	135.0	829
130.0	114.7	80.90	38.28	934	140.0	884
135.0	119.7	85.91	40.79	996	145.0	946
140.0	124.7	91.46	43.56	1063	150.0	1,013
145.0	129.7	97.58	46.62	1138	155.0	1,088
150.0	134.7	104.36	50.01	1221	160.0	1,171
155.0	139.7	111.84	53.75	1312	165.0	1,262

Table 8 – P-T Evaluation - Beltline Level A/B (Cooldown)

Pressure-Temperature Curve Calculation

(Core Not Critical/ Cooldown = Curve B)

Inputs:

Plant =	Yankee	
Component =	Beltline	
Vessel thickness, t =	5.0600	inches, so $\sqrt{t}$ = 2.249 $\sqrt{inch}$
Vessel Radius, R =	103.1875	inches
ART_{NDT} =	89.0	°F
Cooldown Rate, CR =	100	°F/hr
K_{IT} =	5.49	ksi*inch$^{1/2}$ (From N-588, for cooldown rate above)
M_T =	0.26	(From App G, Fig. G-2214-1)
$\Delta T_{1/4t}$ =	9.3	°F = (K_{IT}/M_T) * 0.44 using Figs. G-2214-1 & G-2214-2
Safety Factor =	2.00	(for level A/B)
M_m =	2.083	(From N-588, for inside surface axial flaw)
Temperature Adjustment =	10.0	°F
Pressure Adjustment =	50.0	psig (hydrostatic pressure + uncertainty)

Fluid Temperature T (°F)	1/4t Temperature (°F)	K_{IC} (ksi*inch$^{1/2}$)	K_{IP} (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
50.0	50.0	42.70	18.61	438	60.0	388
55.0	55.0	43.70	19.11	450	65.0	400
60.0	60.0	44.81	19.66	463	70.0	413
65.0	65.0	46.03	20.27	477	75.0	427
70.0	70.0	47.38	20.95	493	80.0	443
75.0	75.0	48.87	21.69	511	85.0	461
80.0	80.0	50.52	22.51	530	90.0	480
85.0	85.0	52.34	23.43	551	95.0	501
90.0	90.0	54.35	24.43	575	100.0	525
95.0	95.0	56.58	25.54	601	105.0	551
100.0	100.0	59.04	26.77	630	110.0	580
105.0	105.0	61.75	28.13	662	115.0	612
110.0	110.0	64.76	29.63	698	120.0	648
115.0	115.0	68.08	31.29	737	125.0	687
120.0	120.0	71.74	33.13	780	130.0	730
125.0	125.0	75.80	35.15	828	135.0	778
130.0	130.0	80.28	37.39	880	140.0	830
135.0	135.0	85.23	39.87	939	145.0	889
140.0	140.0	90.70	42.61	1003	150.0	953
145.0	145.0	96.75	45.63	1074	155.0	1,024
150.0	150.0	103.43	48.97	1153	160.0	1,103
155.0	155.0	110.82	52.66	1240	165.0	1,190
160.0	160.0	118.98	56.75	1336	170.0	1,286

Table 9 – P-T Evaluation - Flange Hydrostatic Test (Heatup)

Pressure-Temperature Curve Calculation

(Pressure Test - Upper Flange 2 - Heatup)

Inputs:

Plant = Yankee
Component = Upper Flange 2 Upper Flange/Hub Intersection Axial Flaw
Vessel thickness, t = N/A inches
Vessel Radius, R = NA inches
ART_{NDT} = 10.0 °F ======> All EFPYs
K_{1T+1P_L} = 47.77 ksi*inch$^{1/2}$
Safety Factor = 1.50 (for hydrotest)
K_{1P} for 1000 psig = 10.30 ksi*inch$^{1/2}$
Temperature Adjustment = 10.0 °F
Pressure Adjustment = 35.0 psig (hydrostatic pressure + Uncertainty)

