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BVY 03-29

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 Specifications Proposed Change No. 258
RPV Fracture Toughness and Material Surveillance Requirements

Pursuant to 10CFR50.90, Vermont Yankee[1] (VY) hereby proposes to amend its Facility Operating License, DPR-28, by incorporating the attached proposed change into the VY Technical Specifications. This proposed change adopts the Boiling Water Reactor Vessel and Internals Project Integrated Surveillance Program and updates pressure and temperature limitations for the reactor coolant system.

Attachments 1 and 2 to this letter contain supporting information and the safety assessment for the proposed change. Attachment 3 contains the determination of no significant hazards consideration. Attachment 4 provides a proposed change to the Updated Final Safety Analysis Report regarding the Integrated Surveillance Program. Attachment 5 provides the marked-up version of the current Technical Specification and Bases pages, and Attachment 6 is the retyped Technical Specification and Bases pages.

VY has reviewed the proposed 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), no environmental impact statement or environmental assessment needs to be prepared for this change.

Upon acceptance of this proposed change by the NRC, VY requests that a license amendment be issued prior to the next scheduled refueling outage (Spring 2004) for implementation within 60 days of its effective date. A license amendment is required prior to the end of the next refueling outage because current Technical Specifications for pressure-temperature limitations are only valid through the end of the current operating cycle, and current requirements for the removal of reactor vessel surveillance specimens would necessitate the removal of a surveillance capsule during the next refueling outage. Accordingly, VY respectfully requests timely approval of this license amendment request.

[1] Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. are the licensees of the Vermont Yankee Nuclear Power Station

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If you have any questions on this transmittal, please contact Mr. Len Gucwa at (802) 258-4225.

Sincerely,

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 the Vermont Yankee Nuclear Power Station, that he is duly authorized to execute and file the foregoing document, and that the statements therein are true to the best of his knowledge and belief.

Thomas B. Silko, Notary Public
My Commission Expires February 10, 2007

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. 258

RPV Fracture Toughness and Material Surveillance Requirements

Supporting Information and Safety Assessment of Proposed Change

1.0 INTRODUCTION

1.1 PURPOSE

This Proposed Change to the licensing basis of the Vermont Yankee Nuclear Power Station (VYNPS) revises the Technical Specifications (TS) and Updated Final Safety Analysis Report (UFSAR) regarding reactor pressure vessel (RPV) fracture toughness and material surveillance requirements. The specific changes are summarized as follows:

1.1.1 RPV Material Surveillance Program

Vermont Yankee (VY) is proposing to revise current, plant-specific RPV material surveillance requirements (SRs) by adopting the Boiling Water Reactor Vessel and Internals Project (BWRVIP) RPV integrated surveillance program (ISP) as the basis for demonstrating compliance with the requirements of Appendix H to 10CFR Part 50, "Reactor Vessel Material Surveillance Program Requirements." In a safety evaluation dated February 1, 2002 (Ref. 1), the NRC staff determined that the BWRVIP ISP was an acceptable alternative to existing BWR plant-specific RPV surveillance programs for the purpose of maintaining compliance with the requirements of Appendix H.

1.1.2 Pressure-Temperature Limitations

VY is proposing to update current pressure and temperature (P-T) limit curves for the reactor coolant system that are required by TS 3.6.A, "Pressure and Temperature Limitations." Currently, TS Figures 3.6.1, 3.6.2 and 3.6.3 expire at the end of the current operating cycle. This proposed change updates the pressure and temperature limits for the reactor coolant system through the end of the current operating license. The updated P-T limits are based on a re-calculated RPV neutron fluence using an NRC staff-accepted neutron fluence methodology for boiling water reactors. The revised P-T limit curves are valid through the end of the current operating license or 32 effective full power years (EFPY) and generally satisfy the requirements of Appendix G to 10CFR Part 50, "Fracture Toughness Requirements."

1.2 DESCRIPTION OF THE PROPOSED CHANGE

1.2.1 RPV Material Surveillance Program

Current TS SR 4.6.A.5 (and associated Bases) regarding irradiated reactor vessel surveillance specimens are being revised. Specifically, the plant-specific SR 4.6.A.5 is being removed from TS, and details regarding the BWRVIP ISP (which is being adopted in place of the current plant-specific requirements) are being added to the UFSAR. In addition, conforming changes are being made to the TS Bases for Sections 3.6 and 4.6.

Current TS SR 4.6.A.5 requires:

> *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.

Attachment 4 of this Proposed Change provides a proposed revision to the UFSAR to adopt the provisions of the BWRVIP ISP in place of the existing plant-specific surveillance program. Because the RPV material surveillance program requirements are being relocated from the TS and incorporated into the UFSAR, the proposed change to the UFSAR regarding the ISP is included in Attachment 4 for NRC review.

As noted in proposed UFSAR Table 4.2.4, instead of withdrawing the second surveillance capsule after 30 years of operation, the capsule will be maintained in a "standby" status. Other, changes to the UFSAR which result from the updated P-T calculations are not included in this submittal, but will be made following issuance of a license amendment.

1.2.2 Pressure-Temperature Limitations

Current TS Figures 3.6.1, 3.6.2 and 3.6.3 (and associated Bases), which establish P-T limitations for the reactor coolant system are being updated. The subject figures currently contain a restriction on their use, such that the figures are no longer valid after the end of the current operating cycle (Cycle 23). The updated set of P-T curves is valid through the end of the 40-year operating license and was re-defined based on a re-calculation of neutron fluence using an NRC staff-accepted neutron fluence methodology for BWRs. The updated curves are also clarified as described below. Otherwise, the set of P-T limits remains as shown in current TS Figures 3.6.1, 3.6.2 and 3.6.3. In addition, conforming changes are being made to the TS Bases for Sections 3.6 and 4.6.

Current TS Figures 3.6.1, 3.6.2 and 3.6.3 are being replaced by the figures in Attachment 6. Specific changes entail:

- Figures 3.6.1, 3.6.2 and 3.6.3 currently contain a statement that each is valid through the end of Cycle 23. That validity duration is being changed to 4.46×10^8 megawatt-hours thermal (MWH(t)).

- To improve legibility of the curves, the grid line divisions have been changed, the ordinate axis has been identified by 100 psi increments, and more data were used to plot the curves to improve resolution.

- A Note is being added to TS Figure 3.6.2 to specify requirements for minimum temperature when using local test instrumentation during flange tensioning and detensioning operations. The new Note will specify:

  During tensioning and detensioning operations with the vessel vented and the vessel fluid level below the flange region, the flange temperature may be monitored with test

instrumentation in lieu of process instrumentation for the downcomer region fluid temperature and permanent flange region outside surface temperature. The test instrumentation uncertainty must be less than +/- 2°F. The flange region temperatures must be maintained greater than or equal to 72°F when monitored with test instrumentation during tensioning, detensioning, and when tensioned.

- The tabulation of pressure and temperature data on Figure 3.6.3 is being revised to more accurately reflect the plot of the curves (the curves are unchanged). At 116°F the bottom head pressure is changed to 413 psig, instead of the current 416 psig. At 120°F, there should be only two data points on Figure 3.6.3, and these are at 253 psig for the upper region and at 439 psig for the bottom head region. Therefore, the tabulation corresponding to a temperature of 120°F will only specify pressures of 439 psig and 253 psig for the bottom head region and upper region, respectively.

1.3 SCHEDULE

VY plans to implement the proposed change to support the next refueling outage (i.e., Spring 2004) and subsequent restart. The proposed change involves the elimination of refueling outage work-scope and its approval is needed for post-outage plant restart. Because current TS SR 4.6.A.5 requires that VY remove a RPV material capsule during the next refueling outage, and the current set of P-T curves expires at the end of the current operating cycle (defined as the end of the next refueling outage), a license amendment is required before the end of the refueling outage. The next refueling outage is currently scheduled to commence on April 3, 2004.

2.0 BACKGROUND

To ensure the structural integrity of RPVs, 10CFR50.60, "Acceptance criteria for fracture prevention measures for light water nuclear power reactors for normal operation," imposes the specific fracture toughness and material surveillance program requirements set forth in Appendices G and H to 10CFR Part 50.

2.1 RPV MATERIAL SURVEILLANCE PROGRAM

Licensees of nuclear power plants are required by Appendix H to 10CFR Part 50 to implement RPV material surveillance programs (including the withdrawal and analysis of surveillance capsules) for monitoring changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region which result from neutron irradiation. These programs consist of surveillance capsules installed inside the RPV that include specimens from RPV plate, weld and heat-affected zone materials. These specimens are removed at periodic intervals, tested and analyzed to monitor the radiation embrittlement of the RPV. Appendix H provides two alternative methods for compliance:

The first alternative is the design and implementation of a plant-specific surveillance program that is consistent with ASTM E-185 (Ref. 2). In accordance with this alternative, licensees must comply with either the edition of ASTM E-185 that was current on the issue date of the American Society of Mechanical Engineers (ASME) Code to which the reactor vessel was purchased, or later editions through the 1982 edition as the basis for establishing surveillance capsule withdrawal schedules.

The second alternative is addressed in paragraph III.C of Appendix H to 10CFR50, "Requirements for an Integrated Surveillance Program," and involves the implementation of an integrated surveillance program in lieu of individual plant-specific RPV surveillance programs. Certain technical and regulatory criteria are set forth in paragraph III.C.

Until recently, each BWR has had its own RPV material surveillance program, and the specimen selection, testing, analysis and monitoring were conducted on a plant-specific basis. Over the past several years, the BWRVIP developed an ISP that meets the criteria defined in Appendix H for an ISP. The NRC staff approved the BWRVIP ISP in a safety evaluation (SE), which was provided to the BWRVIP by letter dated February 1, 2002 (Ref. 1).

The NRC SE concluded that the proposed ISP, if implemented in accordance with the conditions of the SE, is an acceptable alternative to all existing BWR plant-specific RPV surveillance programs for the purpose of maintaining compliance with the requirements of Appendix H to 10CFR 50 through the end of current facility 40-year operating licenses. In NRC Regulatory Issue Summary (RIS) 2002-05 (Ref. 3), NRC endorsed the BWRVIP ISP and provided guidance for BWR licensees in implementing the ISP program.

Implementation of the ISP provides certain benefits. When the original surveillance materials were selected for plant-specific surveillance programs, the state of knowledge concerning RPV material response to irradiation and post-irradiation fracture toughness was not as robust as it is today. As a result, many facilities did not include what would be identified today as the plant's limiting RPV materials in their surveillance programs. Hence, the integrated effort to identify and evaluate materials from other BWRs, which may better represent a facility's limiting materials, should improve the overall evaluation of BWR RPV embrittlement. Also, the inclusion of additional data from the testing of BWR Owners Group Supplemental Surveillance Program capsules will improve overall quality of the data being used to evaluate BWR RPV embrittlement. Implementation of the ISP is also expected to reduce the costs associated with removing capsules from RPVs and surveillance testing and analysis, since surveillance materials that are of little or no value (either because they lack adequate unirradiated baseline Charpy V-notch data or because they are not the best representative materials) will no longer be tested. In addition, the exposure of personnel to radiation due to the removal and handling of irradiated specimens should be reduced.

By letter dated November 12, 2002 (Ref. 4), the BWRVIP submitted Proprietary Report BWRVIP-86-A (Ref. 5) to the NRC staff for information and review. BWRVIP-86-A represents a compilation of information from several sources upon which the NRC staff based its SE (Ref. 1). The NRC staff reviewed the information in BWRVIP-86-A and, by letter dated December 16, 2002 (Ref. 6), found that it accurately incorporates all of the relevant information submitted by the BWRVIP to support NRC staff approval of the BWRVIP ISP.

A major consideration in the NRC staff's SE (Ref. 1) deals with BWR RPV fluence calculations. Specifically, the NRC staff required as a condition to its SE that RPV neutron fluence calculations use a fluence methodology that is acceptable to the NRC staff and is consistent with the guidance found in NRC Regulatory Guide 1.190 (Ref. 7). In addition, if differing fluence methodologies are used (i.e., the methodology used to determine the neutron fluence values for a licensee's RPV differs from the methodology used to establish the neutron fluence values of the ISP surveillance capsules which represent the RPV in the ISP), the results of these differing methodologies are compatible (i.e., within acceptable levels of uncertainty).

2.2 P-T LIMITATIONS

2.2.1 Technical and Regulatory Basis

10CFR50.60, "Acceptance criteria for fracture prevention measures for light water nuclear power reactors for normal operation," imposes the fracture toughness requirements for the reactor coolant pressure boundary set forth in Appendix G to Part 50. Licensees of nuclear power plants are required by Appendix G to 10CFR Part 50, "Fracture Toughness Requirements," to develop and use P-T limits in order to provide adequate margins of safety during any condition of operation, including anticipated operational occurrences and system hydrostatic tests, to which the reactor coolant pressure boundary may be subjected over its service lifetime.

Appendix G to 10CFR50 describes the conditions that require P-T limits and provides the general bases for these limits. Operating limits based on the criteria of Appendix G, as defined by applicable regulations, codes, and standards, provide reasonable assurance that non-ductile or rapidly propagating failure will not occur.

Appendix G of Section XI of the ASME Boiler and Pressure Vessel Code (the Code), (Ref. 8) forms the basis for the requirements of Appendix G to 10CFR50. The operating limits for pressure and temperature are required for three categories of operation: (1) hydrostatic pressure tests and leak tests; (2) non-nuclear heatup/cooldown and low-level physics tests; and (3) core critical operation.

Pressure-retaining components of the reactor coolant pressure boundary that are made of ferritic materials (including the pressure vessel) must meet the requirements of Appendix G of the Code, as supplemented by the additional requirements in Table 1 of Appendix G to 10CFR50 for fracture toughness during system hydrostatic tests and any condition of normal operation, including anticipated operational occurrences. In addition to beltline considerations, non-beltline discontinuities such as nozzles, penetrations, and flanges may influence the construction of P-T curves.

The P-T limits are not derived from design basis accident analyses, but are prescribed for all plant modes to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause non-ductile failure of the reactor coolant pressure boundary. The P-T limits are acceptance limits because they preclude operation in an unanalyzed condition.

P-T limits are revised when necessary in accordance with Appendix H to 10CFR50 for changes in adjusted reference temperature for nil ductility transition (ART_{NDT}) due to neutron fluence values determined from the analysis of irradiated RPV beltline materials. Upon acceptance of this Proposed Change, the ISP discussed above will provide the dosimetry data and results of fracture toughness tests as the bases for changes in ART_{NDT} for the VYNPS RPV.

2.2.2 Neutron Fluence Methodology

10CFR50, Appendix G requires the prediction of the effects of neutron irradiation on vessel embrittlement by calculating the ART_{NDT} and the Charpy Upper Shelf Energy (USE). For reactor vessel beltline materials, including welds, plates, and forgings, the values of ART_{NDT} must account for the effects of neutron irradiation, as part of the surveillance program of Appendix H to

10CFR50. To predict these effects, NRC Generic Letter 88-11 (Ref. 9) imposes the use of methods described in Regulatory Guide 1.99, Revision 2 (Ref. 10). The fluence values calculated using the methodology described in Regulatory Guide 1.190 satisfy the requirements of Appendix G to 10CFR50 and Regulatory Guide 1.99.

2.2.3 Flaw Analysis

The basic parameter in Appendix G to Section XI of the ASME Code (Ref. 8) for calculating P-T limit curves is the stress intensity factor (K_{Ia}), which is a function of the stress and a postulated flaw. The Code methodology specifies that licensees determine the reference K_{Ia} factors. Code Case N-640 (Ref. 14) permits use of the lower bound static initiation fracture toughness value (K_{Ic}) in lieu of K_{Ia}.