K, ksi*inch$^{1/2}$
Preload = 45.7
Thermal = 2.072

Fluid Temperature T (°F)	1/4t Temperature (°F)	K_{IC} (ksi*inch$^{1/2}$)	K_{IP} (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
64	64.0	94.25	0.52	51	74	16
65	65.0	95.49	1.34	131	75	96
66	66.0	96.75	2.18	212	76	177
67	67.0	98.03	3.04	295	77	260
68	68.0	99.34	3.91	380	78	345
69	69.0	100.68	4.80	466	79	431
70	70.0	102.04	5.71	555	80	520
71	71.0	103.43	6.64	645	81	610
72	72.0	104.85	7.58	736	82	701
73	73.0	106.30	8.55	830	83	795
74	74.0	107.77	9.53	926	84	891
75	75.0	109.28	10.54	1023	85	988
76	76.0	110.82	11.56	1123	86	1088
77	77.0	112.38	12.61	1224	87	1189
78	78.0	113.98	13.67	1328	88	1293

Table 10 – P-T Evaluation - Flange Level A/B (Heatup)

Pressure-Temperature Curve Calculation

(Core Not Critical - Upper Flange 2- Heatup)

Inputs:

Plant =	Yankee			
Component =	Upper Flange 2	Upper Flange/Hub Intersection Axial Flaw		
Vessel thickness, t =	N/A	inches		
Vessel Radius, R =	NA	inches		
ART_{NDT} =	10.0	°F ======>	All EFPYs	
K_{1T-1PL}	50.88	ksi*inch$^{1/2}$		
Safety Factor =	2.00	(for level A/B)		K, ksi*inch$^{1/2}$
K_{1P} for 1000 psig =	10.30	ksi*inch$^{1/2}$	Preload =	45.7
Temperature Adjustment =	10.0	°F	Thermal =	5.18
Pressure Adjustment =	35.0	psig (hydrostatic pressure + uncertainty)		

Fluid Temperature T (°F)	1/4t Temperature (°F)	K_{IC} (ksi*inch$^{1/2}$)	K_{IP} (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
67	67.0	98.03	0.73	70	77	35
68	68.0	99.34	1.38	134	78	99
69	69.0	100.68	2.05	199	79	164
70	70.0	102.04	2.73	265	80	230
71	71.0	103.43	3.42	333	81	298
72	72.0	104.85	4.13	401	82	366
73	73.0	106.30	4.86	472	83	437
74	74.0	107.77	5.60	543	84	508
75	75.0	109.28	6.35	616	85	581
76	76.0	110.82	7.12	691	86	656
77	77.0	112.38	7.90	767	87	732
78	78.0	113.98	8.70	845	88	810
79	79.0	115.62	9.52	924	89	889
80	80.0	117.28	10.35	1005	90	970
81	81.0	118.98	11.20	1087	91	1052
82	82.0	120.71	12.07	1171	92	1136
83	83.0	122.48	12.95	1257	93	1222
84	84.0	124.28	13.85	1345	94	1310

Table 11 – P-T Evaluation – Feedwater Nozzle Level A/B

Pressure-Temperature Curve Calculation

(Core Not Critical - FW Injection - Corner Nozzle Crack)

Inputs:

Plant =	Yankee			
Component =	FW Nozzle Blend			
Vessel thickness, t =	N/A	inches		
Vessel Radius, R =	N/A	inches		
ART_{NDT} =	40.0	°F ======>	All EFPYs	
K_{IT} for 552F - 50F Step=	106.56	ksi*inch$^{1/2}$	Temp. Change	502 °F Step
Safety Factor =	2.00	(for level A/B)		
K_{IP} for 1025 psig =	33.80	ksi*inch$^{1/2}$		
Temperature Adjustment =	10.0	°F		
Pressure Adjustment =	45.0	psig (hydrostatic pressure + uncertainty)		

Fluid Temperature T (°F)	1/8t Temperature (°F)	K_{IC} (ksi*inch$^{1/2}$)	K_{IP} (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
50	50.0	58.52	29.26	887	60	842
55	52.5	59.82	29.38	891	65	846
60	55.0	61.19	29.53	896	70	851
65	57.5	62.62	29.72	901	75	856
70	60.0	64.13	29.94	908	80	863
75	62.5	65.72	30.21	916	85	871
80	65.0	67.38	30.51	925	90	880
85	67.5	69.14	30.85	936	95	891
90	70.0	70.98	31.24	948	100	903
95	72.5	72.92	31.68	961	105	916
100	75.0	74.95	32.17	976	110	931
105	77.5	77.09	32.71	992	115	947
110	80.0	79.34	33.30	1010	120	965
115	82.5	81.71	33.96	1030	125	985
120	85.0	84.20	34.67	1051	130	1006
125	87.5	86.81	35.45	1075	135	1030
130	90.0	89.56	36.29	1100	140	1055
135	92.5	92.45	37.20	1128	145	1083
140	95.0	95.49	38.19	1158	150	1113
145	97.5	98.68	39.26	1191	155	1146
150	100.0	102.04	40.41	1225	160	1180
155	102.5	105.57	41.64	1263	165	1218
160	105.0	109.28	42.96	1303	170	1258

Table12 – P-T Evaluation – Recirculation Nozzle Level A/B

Pressure-Temperature Curve Calculation

(Core Not Critical – N2 Recirc Nozzle - Cooldown)

Inputs:

Plant =	Yankee		
Component =	N2 Recirc Noz		
Vessel thickness, t =	N/A		
Vessel Radius, R =	N/A		
ART_{NDT} =	60.0	°F ======>	All EFPYs
K_{1T} =	25.07	ksi*inch$^{1/2}$	
Safety Factor =	2.00	(for level A/B)	
K_{IP} for 1025 psig =	44.25	ksi*inch$^{1/2}$	
Temperature Adjustment =	10.0	°F	
Pressure Adjustment =	55.0	psig (hydrostatic pressure + uncertainty)	

Fluid Temperature T (°F)	1/4t Temperature (°F)	K_{IC} (ksi*inch$^{1/2}$)	K_{IP} (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
50	50.0	50.18	25.09	581	60	526
55	55.0	51.96	25.98	602	65	547
60	60.0	53.93	26.97	625	70	570
65	65.0	56.11	28.06	650	75	595
70	70.0	58.52	29.26	678	80	623
75	75.0	61.19	30.59	709	85	654
80	80.0	64.13	32.07	743	90	688
85	85.0	67.38	33.69	780	95	725
90	90.0	70.98	35.49	822	100	767
95	95.0	74.95	37.48	868	105	813
100	100.0	79.34	39.67	919	110	864
105	105.0	84.20	42.10	975	115	920
110	110.0	89.56	44.78	1037	120	982
115	115.0	95.49	47.74	1106	125	1051
120	120.0	102.04	51.02	1182	130	1127
125	125.0	109.28	54.64	1266	135	1211
130	130.0	117.28	58.64	1358	140	1303

Table 13 – P-T Evaluation – Bottom Head Hydrostatic Test (Cooldown)

Pressure-Temperature Curve Calculation

(Pressure Test w/ Cooldown = Curve A)

Inputs:

Plant = Yankee
Component = Bot. Head
Vessel thickness, t = 5.9375 inches, so √t = 2.437 √inch
Vessel Radius, R = 103.1875 inches
ART_{NDT} = 30.0 °F
Cooldown Rate, CR = 40 °F/hr
K_{IT} = 4.19 ksi*inch$^{1/2}$ (From N-588, for cooldown rate above)
M_T = N/A (From App G, Fig. G-2214-1)
$\Delta T_{1/4t}$ = N/A °F = (K_{IT}/M_T) * 0.44 using Figs. G-2214-1 & G-2214-2
Safety Factor = 1.50 (for hydrotest)
Factor = 1.2808 M_m concentration factor
M_m = 2.256 (From N-588, for inside surface axial flaw)
Temperature Adjustment = 10.0 °F
Pressure Adjustment = 60.0 psig (hydrostatic pressure + Uncertainty)