The methodology of Appendix G to the Code requires that P-T curves satisfy a safety factor of 2.0 on stress intensities arising from primary membrane and bending stresses during normal plant operations (including heatups, cooldowns, and transient operating conditions) and a safety factor of 1.5 on stress intensities arising from primary membrane and bending stresses when leak rate or hydrostatic pressure tests are performed on the reactor coolant system. Table 1 in Appendix G to 10CFR50 provides criteria for meeting P-T limitations of Appendix G to the Code and the minimum temperature requirements for normal and pressure testing operations.

3.0 SAFETY ASSESSMENT

3.1 RPV MATERIAL SURVEILLANCE PROGRAM

VY is a participant in the BWRVIP, which developed the NRC staff-accepted ISP for RPV materials and will formally implement the ISP upon NRC issuance of the requested license amendment.

BWRVIP-86-A (Ref. 5) provides the technical and regulatory basis for the BWRVIP ISP and will be incorporated by reference in the VYNPS UFSAR. As noted in the NRC staff's reply to the BWRVIP dated December 16, 2002 (Ref. 6), reference to BWRVIP-86-A is acceptable in lieu of referencing the separate source documents. Attachment 4 of this proposed change is a proposed revision to the UFSAR, which will become effective upon implementation of the requested license amendment.

The BWRVIP ISP is intended to replace the existing plant-specific RPV material surveillance programs with representative weld and base materials data from host reactors. It is not intended that VYNPS be an ISP host reactor. As indicated in the Test Matrix in BWRVIP-86-A, RPV weld and plate surveillance materials from Susquehanna-1 have been selected from among all the existing plant surveillance programs (including the Supplemental Surveillance Program) to represent the corresponding limiting plate and weld material in the VYNPS RPV. Thus, in accordance with the ISP, no further capsules will be removed and tested from the VYNPS RPV. It is anticipated that the next Susquehanna-1 surveillance capsule should be removed from the vessel in year 2012.

Based on the test results of the removed capsules, fluence calculations will be reevaluated using a methodology approved by the NRC and demonstrated to be consistent with the methods described

in Regulatory Guide 1.190 (Ref. 7). VY used an updated fluence methodology provided by GE Nuclear Energy (GENE) (Ref. 11) and approved by NRC to develop the revised P-T curves.

As shown in Table 4-5 of BWRVIP-86-A, "Detailed Test Plan By Plant," the VYNPS RPV wall is expected to experience the lowest, end-of-life neutron fluence of all domestic BWRs.

Under the ISP, representative capsule data will be provided to each BWR vessel owner for limiting vessel weld and base materials. These data will be evaluated, as appropriate, using the methods in Regulatory Guide 1.99 (Ref. 10) in accordance with Appendix G to 10CFR50 for the determination of ART_{NDT} values. The relevant data (i.e., Charpy shift results) will be used to re-evaluate embrittlement projections for the corresponding vessel beltline materials represented by the materials in the capsule. This re-evaluation will be conducted by VY based on the results determined from testing of representative materials. If changes in P-T limits are required due to a reassessment of the limiting ART_{NDT} values, changes to the licensing basis will be requested, as appropriate.

The reporting of test results to NRC, including the data required by ASTM E-185 (Ref. 2), and the results of all fracture toughness (i.e., Charpy) tests conducted on the surveillance materials will be made by the BWRVIP program administrator.

Although there are no plans to remove additional material surveillance specimens from VYNPS, the remaining two surveillance capsules will continue to reside in the RPV in accordance with the BWRVIP ISP, in case they are needed in the future as a contingency.

Consistent with the guidance provided in RIS 2002-05 (Ref. 3), and because current TS require withdrawal of RPV specimens, VY is submitting this proposed change as a license amendment request. Current TS SR 4.6.A.5 requires that the second VYNPS surveillance capsule be removed during the refueling outage following the year in which 30 years of commercial operation is reached (i.e., the Spring 2004 refueling).

NRC has previously determined, as documented in Generic Letter 91-01 (Ref. 12) that details of RPV material surveillance programs do not need to be included in the TS, because there would be duplication of controls that have been established by regulations (i.e., Appendix H to 10CFR50). Therefore, instead of replacing the plant-specific surveillance program requirements in TS 4.6.A.5 with details regarding the ISP, VY will incorporate the ISP into the UFSAR. Because duplication of controls is unnecessary, and adequate controls already exist, it is acceptable to relocate details of the RPV surveillance program to the UFSAR.

VY is requesting a change to the VYNPS RPV material surveillance program required by 10CFR50, Appendix H, and currently implemented through TS SR 4.6.A.5, to incorporate the BWRVIP ISP into the VYNPS licensing basis. The proposed change to VY's RPV material surveillance program meets the regulatory criteria in Paragraph III. C of Appendix H to 10CFR50. Based on the foregoing considerations, including the prior acceptance of the BWRVIP ISP by the NRC staff, this proposed change is acceptable because it provides an overall improvement in the quality of data that will be obtained, analyzed and reported to NRC for the purpose of monitoring changes in the fracture toughness properties of RPV beltline materials.

3.2 P-T LIMITATIONS

3.2.1 Current Licensing Basis for P-T Curves

VYNPS License Amendment No. 203 (Ref. 13) revised the TS by changing the RPV P-T limit curves specified in TS Limiting Condition for Operation 3.6.A, "Reactor Coolant System – Pressure and Temperature Limitations," as graphically represented in Figure 3.6.1, "Hydrostatic Pressure and Leak Tests, Core Not Critical," Figure 3.6.2, "Normal Operation, Core Not Critical," and Figure 3.6.3, "Normal Operation, Core Critical." However, because VY's neutron fluence estimate used at that time to support generation of the P-T curves was not based on a methodology acceptable to the NRC staff for current licensing applications, a restriction was placed on the application of the P-T curves. That restriction disallows use of the P-T curves beyond the end of the current operating cycle (i.e., Cycle 23).

3.2.2 Updated P-T Curves

The updated P-T curves were established based on the requirements of Appendix G to 10CFR50 to assure that brittle fracture of the RPV is prevented. Attachment 2 to this Proposed Change provides the methodology of calculation used by VY in generating the revised P-T curves (i.e., TS Figures 3.6.1, 3.6.2 and 3.6.3). The revised P-T curves retain the same basic P-T limits as the current curves.

Composite P-T curves were generated for each of the pressure test, core not critical and core critical conditions at 32 EFPY. Attachment 6 includes proposed TS Figures 3.6.1, 3.6.2 and 3.6.3, which also incorporate a tabulation of P-T limits for both the bottom head and upper head regions. The revised P-T curves (and current curves) differentiate between the bottom head region and upper vessel regions. The methodology used to generate the P-T curves in this submittal is similar to the methodology used to generate the curves approved in license amendment no. 203 (Ref. 13). In this update, however, the estimate of the RPV neutron fluence was based on a new fluence methodology that follows the guidance of Regulatory Guide 1.190 (Ref. 7). Part of the analysis conducted in developing the P-T curves was to account for radiation embrittlement effects in the core region, or beltline, and ART_{NDT} values were determined using criteria of Regulatory Guide 1.99 (Ref. 10). However, although VY conducted an analysis in accordance with Regulatory Guide 1.99, the more conservative ART_{NDT} values used in the prior evaluation were retained.

For the hydrostatic pressure and leak test curve (TS Figure 3.6.1), a coolant heatup and cooldown temperature rate of 40°F/hr or less must be maintained at all times. Similarly, for the normal operation, core not critical (TS Figure 3.6.2) and the normal operation, core critical curve (TS Figure 3.6.3), the P-T curves specify a coolant heatup and cooldown temperature rate of 100°F/hr or less for which the curves are applicable.

The change to TS Figures 3.6.1, 3.6.2 and 3.6.3 to extend their applicability to 4.46×10^8 MWH(t) corresponds to an integrated plant operation of 32 EFPY. This limitation is acceptable because it is based on the re-calculated, expected neutron fluence over 40 years of operation at the current licensed power level, accounting for periods of downtime.

The enhancements made to TS Figures 3.6.1, 3.6.2 and 3.6.3 by slightly revising grid divisions, adding additional 100 psi increments to the ordinate axis, and improving curve resolution are

administrative changes of preference. They are acceptable because they do not change any technical requirement and are made to enhance user acuity.

The addition of a Note to TS Figure 3.6.2 to permit use of test instrumentation during tensioning, detensioning, and when tensioned is acceptable because test instrumentation can provide a better method of monitoring bolt-up temperatures during this phase of operations. The use of such instrumentation is limited to the condition when the vessel is vented and vessel fluid level is below the flange region. The establishment of this condition ensures that the vessel cannot be pressurized while relying on test instrumentation. Because test instrumentation is more accurate (conservatively within +/- 2°F) than permanent temperature instrumentation (+/- 10°F), a limit of ≥ 72°F may be established when using test instrumentation. A 72°F limit for test instrumentation corresponds to an 80°F limit for permanent temperature instrumentation when the respective instrumentation uncertainties are included. These values are acceptable because the analytical limit for head bolt-up is 70°F (without instrument uncertainty) as stated in current TS 3.6.A.

The changes to the tabulation in Figure 3.6.3 represent a correction of actual values used to generate the current curves. The current tabulation indicates that four different pressure limits were established corresponding to a temperature of 120°F. As can be seen from the curves, there are only two such points for 120°F. Similarly, the change in bottom head pressure at 116°F to 413 psig reflects a past administrative error in transcribing the actual value from the current curve. These changes to correct the tabulation are acceptable because they do not change actual limits (the curves are unchanged) and reflect the outputs from previous analyses.

3.2.3 Application of ASME Code Case N-640

The updated P-T limits were developed using Section XI, Appendix G of the 1995 Edition with the 1996 Addenda of the ASME Code (Ref. 8). This code edition and addenda incorporated revised stress intensity factors into the Appendix G methodology, which is used to develop the actual P-T limit curves. The revised stress intensity factors are based upon the re-orientation of the postulated defect normal to the direction of maximum stress. NRC has approved this code edition with addenda, as documented in 10CFR50.55a(b)(2).

In addition, the updated P-T limit curves are based, in part, on the application of ASME Code Case N-640 (Ref. 14). Pursuant to 10CFR50.12 and by letter dated April 16, 2001 (Ref. 15), the NRC granted an exemption to allow VY to deviate from the requirements of Appendix G to 10CFR50 in the use of this alternative method.

Code Case N-640 permits application of the lower bound static initiation fracture toughness value equation (K_{ic} equation) as the basis for establishing the P-T curves in lieu of using the lower bound crack arrest fracture toughness value equation (i.e., the K_{ia} equation), which is based on conditions needed to arrest a dynamically propagating crack—the method invoked by Appendix G to Section XI of the ASME Code. Use of the K_{ic} equation in determining the lower bound fracture toughness in the development of the P-T operating limits curve is more technically correct than the use of the K_{ia} equation because the rate of loading during a heatup or cooldown is slow and is more representative of a static condition than a dynamic condition. The K_{ic} equation appropriately implements the use of the static initiation fracture toughness behavior to evaluate the controlled heatup and cooldown process of a reactor vessel.

3.2.4 Neutron Fluence Calculations

In developing the updated P-T limit curves, the VYNPS neutron fluence calculations were also updated. These calculation updates were performed using the NRC-approved General Electric Nuclear Energy (GENE) methodology as documented in GENE's Licensing Topical Report NEDC-32983P-A (Ref. 11). The NRC-accepted (Ref. 16), proprietary methodology is fully described in NEDC-32983P-A and is not repeated herein. In general, GENE's methodology adheres to the guidance in Regulatory Guide 1.190 (Ref. 7) for neutron flux calculations and is based on a two-dimensional discrete ordinates code.

VY's estimate of neutron fluence is based in part on a dosimetry analysis of the first (and only) surveillance capsule removed from VYNPS on March 4, 1983, after 7.54 EFPY of irradiation.

The updated RPV fluence values demonstrate that the vessel fast fluence assumptions in the current P-T curve calculation remain conservative. The updated fluence analysis supports replacing the Cycle 23 expiration date with a 32 EFPY (4.46×10^8 MW-hour) expiration limit.

The revised calculations consist of two parts: First, the GENE methodology was applied to recalculate the surveillance coupon fluence rates. This task served to benchmark the new methodology. The second task involved updating the model to include a modern core design. VYNPS operating Cycle 21 was selected as representative of recent, modern core designs. Sensitivity studies of contemplated core loadings, including the current Cycle 23, indicated that peak vessel fluxes are bounded by Cycle 21. The updated fluence calculation is documented in a proprietary report prepared by GENE for VY. A summary of the VY RPV fluence analysis is presented below.

Table 1
Summary of Flux Results

Location	Flux (n/cm^2-s)
RPV Inside Surface – max location	2.96×10^8
Surveillance Capsule (30°)	1.89×10^8

Using the core design for Cycle 21, the revised, calculated peak fast flux (E >1 MeV) at end of life is summarized in Table 1.

The fast neutron fluences at the end of plant life (32 EFPY) were conservatively calculated to be 2.99×10^{17} n/cm^2 and 1.91×10^{17} n/cm^2 for the peak RPV location and the surveillance capsule, respectively. Through the end of calendar year 2002, VYNPS had accumulated approximately 23.8 EFPY of operation.

3.2.5 Regulatory Guide 1.99 and Adjusted Reference Temperature

The current and updated P-T curves are based on bounding ART_{NDT} values of 89°F at 1/4T and 73°F at 3/4T. To ensure compliance with Regulatory Guide 1.99, the new fast neutron fluence at the end of plant life, 2.99×10^{17} n/cm^2, was used to assess the adjusted RT_{NDT} of beltline

components. The shift evaluation followed Position C.1 (surveillance data not available) and the C.1(3) attenuation formula. This evaluation is documented in Attachment 2 and demonstrates that the limiting beltline component (RPV plate 1-14) remained the same, and the ART_{NDT} values calculated in accordance with Regulatory Guide 1.99 remain bounded by values used to develop the current P-T curves. As demonstrated in Attachment 2, the equivalent fluence, when compared to the updated fast fluence of 2.99×10^{17} n/cm^2, remains very conservative.

Because the capsule and end-of-life (EOL) fluence values have changed, the USE equivalent margin analysis plant applicability assessment (Ref. 17) has been incorporated into Attachment 2 to demonstrate continued compliance with ASME Code Case N-512 (Ref. 18). The prediction of change in Charpy USE was calculated in accordance with Regulatory Guide 1.99. As summarized in Attachment 2, there remains ample margin between the projected decrease in weld and plate USE and the allowable value specified in NEDO-32205 (Ref. 19). Therefore, VYNPS remains in compliance with USE requirements of 10CFR50 Appendix G by demonstrating that the projected decrease in USE per the guidance of Regulatory Guide 1.99 meets bounding limits established in the topical report.

3.2.6 Non-Beltline Regions

Non-beltline regions are defined as the vessel locations that are remote from the active fuel and where the EOL neutron fluence is not sufficient (i.e., $< 10^{17}$ n/cm^2) to cause any significant embrittlement. Non-beltline components include nozzles, closure flanges, some shell plates, the top and bottom head plates, and the control rod drive penetrations.

Detailed stress analyses of the applicable non-beltline components were performed for the purpose of fracture toughness analysis. The analyses took into account the mechanical loading and anticipated thermal transients. The thermal stresses in the vessel wall are caused by a radial thermal gradient that is created by changes in the adjacent reactor coolant during transient conditions. Transients considered include 100°F/hr startup and shutdown, reactor trip, loss of feedwater heaters or flow, loss of recirculation pump flow, and transients involving emergency core cooling injections.

3.2.7 Head Closure Flange

Stresses in the VYNPS RPV head closure flange (predominated by preload stress) establish limits incorporated into the updated P-T curves. For the flange evaluation, membrane and bending stresses were extracted from the original vessel stress report for pressure, preload and thermal expansion loadings. The critical location for head preload is the weld region between the upper head and the head flange. A minimum bolt-up temperature of 70°F was conservatively used and this requirement is maintained in TS 3.6.A.3. This conservatism is appropriate because bolt-up tensioning is one of the more limiting operating conditions (high stress and low temperature) for brittle fracture.

The conclusion of the revised neutron fluence analysis is that the revised TS P-T curves bound the recalculated coupon and RPV fast neutron fluences by a significant margin. The updated P-T curves are acceptable because they satisfy the requirements of 10CFR50.60(a), Appendix G to 10CFR50, and Appendix G to the ASME Code, as exempted by the methods of analyses in ASME Code Case N-640. In addition, the revised P-T curves provide an acceptable margin of safety against RPV brittle fracture.

3.3 Conclusion/Summary

In summary, participation in the ISP will improve the quality of compliance with the regulatory requirements in Appendices G and H to 10CFR50 while reducing cost, exposure, and outage time associated with capsule removal, shipping, and testing. The methodologies used to develop the proposed P-T limit curves satisfy the requirements of the regulations (as modified by application of ASME Code Case N-640). The revised P-T curves and outputs from the ISP (which will be used as appropriate for future adjustments to P-T limits), ensure that adequate RPV safety margins against non-ductile failure will continue to be maintained during normal operations, anticipated operational occurrences, and hydrostatic testing. Together, these measures ensure that the integrity of the reactor coolant system will be maintained for the life of the plant.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (2) such activities will be conducted in compliance with the Commission's regulations; and (3) the issuance of the requested license amendment will not be inimical to the common defense and security or to the health and safety of the public.

4.0 REFERENCES

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2. American Society for Testing and Materials (ASTM) E-185, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels" July 1982

3. NRC Regulatory Issue Summary No. 2002-05, "NRC Approval of Boiling Water Reactor Pressure Vessel Integrated Surveillance Program," April 8, 2002

4. BWRVIP letter from C. Terry to NRC Document Control Desk, "Project No. 704 – BWRVIP-86-A: BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan," November 12, 2002

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7. NRC Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001

8. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Appendix G, 1995 Edition, including Summer 1996 Addenda

9. NRC Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Pressure Vessel Materials and its Impact on Plant Operations," July 12, 1988

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11. NEDC-32983P-A, "Licensing Topical Report, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations," Rev. 1, December 2001

12. NRC Generic Letter 91-01, "Removal of the Schedule for the Withdrawal of Reactor Vessel Material Specimens from Technical Specifications," January 4, 1991

13. NRC letter from Robert M. Pulsifer to Michael A. Balduzzi (VYNPC), "Vermont Yankee Nuclear Power Station – Issuance of Amendment Re: P/T Curves (TAC No. MB0764," May 4, 2001

14. American Society of Mechanical Engineers Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves, Section XI, Division 1," February 26, 1999

15. NRC letter from R. M. Pulsifer to M.A. Balduzzi (VYNPC), "Vermont Yankee Nuclear Power Station – Exemption from the Requirements of 10 CFR Part 50, Appendix G (TAC No. MB0763)," April 16, 2001

16. NRC letter from S. A. Richards to J. F. Klapproth (GENE), "Safety Evaluation for NEDC-32983P, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation (TAC No. MA9891)," MFN 01-050, September 14, 2001

17. NRC letter from W. R. Butler to D. A. Reid (VYNPC), "Vermont Yankee Nuclear Power Corporation, Review of Equivalent Margin Analysis (TAC No. M89225)," July 20, 1994

18. American Society of Mechanical Engineers Code Case N-512, "Assessment of Reactor Vessels With Low Upper Shelf Charpy Impact Energy Levels, Section XI, Division 1," February 12, 1993

19. NEDO-32205, "10CFR50 Appendix G Equivalent Margin Analysis for Low Upper Shelf Energy in BWR/2 Through BWR/6 Vessels," Revision 1, November 1993

Attachment 2

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 258

RPV Fracture Toughness and Material Surveillance Requirements

Calculation Summary Report for Revised P-T Curves

VYC-829, Rev. 4, ATTACHMENT 1
3-18-2003

CALCULATION SUMMARY REPORT FOR REVISED P-Ṫ CURVES FOR VERMONT YANKEE NUCLEAR POWER STATION

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CALCULATION SUMMARY REPORT FOR 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 Case N-640 [2], the 1995 ASME Code, Section XI, Appendix G (including the Summer 1996 Addenda) [3], and 10CFR50 Appendix G [4].

The previous revision of this report was submitted to the NRC on February 23, 2001 in support of VY's TS proposed change 244 [Attachment 2 of Reference 19]. The NRC accepted the P-T curves submitted under proposed change 244 with the condition that for operation beyond Cycle 23, VY submit an amendment request justifying the use of the curves which satisfies the guidance of RG 1.190. [21]

In response VY has revised the vessel fluence evaluation [1]. This revised assessment follows the methodology documented in the GE Licensing Topical Report (LTR) NEDC-32983P-A approved by the U.S. NRC for licensing applications in the Safety Evaluation Report [18] and in general, GE's methodology adheres to the guidance in Regulatory Guide (RG) 1.190 for neutron flux evaluation.

The new EOL fluence value remains enveloped by the conservative RTndt shift values used here and in proposed change 244. This report has been updated to incorporate the revised fluence data and demonstrates that there is no impact to the current P-T limits.

Because the capsule and EOL fluence values have changed, the upper shelf equivalent margin analysis plant applicability assessment [17] has been incorporated into this report to demonstrate continued compliance with ASME Code Case N-512. [16].

In addition to the new fluence value, the grid line divisions on the curves have been changed to make them easier to read. More data was used to plot the curves to improve resolution. In addition, specific requirements for minimum temperature using local test instrumentation have been incorporated for flange tensioning and detensioning operations.

Prior to approval of proposed change 244, the NRC requested that VY provide basis information to support revised initial RTndt values for beltline materials, nozzle geometry data, and stress intensity values used in the development of the P-T curves. VY provided a responce to this RAI

in reference [19]. In this revision there is no change to the initial RTndt and nozzle geometry data provided in Reference [19]. The stress intensity information previously provided [19] has been again included here to facilitate NRC review.

In summary, the revision to this report is being done to incorporate four changes:

1) Incorporate the revised fluence values provided by the GE Report [1].

2) Incorporate the revised upper shelf equivalent margin analysis (EMA) plant applicability form to demonstrate continued compliance with ASME Code Case N-512 [16].

3) Provide enhancements in curve grid division and curve resolution to facilitate operator interpretation.

4) Incorporate detailed minimum temperature requirements for flange tensioning and detensioning.

All changes, except those that are non-essential or of an administrative nature, such as correction of typographical errors, editorial changes or format preferences, are marked with margin bars.

2.0 Material Properties

An assessment of the fracture toughness properties of all material used in the VY reactor vessel plate, weld and forgings is provided in Attachment 2 to VYC-829 R4. 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} values 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 conducted in supporting VY calculations (Attachment 2 of VYC-829 R4) establish the initial RT_{NDT} values for all other materials.

Table 2-1 and Table 2-2 show an evaluation of the expected irradiation shift for the beltline plates. The peak end of license (EOL) fast fluence of 2.99×10^{17} n/cm^2 (E>1.0 MeV) used in Table 2-1 is from the Reference 1 GE report. The methodology used by GE to develop this fluence value is documented in GE's Licensing Topical Report (LTR) NEDC-32983P-A [1], which was approved by the U.S. NRC for licensing applications in the Safety Evaluation Report "Safety Evaluation for NEDC-32983P, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation (TAC No. MA9891)," MFN 01-050, September 14, 2001.

For purposes of determining the P-T curves for the vessel core region materials, VY has elected to maintain the more conservatively shifted ART_{NDT} values previously used by VY: 89°F at the 1/4T point and 73°F at the 3/4T point. Based on guidance of Reg Guide 1.99 Rev. 2 lower values of ART_{NDT} could have been used. The NRC highlighted this in their Reference 11 safety evaluation.

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 2-2, and Regulatory Guide 1.99, Rev. 2 [12] criteria for calculating ART_{NDT}, the use of the conservative ART_{NDT} values equates to a minimum end-of-life surface fluence of 1.24×10^{18} n/cm^2 for the four core region plates. This is well beyond the peak end-of-life surface fluence, 2.99×10^{17} n/cm^{2}calculated for Vermont Yankee by GE [1]. 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.

Reference 1 also provides the axial distribution of 32-EFPY fast neutron fluence at the peak azimuth of the RPV inside surface. The results of the analysis demonstrate the fast fluence outside the active axial fuel zone at the RPV wall is less than 1×10^{17} n/cm^2. The N4 feedwater nozzles are well above the top of active fuel and the N2 recirculation nozzles are below the

bottom of active fuel. Therefore the fluence in these locations is substantially below $1x10^{17}$ n/cm^2.

Based on the revised fluence projection [1], per Reg Guide 1.99 [12] requirements, we have revised the projected decrease in upper shelf energy (USE) data and reevaluated the decrease against criteria from NEDO-32205 [17], the equivalent margin topical report applicable to VY. This topical report follows the methods provided in Code Case N-512 [18] and was accepted by the NRC [19].

As summarized in Table 15, there remains ample margin between the projected decrease in weld and plate upper shelf energy and the allowable decrease recommended in topical report NEDO-32205. Therefore VY remains in compliance with USE requirements of 10CFR50 Appendix G by demonstrating that the projected decrease in USE per the guidance of Regulatory Guide 1.99 meets bounding limits established in the topical report.

3.0 P-T Curve Methodology

The P-T curve methodology is based on the requirements of References [2] through [4]. There are five regions of the reactor pressure vessel (RPV) that were evaluated in this calculation: (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 conservatively bounded. 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 $\sqrt{t}$ inch)

c. Calculate the thermal stress intensity factor, K_{IT}. This is calculated based on ASME Section XI, Appendix G [3] 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 [13] 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$\sqrt{}$ inch)
SF = (Code specified) 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 ASME Appendix G [3] 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. This limit is applicable when the flange is tensioned or in the process of being tensioned or detensioned.

- 10CFR 50 Appendix G requires that temperature be maintained at or above the RTndt of the closure flange.

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 to the 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.

During vessel tensioning and detensioning the permanent flange temperature instrumentation is removed and special test instrumentation is applied to monitor flange temperature. During this procedure, the vessel is vented to atmosphere and the vessel fluid level is below the flange region. During this operation the external temperature is equal or lower than the internal temperature, therefore the external test instrumentation can be used as a more accurate and conservative assessment of flange temperature conditions. The test instrumentation is selected to have less than +/- 2°F uncertainty.

3.4 Beltline Evaluation

For the beltline evaluation, the equations in ASME Section XI, Appendix G [3] 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 the weld region between the upper head and the head

flange [13]. Stress intensity factors were calculated based on the equations similar to ASME Section XI, Appendix G for membrane and bending stresses except that actual stresses were substituted for the pressure stresses in ASME Section XI. 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 G 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).

The tabulated stress intensity summary for the flange under hydrostatic pressure and leak tests has been updated in this summary report. Table 9 submitted with PC change 244 conservatively applied a 2.0 safety factor to the preload stress intensity for the Pressure Test condition. Table 9 has been updated to include the 1.5 safety factor per ASME XI. This change was done to better highlight the margin between ASME XI Appendix G temperature limits and the GE recommended minimum temperature requirement. The revised stress intensity information is included in the stress intensity summary included in Table 16-1. This change has no impact on the limiting P-T curve.

At low pressure all vessel components, except those components in the flange region, have little stress and are not at risk to brittle failure. The stress of flange region components is predominantly due to preload. With preload removed (unbolted condition) and the vessel depressurized the ASME XI Appendix G minimum temperature requirement for all vessel components are well below 0°F. In Table 17 the ASME XI P-T limits for the flange region without preload are given using the highest thermal and pressure stress intensity from the controlling flange locations. At 0°F the allowable pressure is 637 psig.

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 the VY calculation (VYC-1005) that supported VY's NUREG-0619 feedwater nozzle inspection interval evaluation. In VYC-1005 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 12 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 penetrations showed that the membrane stresses in the bottom head would 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.

Sensitivity 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 from 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 Appendix G to 10CFR50 margins discussed in Section 3.2 are shown in Figures 3 through 5.

During vessel tensioning and detensioning the permanent flange temperature instrumentation is removed and special test instrumentation is applied to monitor flange temperature. When monitoring external flange temperature with local test instrumentation during tensioning and detensioning the temperature should be at least:

+ 10°F (RT_{NDT} of the of the limiting flange material)
+ 60°F (GE Margin)
+ 2°F (Maximum Test Instrument Uncertainty)
= 72°F

Therefore when monitoring external flange temperature with local test instrumentation during tensioning and detensioning the flange region temperatures must be maintained greater than or equal to 72 °F. A note has been added to the P-T curve in Figure 4 to specify this requirement.

With the vessel depressurized and the flange detensioned the minimum vessel temperature per 10CFR50 Appendix G is 20°F (RT_{NDT} of the limiting flange material, +10°F, plus instrument uncertainty of permanently installed process instrumentation, 10°F).

5.0 References

1. GE-NE-0000-0007-2342-R0, DRF 0000-0007-2342, Revision 0, Class 3, January 2003" Final Report Entergy Northeast Vermont Yankee Neutron Flux Evaluation".

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.

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.

16. ASME Boiler and Pressure Vessel Code, Code Case N-512, Assessment of Reactor Vessels With Low Upper Shelf Charpy Impact Energy Levels, Section XI, Division 1, 02-12-93.

17. NEDO-32205 Class I, November 1993, Revision 1, "10CFR50 Appendix G Equivalent Margin Analysis for Low Upper Shelf Energy in BWR/2 Through BWR/6 Vessels".

18. Letter from U.S.NRC to Chairman of BWR Owner's Group, "Acceptance for Referencing Topical report NEDO-32205, revision 1, "10CFR50 Appendix G Equivalent Margin Analysis for Low Upper Shelf Energy in BWR/2 Through BWR/6 Vessels".

19. Letter from Vermont Yankee Nuclear Power Corporation BVY 01-14, to U.S. NRC, "Technical Specification proposed Change No. 244, Response to Request for Additional Information," 2/23/2001.

20. Letter from Nuclear Regulatory Commission, NVY 01-046 to Vermont Yankee Nuclear Power Corporation, "Issuance of Amendment RE: P/T Limit Curves (TAC No. MB0764) (Tac No. MB0764).

21. Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," U.S. NRC, March 2001.