Fluid Temperature T (°F)	1/4t Temperature (°F)	K_{IC} (ksi*inch$^{1/2}$)	K_{IP} (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
50.0	50.0	64.13	39.96	579	60.0	519
55.0	55.0	67.38	42.13	610	65.0	550
60.0	60.0	70.98	44.52	645	70.0	585
65.0	65.0	74.95	47.17	683	75.0	623
70.0	70.0	79.34	50.10	725	80.0	665
75.0	75.0	84.20	53.34	772	85.0	712
80.0	80.0	89.56	56.91	824	90.0	764
85.0	85.0	95.49	60.86	881	95.0	821
90.0	90.0	102.04	65.23	945	100.0	885
95.0	95.0	109.28	70.06	1014	105.0	954
100.0	100.0	117.28	75.39	1092	110.0	1,032
105.0	105.0	126.12	81.29	1177	115.0	1,117
110.0	110.0	135.90	87.80	1271	120.0	1,211
115.0	115.0	146.70	95.00	1376	125.0	1,316

Table 14 – P-T Evaluation – Bottom Head Level A/B (Cooldown)

Pressure-Temperature Curve Calculation

(Core Not Critical/ Cooldown = Curve B)

Inputs:

Plant =	Yankee	
Component =	Bot. Head	
Vessel thickness, t =	5.9375	inches, so $\sqrt{t}$ = 2.437 $\sqrt{\text{inch}}$
Vessel Radius, R =	103.1875	inches
ART_{NDT} =	30.0	°F
Cooldown Rate, CR =	100	°F/hr
K_{IT} =	10.49	ksi*inch$^{1/2}$ (From N-588, for cooldown rate above)
M- =	N/A	(From App G, Fig. G-2214-1)
$\Delta T_{1/4t}$ =	N/A	°F = (K_{IT}/M-) * 0.44 using Figs. G-2214-1 & G-2214-2
Safety Factor =	2.00	(for level A/B)
Factor =	1.2808	M_m concentration factor
M_m =	2.256	(From N-588, for inside surface axial flaw)
Temperature Adjustment =	10.0	°F
Height of Water for a Full Vessel =	N/A	inches
Pressure Adjustment =	60.0	psig (hydrostatic pressure + uncertainty)

Fluid Temperature T (°F)	1/4t Temperature (°F)	K_{IC} (ksi*inch$^{1/2}$)	K_{IP} (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
50.0	50.0	64.13	26.82	388	60.0	328
55.0	55.0	67.38	28.45	412	65.0	352
60.0	60.0	70.98	30.25	438	70.0	378
65.0	65.0	74.95	32.23	467	75.0	407
70.0	70.0	79.34	34.43	499	80.0	439
75.0	75.0	84.20	36.86	534	85.0	474
80.0	80.0	89.56	39.54	573	90.0	513
85.0	85.0	95.49	42.50	615	95.0	555
90.0	90.0	102.04	45.78	663	100.0	603
95.0	95.0	109.28	49.40	715	105.0	655
100.0	100.0	117.28	53.40	773	110.0	713
105.0	105.0	126.12	57.82	837	115.0	777
110.0	110.0	135.90	62.71	908	120.0	848
115.0	115.0	146.70	68.11	986	125.0	926
120.0	120.0	158.63	74.07	1073	130.0	1,013
125.0	125.0	171.83	80.67	1168	135.0	1,108
130.0	130.0	186.40	87.96	1274	140.0	1,214
135.0	135.0	200.00	94.76	1372	145.0	1,312