22. VY Document TE-2002-050, "Updated RPV Fluence Calculations Evaluation in Support of P/T Curves."

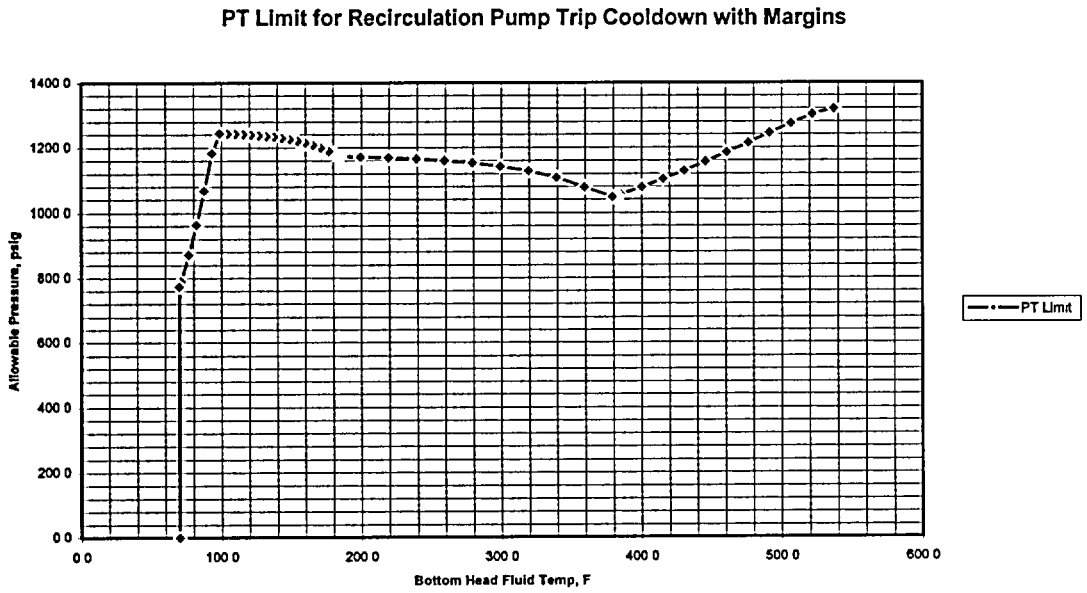


Figure 1: Bottom Head Recirculation Pump Trip Pressure/Temperature Limit Curve

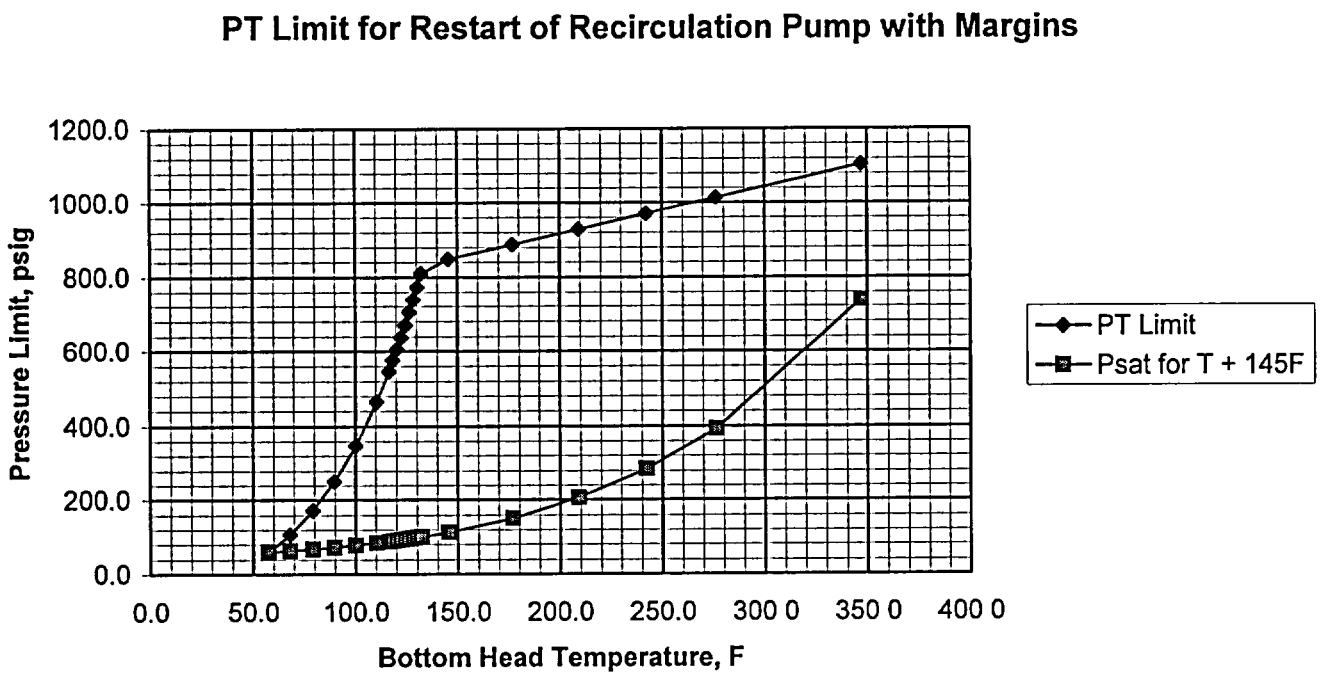
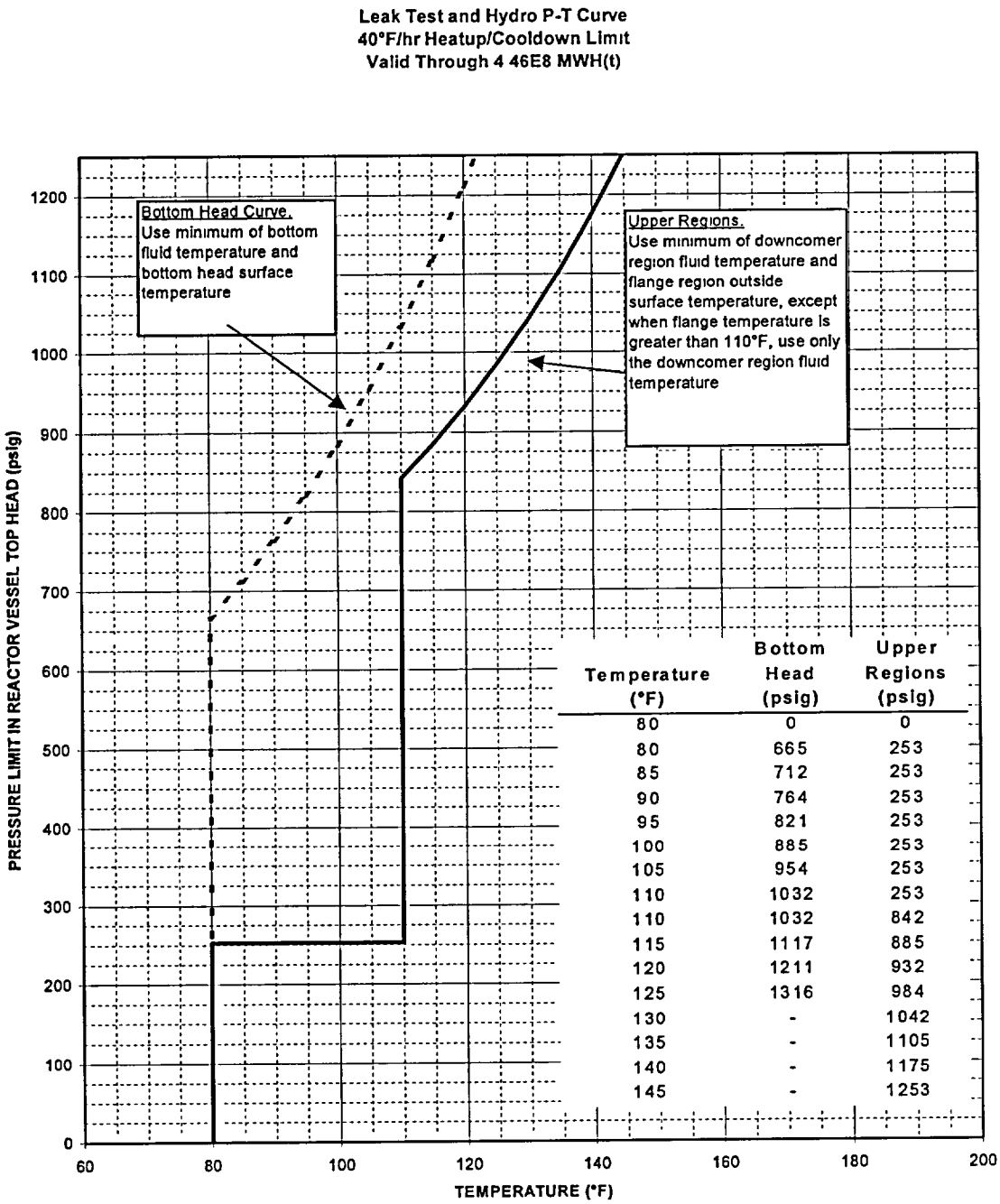


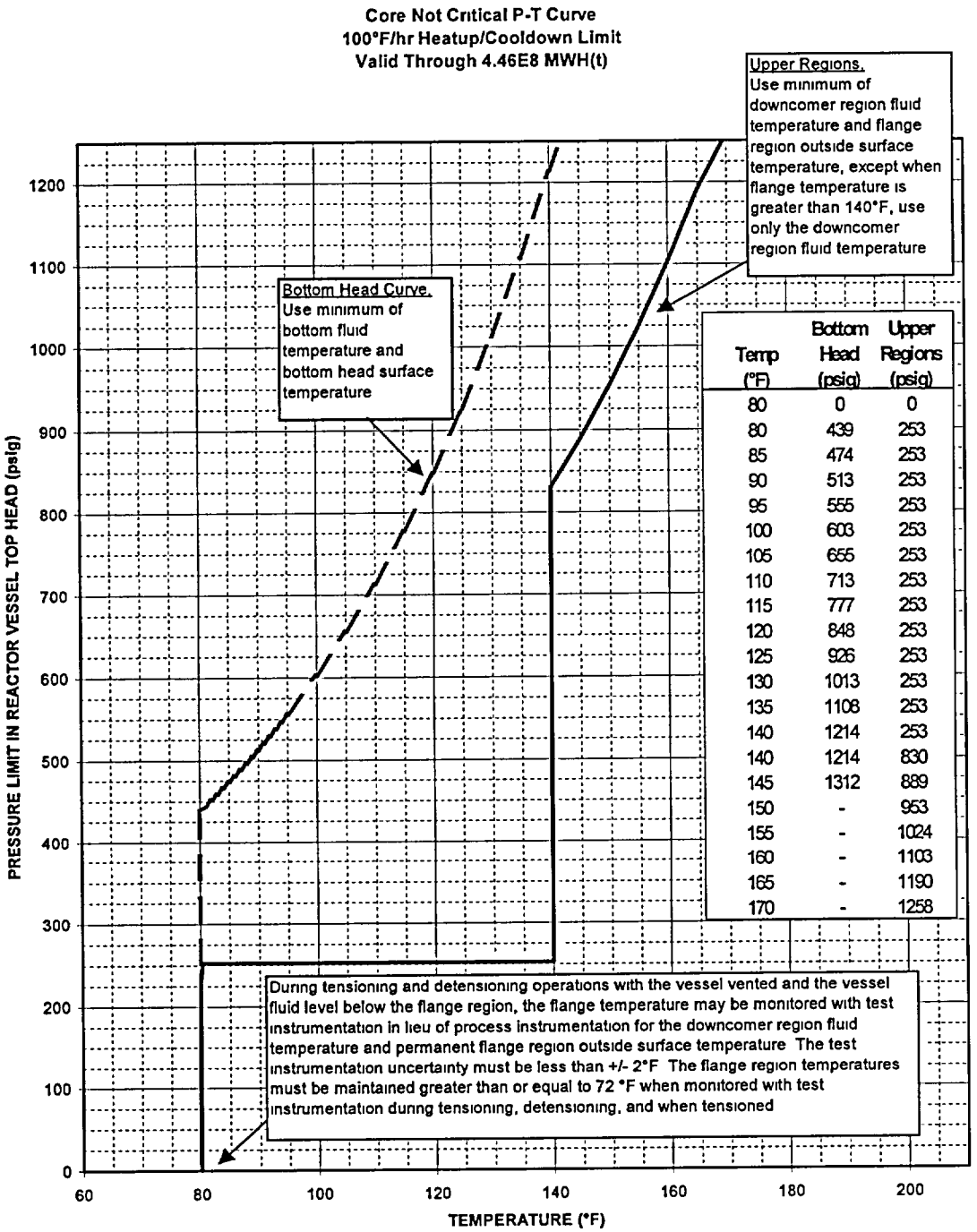
Figure 2: Pressure/Temperature Limit Curve for Recirculation Pump Start



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)

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



Temp (°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 5: Core Critical P-T Curve (Curve C)

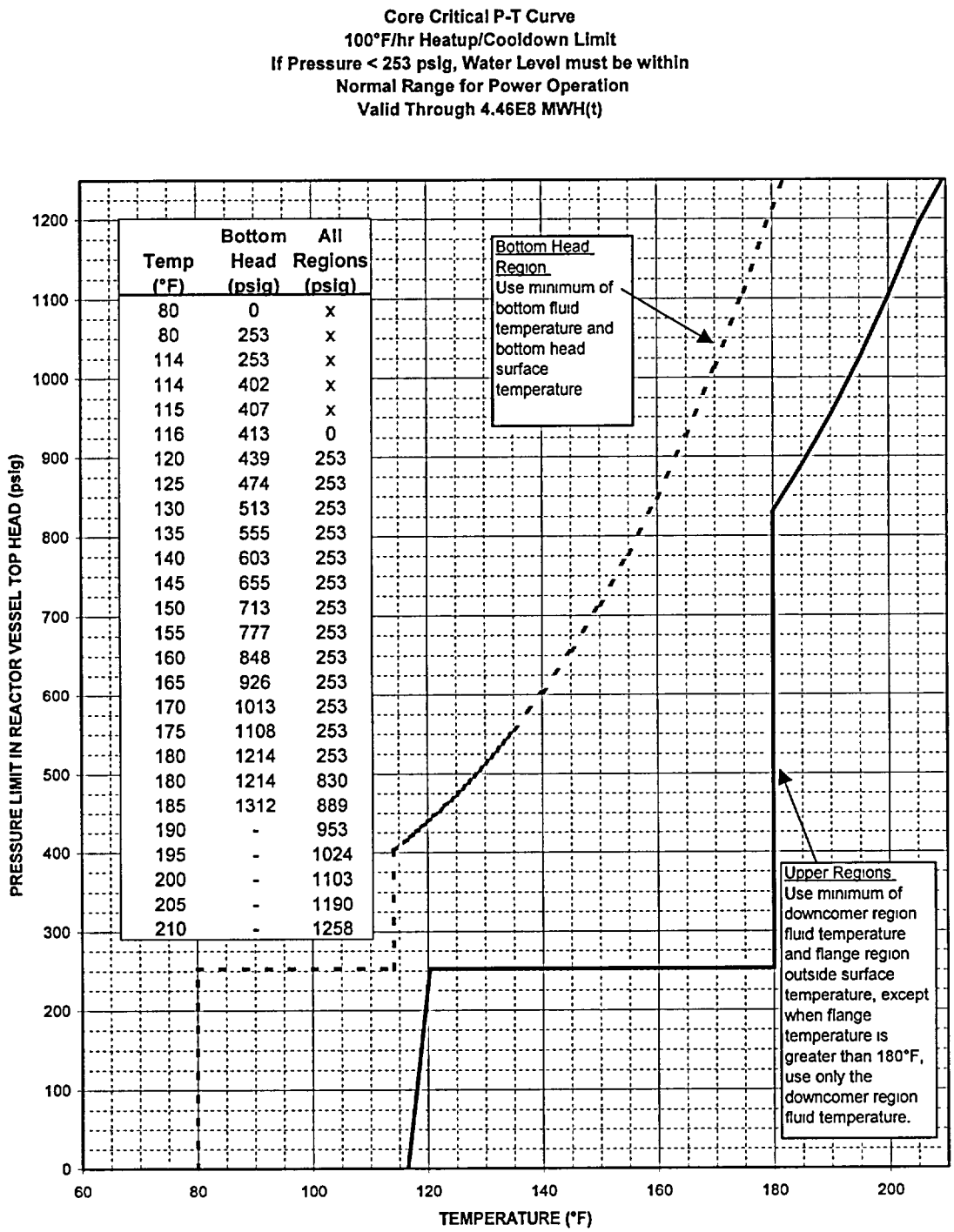


Table 1: Initial RT_{NDT} for Materials in Vermont Yankee Reactor Vessel

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	-70

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 2-1: Calculation of Peak Fluence Values

Calculation of Effective Peak Fluence Values		
	Units	
EFPY	years	32
Seconds per Year =3600*365*24	sec per year	31536000
Flux at Inside Surface [GE reference 1]	n/cm^2/s	2.96E+08
Flux at 1/4 from inside Surface [GE reference 1]	n/cm^2/s	2.05E+08
Flux at 3/4 from inside Surface [GE reference 1]	n/cm^2/s	8.56E+07
Fluence at Inside Surface using GE flux = flux*EFPY*sec/yr	n/cm^2	2.99E+17
Fluence at 1/4 thickness using GE flux = flux*EFPY*sec/yr	n/cm^2	**2.07E+17**
Fluence at 3/4 thickness using GE flux = flux*EFPY*sec/yr	n/cm^2	**8.64E+16**
Vessel Thickness	inches	5.06
Fluence at 1/4 thickness by RG1.99 =GE ID Fluence *EXP(-0.24*t/4)	n/cm^2**	**2.20E+17**
Fluence at 3/4 thickness by RG1.99 =GE ID Fluence *EXP(-0.24*3*t/4)	n/cm^2**	**1.20E+17**
**The RG1.99 C.1(3) attenuation formula results in conservative Fluence Values at the 1/4t and 3/4t locations when compared to values calculated from GE flux values provided in Reference 1. Conservatively these higher values are used in the Ref Guide 1.99 Section C.1 shift evaluation below.		

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

Shift in accordance with 1.99 Rev. 2						
Plate		I-14	I-15	I-16	I-17	Weld
Initial RTNDT	°F	30	-10	0	0	-70
Cu w/%		0.11	0.14	0.13	0.12	0.04
Ni w/%		0.63	0.66	0.59	0.61	1
Chemistry Factor, CF		74	102	91	83	54
delta RTNDT @ 1/4 T Based on Higher RG1.99 fluence.	°F	13.5	18.6	16.6	15.2	9.9
delta RTNDT @ 3/4 TBased on Higher RG1.99 fluence.	°F	9.2	12.6	11.3	10.3	6.7
Sig-I, Standard Deviation of Initial RTNDT		0.0	0.0	0.0	0.0	0.0
Margin@ 1/4T=2*sqrt(Sig-I^2+Sig-delta^2)	°F	13.5	18.6	16.6	15.2	9.9
Sig-delta, Standard Deviation of delta RTNDT @ 1/4T	°F	6.8	9.3	8.3	7.6	4.9
Margin@ 3/4T=2*sqrt(Sig-I^2+Sig-delta^2)	°F	9.2	12.6	11.3	10.3	6.7
Sig-delta, Standard Deviation of delta RTNDT @ 3/4T	°F	4.6	6.3	5.6	5.1	3.3
Adjusted RTNDT @ 1/4T	°F	57.0	27.3	33.2	30.3	-50.3
Adjusted RTNDT @ 3/4T	°F	48	15	23	21	-57
NOTE: Sig-delta lesser value of 17°F for base metals and 28°F for welds or 1/2 delta RTNDT						

Table 3: Calculation of Equivalent Peak Beltline Fluence Values

Find Reg Guide 1.99 equivalent fluence				
Calculation of Effective Peak Beltline Fluence Value	Units	that matches ARTNDT used by VY		
Plate		1-14	1-15	1-16
Equivalent Factor on Fluence, k*2.99x10^17	k	4.13	11.15	8.85
Shift in accordance with 1.99 Rev. 2		32 EFPY	32 EFPY	32 EFPY
Effective Inside Surface Fluence Value=k*2.99x10^17	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 RTNDT	°F	30	-10	0
Chemistry Factor, CF		74	102	91
delta RTNDT @ 1/4 T	°F	29.5	63.3	51.3
delta RTNDT @ 3/4 T	°F	21.6	48.8	39.1
Sig-I, Standard Deviation of Initial RTNDT		0.0	0.0	0.0
Margin@ 1/4T=2*sqrt(Sig-I^2+Sig-delta^2)	°F	29.5	34.0	34.0
Sig-delta, Standard Deviation of delta RTNDT @ 1/4T	°F	14.7	17.0	17.0
Margin@ 3/4T=2*sqrt(Sig-I^2+Sig-delta^2)	°F	21.6	34.0	34.0
Sig-delta, Standard Deviation of delta RTNDT @ 3/4T	°F	10.8	17.0	17.0
Adjusted RTNDT @ 1/4T	°F	89.0	87.3	85.3
Adjusted RTNDT @ 3/4T	°F	73	73	73
NOTE: Sig-delta lesser value of 17°F or 1/2 delta RTNDT				

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 √t = 2 249 √inch
Vessel Radius, R =	**103.1875**	inches
ART_{NDT} =	**73.0**	°F
Heatup Rate, HU =	**40**	°F/hr
K_{IT} =	**1.73**	ksi*inch$^{1/2}$ (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**	(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 √t = 2 249 √inch
Vessel Radius, R =	**103.1875**	inches
ART_{NDT} =	**89.0**	°F
Cooldown Rate, CR =	**40**	°F/hr
K_{IT} =	**2.20**	ksi*inch$^{1/2}$ (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**	(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 √t = 2 249 √inch
Vessel Radius, R =	**103.1875**	inches
ART_{NDT} =	**73.0**	°F
Heatup Rate, HU =	**100**	°F/hr
K_{IT} =	**4.34**	ksi*inch1/2 (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**	(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 √t = 2 249 √inch
Vessel Radius, R =	**103.1875**	inches
ART_{NDT} =	**89.0**	°F
Cooldown Rate, CR =	**100**	°F/hr
K_{IT} =	**5.49**	ksi*inch1/2 (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**	(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_{IT} + 1.5 x K_{IPL}	70.62	ksi*inch$^{1/2}$ (Note Factor of 1 5 is Safety Factor)		
Safety Factor =	1.50	(for hydrotest)		K, ksi*inch$^{1/2}$
K_{1P} for 1000 psig =	10.30	ksi*inch$^{1/2}$	K_{IPL}=1 0*Preload =	45.7
Temperature Adjustment =	10.0	°F	K_{IT}=Thermal =	2.072
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)
0	0.0	50 18	-13 63	-1323	10	-1358
5	5 0	51 96	-12 44	-1208	15	-1243
10	10.0	53 93	-11 13	-1080	20	-1115
15	15 0	56 11	-9 67	-939	25	-974
20	20 0	58 52	-8 06	-783	30	-818
25	25 0	61.19	-6 29	-611	35	-646
30	30 0	64.13	-4.33	-420	40	-455
35	35 0	67.38	-2.16	-210	45	-245
40	40 0	70 98	0 24	23	50	-12
45	45 0	74.95	2 89	280	55	245
50	50 0	79.34	5 81	565	60	530
55	55 0	84 20	9 05	879	65	844
60	60 0	89 56	12 63	1226	70	1191
65	65 0	95 49	16 58	1609	75	1574
67	66.9	97.93	18 20	1767	77	1732
70	70.0	102 04	20.94	2033	80	1998
75	75 0	109 28	25.77	2502	85	2467
80	80 0	117 28	31.11	3020	90	2985

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
$K_{IT} + 2 \times K_{IPL}$	96.58	ksi*inch$^{1/2}$ (Note Factor of 2 is Safety Factor)	
Safety Factor =	2.00	(for level A/B)	K, ksi*inch$^{1/2}$
K_{1P} for 1000 psig =	10.30	ksi*inch$^{1/2}$	K_{IPL}=1.0*Preload = 45.7
Temperature Adjustment =	10.0	°F	K_{IT}=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)
-15	-15 0	45 78	-25 40	-2466	-5	-2501
-10	-10 0	47 10	-24.74	-2402	0	-2437
-5	-5 0	48 56	-24.01	-2331	5	-2366
0	0 0	50 18	-23.20	-2253	10	-2288
5	5 0	51.96	-22.31	-2166	15	-2201
10	10 0	53 93	-21.32	-2070	20	-2105
15	15 0	56 11	-20 23	-1964	25	-1999
20	20 0	58 52	-19 03	-1847	30	-1882
25	25 0	61.19	-17.70	-1718	35	-1753
30	30 0	64.13	-16 22	-1575	40	-1610
35	35 0	67.38	-14 60	-1417	45	-1452
40	40 0	70.98	-12 80	-1243	50	-1278
45	45 0	74.95	-10 81	-1050	55	-1085
50	50 0	79 34	-8 62	-837	60	-872
55	55.0	84 20	-6 19	-601	65	-636
60	60.0	89 56	-3.51	-341	70	-376
65	65 0	95 49	-0 55	-53	75	-88
66	66 0	96 75	0.08	8	76	-27
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

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 ======>
K_{1T} for 552F - 50F Step =	106.56	ksi*inch$^{1/2}$
Safety Factor =	2.00	(for level A/B)
K_{1P} for 1025 psig =	33.80	ksi*inch$^{1/2}$
Temperature Adjustment =	10.0	°F
Pressure Adjustment =	45.0	psig (hydrostatic pressure + uncertainty)

All EFPYs

Temp. Change 502 °F Step

Fluid Temperature T (°F)	1/8t Temperature (°F)	K_{IC} (ksi*inch$^{1/2}$)	K_{IT} (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	0 00	29 26	887	60	842
55	52 5	59 82	1.06	29 38	891	65	846
60	55 0	61.19	2 12	29 53	896	70	851
65	57.5	62 62	3 18	29 72	901	75	856
70	60 0	64 13	4 25	29 94	908	80	863
75	62 5	65 72	5 31	30 21	916	85	871
80	65 0	67.38	6 37	30 51	925	90	880
85	67.5	69.14	7.43	30 85	936	95	891
90	70 0	70.98	8 49	31 24	948	100	903
95	72.5	72.92	9 55	31 68	961	105	916
100	75 0	74 95	10.61	32 17	976	110	931
105	77.5	77.09	11.67	32.71	992	115	947
110	80 0	79.34	12 74	33.30	1010	120	965
115	82.5	81 71	13 80	33 96	1030	125	985
120	85 0	84.20	14 86	34 67	1051	130	1006
125	87.5	86 81	15 92	35 45	1075	135	1030
130	90 0	89.56	16 98	36 29	1100	140	1055
135	92 5	92 45	18 04	37 20	1128	145	1083
140	95 0	95 49	19.10	38 19	1158	150	1113
145	97.5	98 68	20 17	39 26	1191	155	1146
150	100.0	102.04	21.23	40 41	1225	160	1180
155	102.5	105.57	22.29	41.64	1263	165	1218
160	105 0	109 28	23.35	42 96	1303	170	1258

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

Pressure-Temperature Curve Calculation

(Core Not Critical - N2 Recirc Nozz - 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_{1P} 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)
0	0.0	39 44	7.19	166	10	111
5	5 0	40.10	7.52	174	15	119
10	10 0	40 83	7.88	183	20	128
15	15.0	41.63	8 28	192	25	137
20	20 0	42.52	8 72	202	30	147
25	25 0	43 50	9 21	213	35	158
30	30 0	44 58	9 75	226	40	171
35	35.0	45 78	10 35	240	45	185
40	40.0	47.10	11.01	255	50	200
45	45 0	48.56	11 75	272	55	217
50	50 0	50.18	12.55	291	60	236
55	55 0	51.96	13 45	311	65	256
60	60 0	53.93	14 43	334	70	279
65	65.0	56 11	15 52	360	75	305
66	66 4	56 78	15 86	367	76	312
70	70 0	58 52	16 73	387	80	332
70	70 3	58 70	16 81	389	80	334
75	75.0	61.19	18 06	418	85	363
80	80.0	64.13	19.53	452	90	397
85	85.0	67.38	21.16	490	95	435
90	90.0	70 98	22.95	532	100	477
95	95 0	74 95	24.94	578	105	523
100	100.0	79.34	27.14	629	110	574
105	105 0	84 20	29 56	685	115	630
110	110.0	89.56	32 25	747	120	692
115	115 0	95 49	35 21	816	125	761
120	120 0	102.04	38 48	891	130	836
125	125 0	109.28	42.10	975	135	920
130	130 0	117.28	46 11	1068	140	1013

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$^{1/2}$ (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**	(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 √t = 2.437 √inch
Vessel Radius, R =	**103.1875**	inches
ART_{NDT} =	**30.0**	°F
Cooldown Rate, CR =	**100**	°F/hr
K_{IT} =	**10.49**	ksi*inch1/2 (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 =	**2.00**	(for level A/B)
Factor =	**1.2808**	M_m concentration factor
M_m =	**2.256**	(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

Table 15

Equivalent Margin Upper Shelf Energy Summary					
NEDO-32205 App B Worksheet Surveillance Info	Cu % (Ref. 9)	Capsule Fluence n/cm^2 (Ref. 1, 22)	Measured Decrease % (Ref. Charpy curves)	RG1.99 Predicted Decrease %	Ratio of Measured to Predicted F1, Factor
Surveillance Plate USE	0.11%	4.50E+16	8 0%	5 5%	1.447
Surveillance Weld USE	0.03%	4.50E+16	4.80%	4.78%	1.005
NEDO-32205 App B Worksheet Beltline Info	Cu % (Table 2-2)	EOL 1/4*T Fluence n/cm^2 (Table 2-1)	RG1.99 Predicted Decrease %	Adjusted Decrease= Pred * F1 %	NEDO-32205 Limit %
Limiting Plate USE	0 14%	2.20E+17	9.4%	13.5%	21%
Limiting Weld USE	0 04%	2 20E+17	7.3%	7.4%	34%

Table 16-1

Stress Intensity Value Summary

Pressure Test Condition				
RPV Component	Load Condition	Location	Temperature (deg F)	K_{IT} (ksi*sqrt*(inch))
Bottom Head CD	40 F/HR CD	1/4T	note 1	4.19
Bottom Head HU	40 F/HR HU	3/4 T	note 2	3.31
FW Blend HU-CD	Injection Transient	1/8 T	(Tfluid + 50F)/2	*see Table 16-2*
FWBore HU-CD	Injection Transient	1/8 T	(Tfluid + 50F)/2	*see Table 16-3*
N2 Recirc Nozzle CI	40 F/HR CD	1/4T	note 1	10.03
RPV Component	Load Condition	Location	Temperature (deg F)	$K_{IT} + 1.5 \times K_{IPL}$ *(ksi*sqrt*(inch))*
Upper Flange 1 CD	40 F/HR CD plus Bolt Preload	3/4T	note 1	*50.25*
Upper Flange 1 HU	40 F/HR HU plus Bolt Preload	3/4T	note 2	*50.91*
Upper Flange 2 CD	40 F/HR CD plus Bolt Preload	3/4T	note 1	*51.56*
Upper Flange 2 HU	40 F/HR HU plus Bolt Preload	3/4T	note 2	*70.62*
Normal Operation Condition				
RPV Component	Load Condition	Location	Temperature (deg F)	K_{IT} (ksi*sqrt*(inch))
Bottom Head CD	100 F/HR CD	1/4T	note 1	10.49
Bottom Head HU	100 F/HR HU	3/4 T	note 2	8.28
FW Blend HU-CD	Injection Transient	1/8 T	(Tfluid + 50)/2	*see Table 16-2*
FWBore HU-CD	Injection Transient	1/8 T	(Tfluid + 50)/2	*see Table 16-3*
N2 Recirc Nozzle CI	100 F/HR CD	1/4T	note 1	25.07
RPV Component	Load Condition	Location	Temperature (deg F)	$K_{IT} + 2 \times K_{IPL}$ *(ksi*sqrt*(inch))*
Upper Flange 1 CD	100 F/HR CD plus Bolt Preload	3/4T	note 1	*67.91*
Upper Flange 1 HU	100 F/HR HU plus Bolt Preload	3/4T	note 2	*67.88*
Upper Flange 2 CD	100 F/HR CD plus Bolt Preload	3/4T	note 1	*69.51*
Upper Flange 2 HU	100 F/HR HU plus Bolt Preload	3/4T	note 2	*96.58*
Note 1	For cooldown transients, temperature lag of metal verses fluid conservatively ignored.			
Note 2	For these components both inside fluid temperature and outside skin temperature are monitored. The minimum temperature is used for monitoring PT limits. Therefore HU lag does not need to be used.			

Table 16-2

Stress Intensity Value Feedwater Nozzle Blend

Temperature and K_{IT} Values

(FW Injection (Blend) - Corner Nozzle Crack)

Inputs:

Plant =	Yankee	
Component =	FW Nozzle Blend	
ART_{NDT} =	40.0	
Anlaysis Basis	502	°F Step
K_{IT} for 552F - 50F Step=	106.56	ksi*inch$^{1/2}$
K_{IP} for 1025 psig =	33.80	ksi*inch$^{1/2}$

Fluid Temperature T (°F)	1/8t Temperature (°F)	K_{IC} (ksi*inch$^{1/2}$)	Kit (ksi*inch$^{1/2}$)
50	50.0	58.52	0 00
55	52.5	59.82	1.06
60	55.0	61.19	2.12
65	57.5	62.62	3.18
70	60.0	64.13	4 25
75	62.5	65.72	5.31
80	65.0	67.38	6 37
85	67.5	69.14	7.43
90	70.0	70.98	8.49
95	72.5	72 92	9.55
100	75 0	74.95	10.61
105	77.5	77.09	11.67
110	80.0	79.34	12.74
115	82.5	81.71	13 80
120	85.0	84 20	14 86
125	87.5	86 81	15 92
130	90.0	89.56	16 98
135	92.5	92.45	18.04
140	95.0	95.49	19.10
145	97.5	98.68	20.17
150	100.0	102.04	21.23
155	102.5	105.57	22.29
160	105 0	109.28	23.35

Table 16-3

Stress Intensity Value Feedwater Nozzle Bore

Temperature and K_{IT} Values

(FW Injection (Bore)- Corner Nozzle Crack)

Inputs:

Plant =	Yankee	
Component =	FW Nozzle Bore	
ART_{NDT} =	40.0	°F ======>
Analysis Basis	502	°F Step
K_{IT} for 552F - 50F Step=	133.39	ksi*inch$^{1/2}$
K_{IP} for 1025 psig =	28.36	ksi*inch$^{1/2}$

Fluid Temperature T (°F)	1/8t Temperature (°F)	K_{IC} (ksi*inch$^{1/2}$)	Kit (ksi*inch$^{1/2}$)
50	50.0	58 52	0.00
55	52.5	59.82	1.33
60	55.0	61.19	2 66
65	57.5	62.62	3 99
70	60.0	64 13	5.31
75	62.5	65.72	6 64
80	65 0	67.38	7.97
85	67.5	69 14	9.30
90	70.0	70 98	10 63
95	72.5	72.92	11.96
100	75.0	74.95	13.29
105	77.5	77.09	14.61
110	80.0	79.34	15.94
115	82.5	81.71	17.27
120	85 0	84.20	18 60
125	87.5	86 81	19.93
130	90.0	89 56	21.26
135	92 5	92 45	22.59
140	95.0	95.49	23 91
145	97.5	98.68	25.24
150	100.0	102.04	26 57
155	102.5	105.57	27.90
160	105.0	109 28	29.23

Table 17

Bounding Flange Case with No Preload

Pressure-Temperature Curve Calculation

(Core Not Cntical - Bounding Flange Case no Preload)

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
$K_{IT} + 2 \times K_{IPL}$	5.18	ksi*inch$^{1/2}$ (Note Factor of 2 is Safety Factor)
Safety Factor =	2.00	(for level A/B)
K_{IP} for 1000 psig =	31.21	ksi*inch$^{1/2}$
Temperature Adjustment =	10.0	°F
Pressure Adjustment =	35.0	psig (hydrostatic pressure + uncertainty)

	K, ksi*inch$^{1/2}$
K_{IPL}=0 0*Preload =	0
K_{IT}=Thermal =	5.18

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)
-15	-15.0	45 78	20.30	650	-5	615
-10	-10 0	47 10	20.96	672	0	637
-5	-5 0	48 56	21.69	695	5	660
0	0 0	50 18	22 50	721	10	686
5	5 0	51.96	23 39	749	15	714
10	10 0	53 93	24 38	781	20	746
15	15 0	56 11	25 47	816	25	781
20	20.0	58.52	26 67	855	30	820
25	25 0	61.19	28 00	897	35	862
30	30.0	64 13	29 48	944	40	909
35	35 0	67.38	31.10	997	45	962
40	40 0	70 98	32.90	1054	50	1019
45	45 0	74.95	34 89	1118	55	1083
50	50 0	79.34	37.08	1188	60	1153
55	55 0	84.20	39 51	1266	65	1231
60	60 0	89.56	42 19	1352	70	1317
65	65 0	95.49	45 15	1447	75	1412
66	66 0	96.75	45 78	1467	76	1432
67	67.0	98.03	46 43	1488	77	1453
68	68.0	99 34	47.08	1508	78	1473
69	69 0	100.68	47.75	1530	79	1495
70	70.0	102 04	48 43	1552	80	1517

Attachment 3

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 258

RPV Fracture Toughness and Material Surveillance Requirements

Determination of No Significant Hazards Consideration

Description of amendment request:

The Proposed Change revises the reactor pressure vessel material surveillance program as currently specified in Technical Specifications Surveillance Requirement 4.6.A.1 and the reactor coolant system Pressure-Temperature limit curves (Technical Specifications Figures 3.6.1, 3.6.2 and 3.6.3). In addition, conforming changes are also being made to the associated Technical Specification Bases and the Updated Final Safety Analysis Report. The Proposed Change incorporates contemporary methodologies and industry programs for establishing material surveillance and fracture toughness requirements that have been previously found to be acceptable to the NRC staff. The two primary components to the Proposed Change are described in the accompanying safety assessment and meet the following regulatory bases:

First, Vermont Yankee (VY) is proposing to revise the licensing basis for the Vermont Yankee Nuclear Power Station by replacing the plant-specific reactor pressure vessel (RPV) material surveillance program with the Boiling Water Reactor Vessel Internals Project (BWRVIP) Integrated Surveillance Program (ISP), which has been approved by the NRC staff as meeting the requirements of paragraph III.C of Appendix H to 10 CFR 50 for an integrated surveillance program.

Second, VY is proposing to revise the P-T limit curves for the reactor coolant system in accordance with the requirements of Appendix G to 10CFR50 and an NRC-granted allowance to use the American Society of Mechanical Engineers Boiler and Pressure Vessel Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves Section XI, Division 1."

There are no plant modifications associated with these changes.

Basis for No Significant Hazards Determination:

Pursuant to 10CFR50.92, Vermont Yankee 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). These criteria require that the operation of the facility in accordance with the proposed amendment will not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. The discussion below addresses each of these criteria and demonstrates that the proposed amendment does not constitute a significant hazard.

The proposed change does not involve a significant hazards consideration because the changes would not:

1) <u>Involve a significant increase in the probability or consequences of an accident previously evaluated</u>.

   The proposed change implements an integrated surveillance program that has been previously evaluated and accepted by the NRC staff as meeting the requirements of paragraph III.C of Appendix H to 10CFR50. In addition, the proposed change revises P-T limits in accordance with Appendix G to 10CFR50 (as modified by use of an accepted ASME Code Case). Brittle fracture of the reactor pressure vessel is not a postulated or evaluated design basis accident. No evaluations of other postulated accidents are affected by this proposed change. Because the

applicable regulatory requirements continue to be met, the change does not significantly increase the probability of any accident previously evaluated. The proposed change provides the same assurance of RPV integrity as previously provided.

The change will require that the reactor pressure vessel and interfacing coolant system continue to be operated within their design, operational or testing limits. Also, the change will not alter any assumptions previously made in evaluating the radiological consequences of accidents.

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

2) Create the possibility for a new or different kind of accident from any previously evaluated.

The proposed change does not involve a modification of the design of plant structures, systems, or components. The change will not impact the manner in which the plant is operated and will not degrade the reliability of structures, systems, or components important to safety as equipment protection features will not be deleted or modified, equipment redundancy or independence will not be reduced, supporting system performance will not be affected, and no severe testing of equipment will be imposed. No new failure modes or mechanisms will be introduced as a result of this proposed change.

Therefore, the changes to the material surveillance program and pressure-temperature limits that compose this proposed change do not create the possibility of a new or different kind of accident than those previously evaluated.

3) Involve a significant reduction in a margin of safety.

The proposed implementation of the BWRVIP ISP has been previously evaluated generically by the NRC staff and was found to provide an acceptable alternative to plant-specific RPV material surveillance programs. The NRC staff also found that the ISP met the requirements of Appendix H to 10CFR50 for an integrated RPV material surveillance program.

Appendix G to 10CFR50 describes the conditions that require pressure-temperature (P-T) limits and provides the general bases for these limits. Operating limits based on the criteria of Appendix G, as defined by applicable regulations, codes, and standards, provide reasonable assurance that non-ductile or rapidly propagating failure will not occur. The P-T limits are not derived from design basis accident analyses (DBA); but, are prescribed for all plant modes to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause non-ductile failure of the reactor coolant pressure boundary. Calculation of P-T limits in accordance with the criteria of Appendix G to 10CFR50 and applicable regulatory requirements ensures that adequate margins of safety are maintained and there is no significant reduction in a margin of safety.

The proposed change does not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. There is no change or impact on any safety analysis assumption or in any other parameter affecting the course of an accident analysis supporting the Bases of any Technical Specification. The proposed change does not involve any increase in calculated off-site dose consequences. Since the proposed change for RPV material surveillance is in accordance with the NRC staff's safety evaluation for the ISP, and P-T curves were revised in accordance with the requirements of Appendix G to 10CFR50 (as modified by

use of ASME Code Case N-640), adequate safety margins are maintained without any significant reduction.

Conclusion

On the basis of the above, VY has determined that operation of the facility in accordance with the proposed change does not involve a significant hazards consideration as defined in 10CFR50.92(c), in that it: (1) does not involve a significant increase in the probability or consequences of an accident previously evaluated; (2) does not create the possibility of a new or different kind of accident from any accident previously evaluated; and (3) does not involve a significant reduction in a margin of safety.

Attachment 4

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 258

RPV Fracture Toughness and Material Surveillance Requirements

Revised Updated Final Safety Analysis Report

PROPOSED CHANGE 258 - PROPOSED UFSAR MARK-UP

1. VYNPS UFSAR, Affected Page List

   Current UFSAR Section 4.2.6 (pages 4.2-14 and 4.2-21)

2. Marked-up Pages

   See attached mark-up of UFSAR pages 4.2.14 and 4.2-21 (Table 4.2.4).

   Note: Deleted text is shown by strike-through. Added text is shown by underline.

fabrication and quality control organizations and a system capable of assuring and documenting the required quality level.

The qualifications are backed up with Rotterdam's extensive experience in core structure fabrication with such United States plants as TVA I, II, and III, Peach Bottom II and III, Monticello, and Vermont Yankee. Also, Rotterdam fabricated parts of Quad Cities II reactor pressure vessels, as well as complete vessels for foreign plants, such as AKM and Nuclenor.

The Reactor Coolant System was cleaned and flushed before fuel was loaded initially. During the preoperational test program, the reactor vessel and Reactor Coolant System were given a hydrostatic test in accordance with code requirements at 125% of design pressure. The vessel temperature is maintained at a minimum of 60°F above the NDT temperature prior to pressurizing the vessel for hydrostatic test. A system leakage test at a pressure not to exceed system operating pressure is made following each removal and replacement of the reactor vessel head. Other preoperational tests include calibrating and testing the reactor vessel flange seal-ring leakage detection instrumentation, adjusting reactor vessel stabilizers, checking all vessel thermocouples, and checking the operation of the vessel flange stud tensioner.

The reactor vessel temperatures are monitored during vessel heatup and cooldown to assure that thermal stress on the reactor vessel is not excessive during startup and shutdown.

4.2.6 Inspection and Testing

The plant has been designed to prevent occurrence of a gross defect. The inservice inspection program has been designed to provide for the inspection during service of those components and systems whose structural integrity must be maintained for continued safe operation of the plant. The selection of components and inspection locations is based on the ASME Code, Section XI, and 10CFR50.55(a). The program is presented in Reference 2.

Vermont Yankee is a participant in the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) for the purpose of monitoring changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region due to exposure of these materials to neutron irradiation. The Nuclear Regulatory Commission staff has determined that the BWRVIP ISP is an acceptable alternative to plant-specific material surveillance programs for the purpose of maintaining compliance with the requirements of Appendix H to 10CFR50, ''Reactor Vessel Material Surveillance Program Requirements.'' Under the ISP, dosimetry data and the results of fracture toughness tests from surveillance capsules in host BWRs are shared with comparable BWRs. As required by Appendix H to 10CFR50, VY will evaluate changes in the properties of representative materials for the purpose of determining whether changes are necessary in pressure and temperature limits and operating procedures. The report, ''BWRVIP-86-A: BWR Vessel and Internals Project Updated BWR Integrated Surveillance Program

(ISP) Implementation Plan,'' establishes the regulatory basis for the surveillance program.

The Vermont Yankee Nuclear Power Station is not a host ISP plant for providing surveillance capsules; however, the remaining two VYNPS material surveillance capsules will continue to reside in the reactor in case they are needed in the future as a contingency. The VYNPS surveillance capsules ~~Surveillance Test Program~~ consist of tensile and Charpy V-Notch specimens representative of the three areas of interest: reactor vessel base metal, weld Heat-Affected Zone (HAZ) metal, and weld metal from a reactor steel joint which simulates a welded joint in the reactor vessel. The specimens were placed in three separate surveillance ~~are contained in~~ capsules ~~placed at three locations in the reactor vessel~~ radially located adjacent to the inner vessel wall, ~~radially adjacent to the~~ at core mid-plane, where the neutron flux ~~will be~~ is highest. The specimen types contained in the capsules are listed in Table 4.2.4. In addition to the specimens listed in Table 4.2.4, sufficient specimens are provided for obtaining unirradiated base line data and for retention as archive material.

VY's neutron fluence calculations (and future re-evaluations) that support reactor coolant system pressure-temperature limits and the ISP are based on a fluence methodology that is acceptable to the NRC staff, consistent with the guidance in NRC Regulatory Guide 1.190, ''Calculational Methods for Determining Pressure Vessel Neutron Fluence.''

TABLE 4.2.4

SURVEILLANCE CAPSULE REMOVAL SCHEDULE

Capsule ~~Location~~ No.	Specimen Type (1)	Number of Specimens			Vessel Azimuth Location	Withdrawal Schedule (2) ~~(Note 1)~~
		Base	Weld	HAZ		
1	C~~*~~	12	12	12	30°	10 years (3)
	T~~*~~	2	2	2		
2	C	8	8	8	120°	~~30 years~~ Standby
	T	2	2	2		
3	C	8	8	8	300°	Standby
	T	2	2	2		

Notes:

(1) C = standard Charpy V-Notch impact specimen
T = tensile specimen

(2) Specified capsules will be withdrawn during the refueling outage following the year specified, referenced to the date of commercial operation.

(3) Capsule No. 1 was removed from the vessel for analysis in March 1983.

~~(Note 1) Specified capsules will be withdrawn during refueling outage following the year specified, referenced to the date of commercial operation.~~

~~* C = standard Charpy V-Notch impact specimen~~
~~* T = tensile specimen~~

Attachment 5

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 258

RPV Fracture Toughness and Material Surveillance Requirements

Marked-up Version of the Current Technical Specifications

Description of Technical Specification Changes

1. Delete TS SR 4.6.A.5 on current page 116 in its entirety.

2. Modify TS Figures 3.6.1, 3.6.2 and 3.6.3 (current pages 135-137) as follows:

   - The validity of each figure is changed from the "end of cycle 23" to "4.46 E8 MWH(t)."
   - For each figure, the grid line divisions are changed, additional 100 psi increments are added to the ordinate axis, and more data are used to plot the curves.
   - A Note is added to Figure 3.6.2 for the use of test instrumentation during tensioning and detensioning operations with the vessel vented and fluid level below the flange region.
   - Corrections are made to the tabulation of pressure and temperature values in Figure 3.6.3.

3. Replace the last sentence of the 4th paragraph on current page 138 – Bases to 3.6.A and 4.6.A – with the following:

   Based upon plate and weld chemistry, initial RT_{NDT} values, predicted peak fast neutron fluence (2.99×10^{17} n/cm^2 at the reactor vessel inside surface) for a gross power generation of 4.46×10^8 MWH(t), these core region ART_{NDT} values conservatively bound the guidance of Regulatory Guide 1.99, Revision 2.

4. Add amplifying clarification to the first sentence of the last paragraph on current page 139 – Bases 3.6.A and 4.6.A.

5. After the last paragraph on current page 139 – Bases 3.6.A and 4.6.A – insert the following two paragraphs:

   Specification 3.6.A.3 requires that the temperature of the vessel head flange and the head be greater than 70°F before tensioning. The 70°F is an analytical limit and does not include instrumentation uncertainty, which must be procedurally included depending upon which temperature monitoring instrumentation is being used. The temperature values shown on Figures 3.6.1, 3.6.2 and 3.6.3 include a 10°F instrumentation uncertainty.

   A Note is included in Figure 3.6.2 that specifies test instrumentation uncertainty must be +/- 2°F and the flange region temperatures must be maintained greater than or equal to 72°F when using such instrumentation in lieu of permanently installed instrumentation. Qualified test instrumentation may only be used for the purpose of maintaining the temperature limit when the vessel is vented and the fluid level is below the flange region. If permanently installed instrumentation (with a 10°F uncertainty) is used during head tensioning and detensioning operations, the 80°F limit must be met.

6. Delete the first paragraph on current page 140 – Bases to 3.6.A and 4.6.A.

7. Delete the current, last paragraph of Bases 3.6.A and 4.6.A (on current page 140), and replace it with the following:

   Vermont Yankee is a participant in the Boiling Water Reactor Vessel and Internals Project Integrated Surveillance Program (ISP) for monitoring changes in the fracture toughness properties of ferritic materials in the reactor pressure vessel (RPV) beltline region. (See UFSAR Section 4.2 for additional ISP details.) As ISP capsule test reports become available for RPV materials representative of VYNPS, the actual shift in the reference temperature for nil-ductility transition (RT_{NDT}) of the vessel material may be re-established. In accordance with Appendix H to 10CFR50, VY is required to review relevant test reports and make a determination of whether or not a change in Technical Specifications is required as a result of the surveillance data.

3.6 LIMITING CONDITIONS FOR 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.

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. 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.

CHANGE GRID LINE DIVISIONS AND USE MORE DATA TO PLOT CURVES

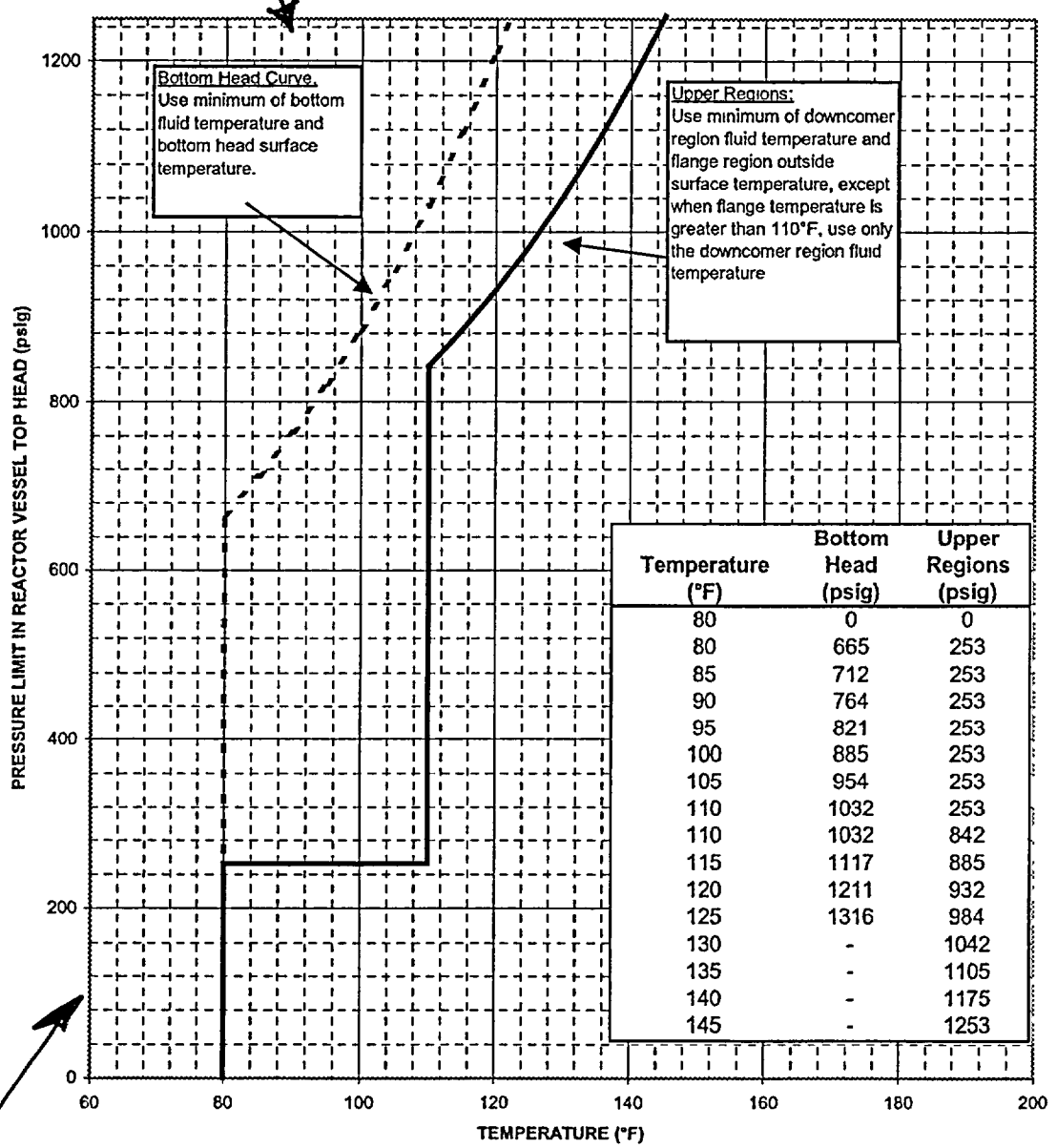
VYNPS

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 ~~End of Cycle 23~~

4.46 E8 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

SHOW 100 PSI INCREMENTS ON ORDINATE AXIS

CHANGE GRID LINE DIVISIONS AND USE MORE DATA TO PLOT CURVES

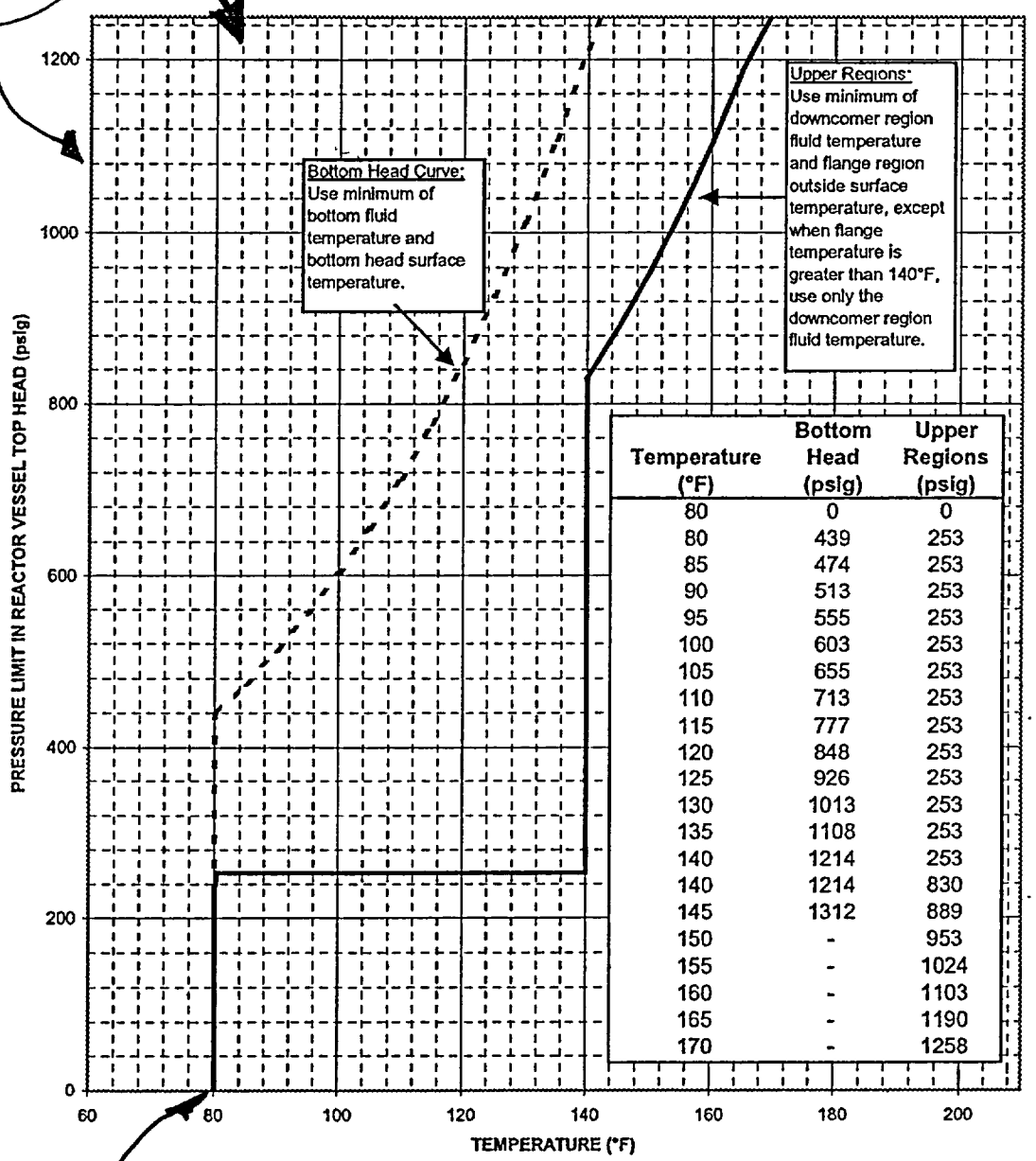
VYNPS

FIGURE 3.6.2

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

100°F/hr Heatup/Cooldown Limit
Valid Through ~~End of Cycle 23~~ 4.46 E8 MWH(t)

SHOW 100 PSI INCREMENTS ON ORDINATE AXIS



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

<INSERT>

During tensioning and detensioning operations with the vessel vented and the vessel fluid level below the flange region, the flange temperature may be monitored with test instrumentation in lieu of process instrumentation for the downcomer region fluid temperature and permanent flange region outside surface temperature. The test instrumentation uncertainty must be less than +/- 2°F. The flange region temperatures must be maintained greater than or equal to 72°F when monitored with test instrumentation during tensioning, detensioning, and when tensioned.

CHANGE GRID LINE DIVISIONS AND USE MORE DATA TO PLOT CURVES

VYNPS

FIGURE 3.6.3

Reactor Vessel Pressure-Temperature Limitations
Normal Operation, Core Critical

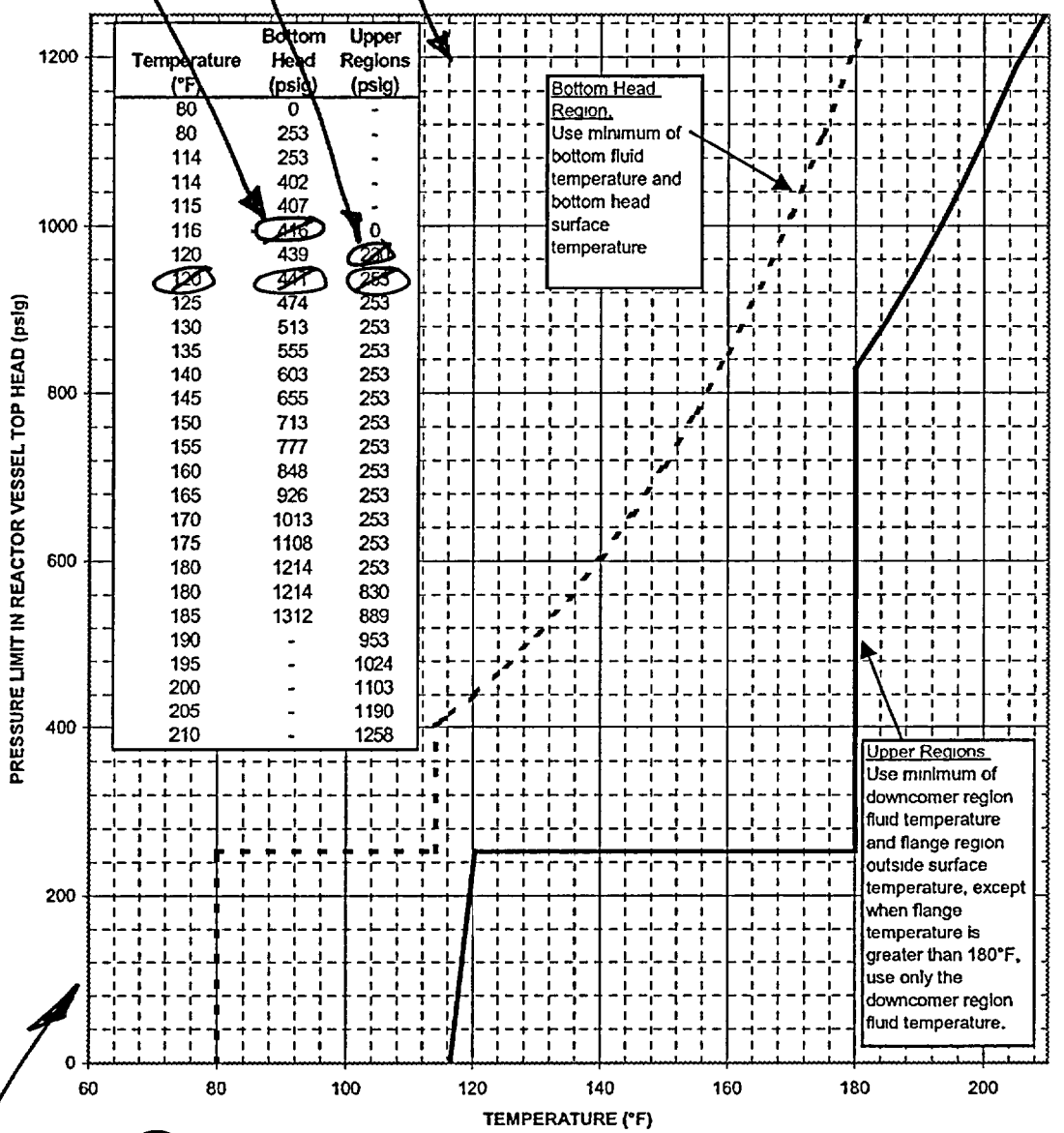
100°F/hr Heatup/Cooldown Limit
If Pressure < 253 psig, Water Level must be within Normal Range for Power Operation
Valid Through ~~End of Cycle 23~~

4.46 E8 MWH(t)

413

253

Temperature (°F)	Bottom Head (psig)	Upper Regions (psig)
80	0	-
80	253	-
114	253	-
114	402	-
115	407	-
116	416	0
120	439	220
120	447	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



SHOW 100 PSI INCREMENTS ON ORDINATE AXIS

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 Case 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.

INSERT

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). 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.

P/T (FIGURES 3.6.1, 3.6.2 AND 3.6.3)

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.

<INSERT>

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.

INSERT

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.

Attachment 6

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 258

RPV Fracture Toughness and Material Surveillance Requirements

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 included herein. The revised pages contain vertical lines in the margin indicating the areas of change.

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

3.6 LIMITING CONDITIONS FOR 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.

4.6 SURVEILLANCE REQUIREMENTS

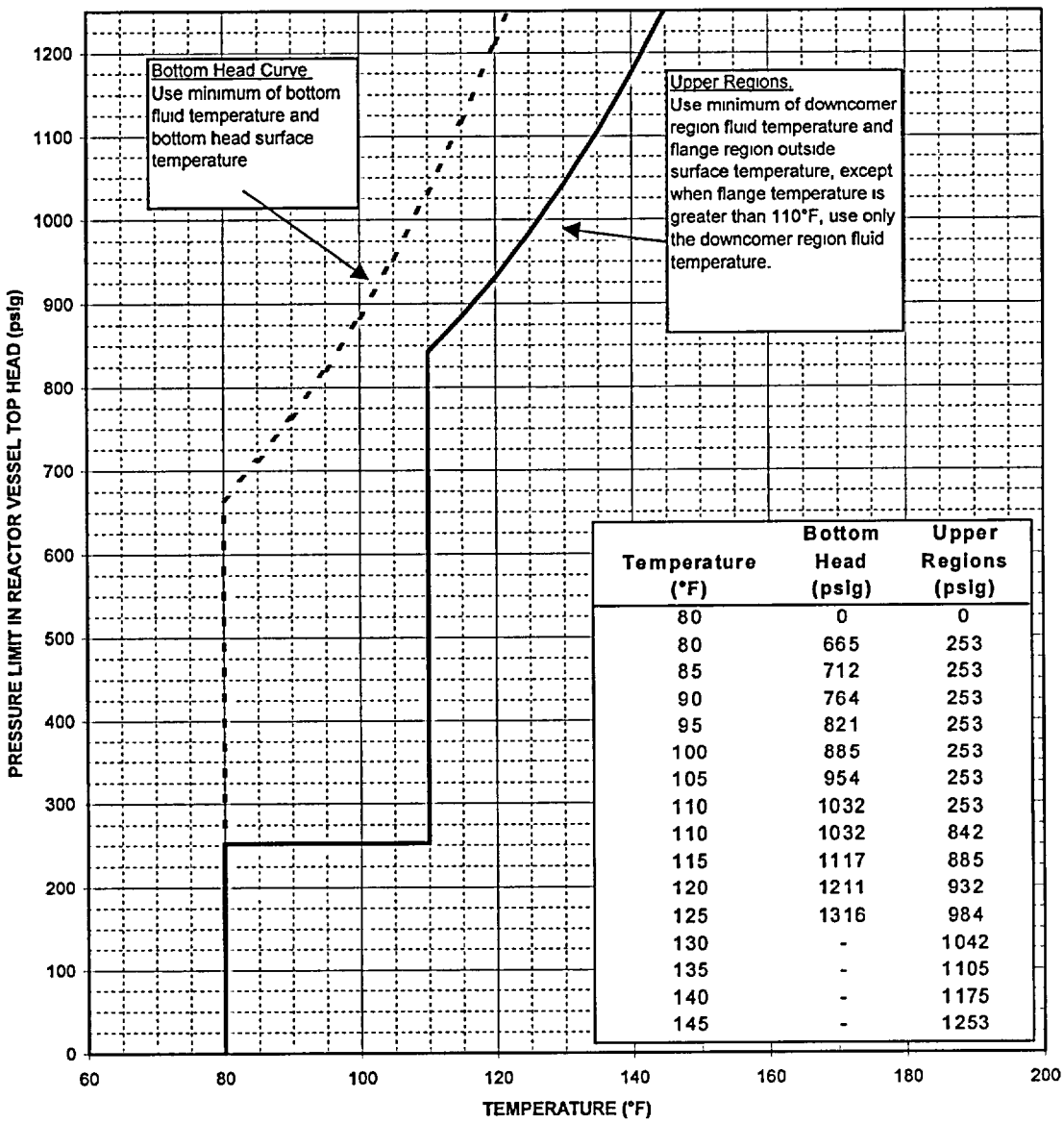
B. Coolant Chemistry

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)

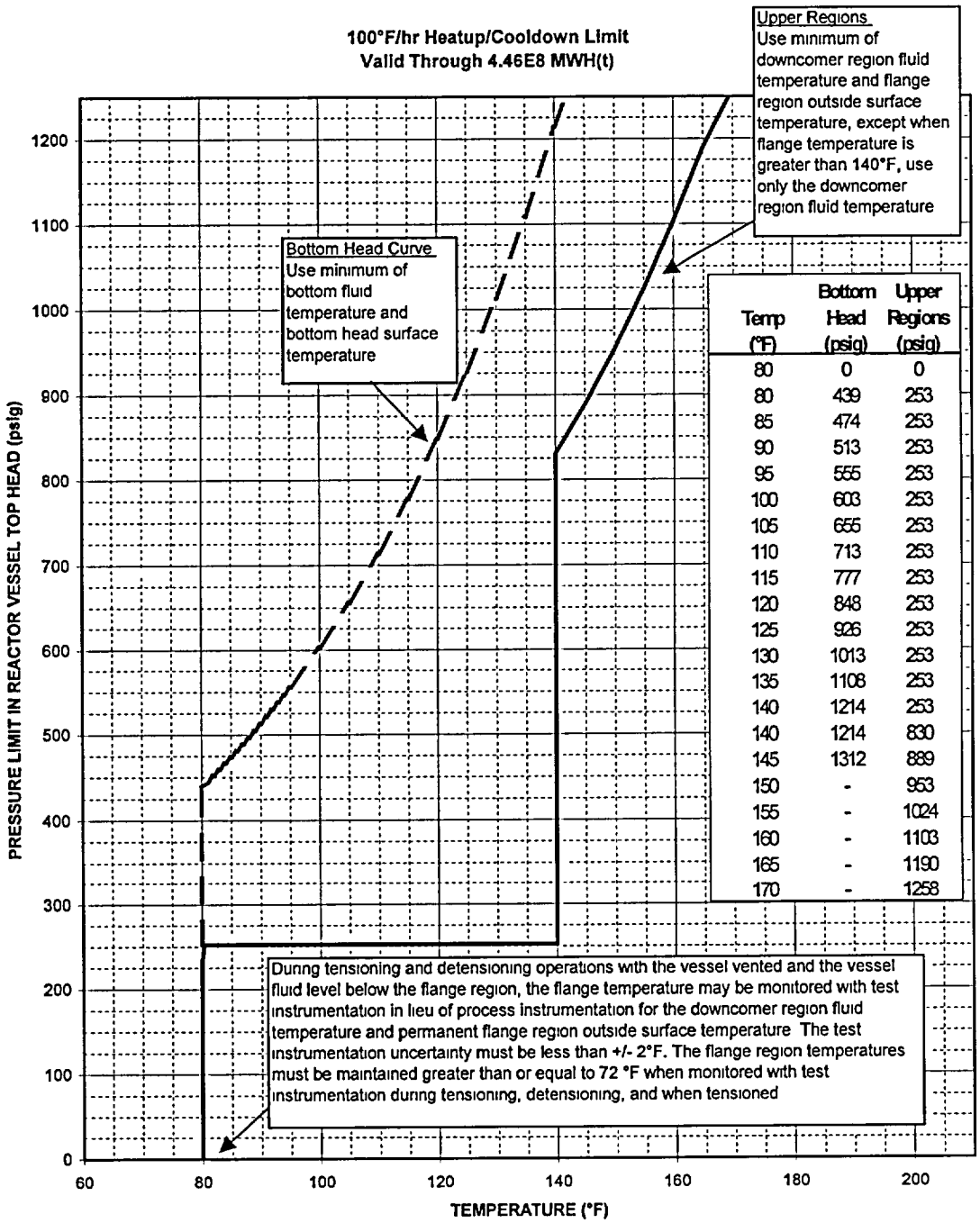


Temperature (°F)	Bottom Head (psig)	Upper Regions (psig)
80	0	0
80	665	253
85	712	253
90	764	253
95	821	253
100	885	253
105	954	253
110	1032	253
110	1032	842
115	1117	885
120	1211	932
125	1316	984
130	-	1042
135	-	1105
140	-	1175
145	-	1253

FIGURE 3.6.2

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

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

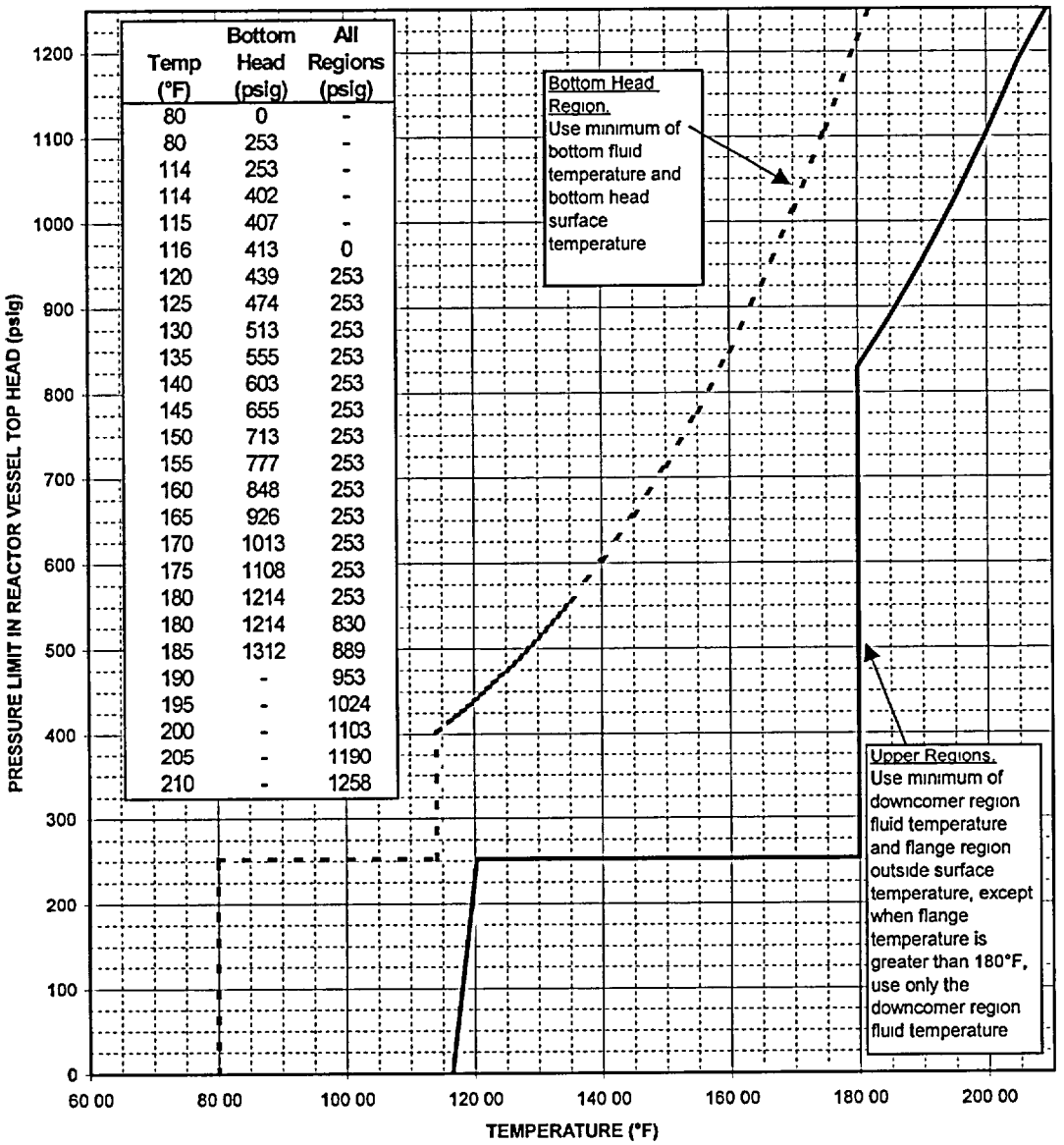


Temp (°F)	Bottom Head (psig)	Upper Regions (psig)
80	0	0
80	439	253
85	474	253
90	513	253
95	555	253
100	603	253
105	655	253
110	713	253
115	777	253
120	848	253
125	926	253
130	1013	253
135	1108	253
140	1214	253
140	1214	830
145	1312	889
150	-	953
155	-	1024
160	-	1103
165	-	1190
170	-	1258

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)



Temp (°F)	Bottom Head (psig)	All Regions (psig)
80	0	-
80	253	-
114	253	-
114	402	-
115	407	-
116	413	0
120	439	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 Case 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 fast neutron fluence (2.99×10^{17} n/cm^2 at the reactor vessel inside surface) for a gross power generation of 4.46×10^8 MWH(t), 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.

BASES: 3.6 and 4.6 (Cont'd)

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). 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 P/T curves (Figures 3.6.1, 3.6.2 and 3.6.3). 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.

Specification 3.6.A.3 requires that the temperature of the vessel head flange and the head be greater than 70°F before tensioning. The 70°F is an analytical limit and does not include instrumentation uncertainty, which must be procedurally included depending upon which temperature monitoring instrumentation is being used. The temperature values shown on Figures 3.6.1, 3.6.2 and 3.6.3 include a 10°F instrumentation uncertainty.

BASES: 3.6 and 4.6 (Cont'd)

A Note is included in Figure 3.6.2 that specifies test instrumentation uncertainty must be +/- 2°F and the flange region temperatures must be maintained greater than or equal to 72°F when using such instrumentation in lieu of permanently installed instrumentation. Qualified test instrumentation may only be used for the purpose of maintaining the temperature limit when the vessel is vented and the fluid level is below the flange region. If permanently installed instrumentation (with a 10°F uncertainty) is used during head tensioning and detensioning operations, the 80°F limit must be met.

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.

Vermont Yankee is a participant in the Boiling Water Reactor Vessel and Internals Project Integrated Surveillance Program (ISP) for monitoring changes in the fracture toughness properties of ferritic materials in the reactor pressure vessel (RPV) beltline region. (See UFSAR Section 4.2 for additional ISP details.) As ISP capsule test reports become available for RPV materials representative of VYNPS, the actual shift in the reference temperature for nil-ductility transition (RT_{NDT}) of the vessel material may be re-established. In accordance with Appendix H to 10CFR50, VY is required to review relevant test reports and make a determination of whether or not a change in Technical Specifications is required as a result of the surveillance data.

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.